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

April 28, 2005

Duke Energy Corporation
ATTN: Mr. G. R. Peterson
Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000369/2005002 AND 05000370/2005002 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 0720038/20050001

Dear Mr. Peterson:

On March 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your McGuire Nuclear Station. The enclosed report documents the inspection findings which were discussed on April 5, 6, and 19, with you and members of your staff.

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

Based on the results of this inspection, there were five findings (four NRC-identified and one self-revealing) of very low safety significance (Green) identified in the report which were determined to be violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. In addition, the NRC identified a Severity Level IV violation of 10 CFR 50.73 for failure to report a condition prohibited by Technical Specifications related to past inoperability of main steam isolation valve 1SM-1. It was determined that this violation should also be non-cited in accordance with Section VI.A of the NRC's Enforcement Policy. Furthermore, one licensee-identified violation which was determined to be of very low safety significance (Green) is listed in Section 4OA7 of this report. If you contest the non-cited violations in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the McGuire facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael Ernstes, 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 05000369/2005002, 05000370/2005002, and 0720038/20050001 w/Attachment - Supplemental Information

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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 Nos:	05000369/2005002, 05000370/2005002, 0720038/2005001
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station, Units 1 and 2
Location:	12700 Hagers Ferry Road Huntersville, NC 28078
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	J. Brady, Senior Resident Inspector S. Walker, Resident Inspector G. Kuzo, Senior Health Physicist (Sections 2OS1, 4OA1, 4OA5.2, and 4OA7) H. Gepford, Health Physicist (Sections 2PS1 and 4OA1) A. Nielsen, Health Physicist (Sections 2OS3, 2PS3, and 4OA1) J. Fuller, Reactor Inspector (Sections 1R08 and 4OA5.3) S. Vias, Senior Reactor Inspector (Sections 1R08 and 4OA5.3)
Approved by:	Michael Ernstes Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR05000369/2005002, IR05000370/2005002, and 0720038/2005001; 01/01/2005 - 03/31/2005; McGuire Nuclear Station, Units 1 and 2; Operability Evaluations, Surveillance Testing, Access Controls to Radiologically Significant Areas, Radiation Monitoring Instrumentation and Protective Equipment, Event Followup, and Other Activities.

The report covered a three month period of inspection by resident inspectors and announced inspections by regional health physics and reactor inspectors. Six non-cited violations (NCV) were identified; five Green and one Severity Level (SL) IV. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A non-cited violation of Technical Specification (TS) 3.4.15, Reactor Coolant System (RCS) Leakage Detection Instrumentation, was identified by the inspectors for failing to take actions required for containment radiation gaseous monitors being inoperable. Specifically, the monitors were unable to detect a 1 gpm RCS leak in 1 hour due to current activity concentrations (i.e., < 0.1 percent failed fuel) and TS required Actions B.1 (24-hour containment atmosphere sample) or B.2 (24-hour RCS water inventory balance) were not performed.

  The finding is greater than minor because the containment particulate and gas channel radiation monitors were not capable of performing the design bases function of alerting control room operators of elevated reactor coolant system unidentified leakage, for an extended period of time. This inoperability resulted in a potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1 gpm was indicated through the reactor coolant system water balance surveillance. This issue contained elements of problem identification and resolution, as well as human performance, in that licensee operations and engineering personnel determined the radiation monitors to be operable without consideration of all available information. (Section 1R15)

- Green. A non-cited violation of TS 5.4.1.a was identified by the inspectors for failing to establish, implement, and maintain adequate Reactor Coolant System Leakage Detection Instrumentation surveillance procedures for surveillance requirement (SR) 3.4.15.2, channel operational test of containment atmosphere radioactivity monitor; SR 3.4.15.3, channel calibration of containment floor and equipment sump (F&ES) level monitoring system; SR 3.4.15.4, channel calibration of containment atmosphere radioactivity monitor; and SR 3.4.15.5, channel calibration of containment ventilation condensate drain tank (VCDT) level monitor. Procedures for containment radiation particulate and gas monitors had not set the alarms to leakage values equivalent to 1 gallon per minute in 1 hour and had not tested the end device used by the operators to provide alarm indication of potentially excessive reactor coolant system unidentified leakage for multiple containment leakage monitors, including level indication (F&ES and VCDT) and radiation monitors.

The finding was greater than minor because the surveillance procedures had not provided assurance that the necessary quality of systems or components were maintained. Consequently, this resulted in a potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was of very low safety significance because excessive leakage had not existed based on reactor coolant inventory water balances and that the alarm indication functioned properly when tested. This issue contained elements of problem identification and resolution, in that the licensee's operability determination failed to adequately evaluate whether surveillance requirements had been met and actions to determine the "time to alarm" given current RCS activity levels were not prompt. (Section 1R22b.(1))

Cornerstone: Mitigating Systems

- Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, was identified by the inspectors for failing to take timely and adequate corrective actions to resolve adverse conditions that resulted in two Unit 1 main steam isolation valves (MSIVs) being inoperable beyond their Technical Specification allowed out-of-service time.

  The finding is considered greater than minor because it had a direct impact on the MSIVs' ability to perform their safety function, which is to close during a high energy line break or steam generator tube rupture. The finding affects both the Mitigating Systems and Barrier Integrity cornerstones, in that the failure to close impacts the equipment performance (reliability, availability) attribute and containment isolation (minimization of radiological releases) attribute, respectively. Based on the results of the Phase 3 SDP analysis, the finding is considered of very low safety significance. This issue contained elements of problem identification and resolution, as well as human performance, as it involved failures to properly evaluate data and deficiencies associated with the MSIVs; therefore, failing to take prompt corrective action to preclude the valves from becoming inoperable. (Section 4OA5.4)

C SLIV. A non-cited violation was identified by the inspectors for failure to report a condition prohibited by Technical Specifications related to past inoperability for main steam isolation valve 1SM-1, as required by 10 CFR 50.73.

Based on the very low safety significance of the technical issue, this violation is categorized as a Severity Level IV violation under the NRC Enforcement Policy, Supplement I. (Section 4OA3.1)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a non-cited violation of Technical Specification 5.4.1(a) for failure to follow radiation protection procedures used to demonstrate compliance with 10 CFR Parts 20 and 72. Specifically, on August 24, 2004, Independent Spent Fuel Storage Installation (ISFSI) area dose rate surveys were conducted using portable radiation monitoring instrumentation, a RO-20 ion chamber survey meter, which did not cover the lower range of radiation levels expected (i.e., less than 0.05 millirem per hour), for selected boundary trending points. Further, the dose rate values documented (i.e., less than 0.1 mrem/hr) for the subject trending point locations, did not allow verification that the established procedural limits used to demonstrate compliance with 10 CFR Parts 20 and 72 requirements were met.

  This finding is more than minor in that the failure to accurately monitor and properly evaluate the quarterly dose rate results could prevent identification of unexpected/elevated dose rates associated with ISFSI operations and is associated with the Program and Process attribute of the Occupational Radiation Safety Cornerstone. The finding affects the cornerstone objective to prevent/minimize radiation exposure to personnel. The issue is of very low safety significance because the procedurally established dose rate limits are based on conservative occupancy factors, and results of proper dose rate surveys conducted prior and subsequent to the subject date were within established dose rate limits. (Section 2OS1)

- Green. A self-revealing non-cited violation of 10 CFR 20.1703(e) was identified for use of inadequate in-service breathing air (VB) system equipment to supply ‘Delta Suit’ respiratory protective equipment. Specifically, on March 25, 2004, available VB system capacity was inadequate to supply adequate air flow to six workers using supplied-air ‘Delta Suits’ for steam generator (SG) work activities.

  The finding is more than minor in that it is associated with the Occupational Radiation Safety Cornerstone Plant Equipment and Instrumentation attribute and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radioactive material during routine civilian nuclear reactor operations. The issue is of very low safety significance because the flow monitoring equipment used to identify degraded or failed VB system operations alerted responsible staff. The subject SG workers immediately ceased work activities and exited the work area without any unexpected internal contamination or resultant doses. (Section 2OS3)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. The violation and the corrective action tracking number is listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status:

Unit 1 began the inspection period at approximately 100 percent rated thermal power (RTP) and remained there.

Unit 2 began the inspection period at 100 percent RTP. On March 1, 2005, a steam leak on the 2B2 Moisture Separator Reheater (MSR) 2nd stage vent line caused an early shutdown for the scheduled Unit 2 end-of-cycle 16 (2EOC16) refueling outage. The unit remained in the refueling outage through the end of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns

During this inspection period, the inspectors performed the following four partial system walkdowns, while the indicated structures, systems, and components (SSCs) were out of service for maintenance and testing:

- Unit 1 train A Component Cooling (KC) with train B out of service on January 4, 2005
- Unit 2 train B Chemical and Volume Control System (CVCS) and High Head Safety Injection (HHSI) System with train A out of service on January 11, 2005
- Unit 1 train B Emergency Diesel Generator (EDG) with train A out of service on January 18, 2005
- Unit 2 train A and B Motor Driven Auxiliary Feedwater (MDCA) Pumps with Turbine Driven Pump out of service on February 23, 2005

To evaluate the operability of the selected trains or systems under these conditions, the inspectors verified correct valve and power alignments by comparing observed positions of accessible valves, switches, and electrical power breakers to the procedures and drawings listed in the Attachment to this report. In addition, the inspectors used the operator aid computer to determine whether system parameters were as expected for the system and plant conditions, and whether equipment status shown for inaccessible equipment supported operability of the system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the seven areas identified below, the inspectors reviewed the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures, to verify that those items were consistent with UFSAR Section 9.5.1, Fire Protection System, and the fire protection program as described in the Design Basis Specification for Fire Protection (MCS-1465.00-00-0008). The inspectors walked down accessible portions of each area, as well as reviewed results from related surveillance tests, and reviewed the associated pre-fire plan strategy to verify that conditions in these areas were consistent with descriptions of the areas in the Design Basis Specification. Documents reviewed during this inspection are listed in the Attachment to this report.

In addition, the inspectors observed a fire drill on February 26 for the 'D' shift in accordance with annual requirements prescribed in IP 71111.05. The inspectors evaluated the licensee's critique following the drill to determine if the licensee's evaluation was adequate.

The inspected areas included:

- Diesel Generator 1A Room (fire area 5)
- Diesel Generator 1B Room (fire area 6)
- Diesel Generator 2A Room (fire area 7)
- Diesel Generator 2B Room (fire area 8)
- U2 Reactor Building Pipe Chase (fire area RB2)
- U2 Reactor Building Lower Containment (fire area RB3)
- U2 Reactor Building Annulus (fire area RB1)

b. Findings

Introduction: The inspectors identified a finding for failure to establish, implement, and maintain adequate procedures to implement fire protection sprinkler inspection requirements for the reactor building annulus contained in Updated Final Safety Analysis Report (UFSAR) Chapter 16, Selected Licensee Commitments (SLC). It has been identified as an Unresolved Item (URI), pending risk significance determination.

Description: On March 16, 2005, the inspectors found that several fire protection sprinklers in the Unit 2 reactor building annulus area had their spray patterns obstructed by cables in the cable trays immediately below them. The sprinkler heads were located in the upper level of cables. The inspectors found that UFSAR Chapter 16, SLC section 16.9.2 covers spray and/or sprinkler systems listed in Table 16.9.2-1, which includes the Reactor Building Annulus. Testing Requirement (TR) 16.9.2.6 requires that these sprinklers be inspected by a visual inspection every 18 months to ensure that each nozzle's spray pattern was not obstructed. The inspectors found that the licensee's implementation of that requirement had not identified these deficiencies. The licensee's investigation found a total of six sprinklers in the annulus area that were obstructed.

Analysis: The finding is greater than minor because it is associated with an increase in the likelihood of an initiating event. In the event of a Unit 2 annulus fire, the cables affected by the obstructed sprinklers include those which could cause all four reactor coolant pumps to trip, consequently causing a reactor trip. The significance of this issue is undetermined pending completion of the Significance Determination Process Phase 2 in accordance with MC 0609, Appendix F, Fire Protection SDP.

Enforcement: Technical Specification (TS) 5.4.1.d requires that written procedures be established, implemented and maintained covering activities for commitments contained in the UFSAR Chapter 16.0, Selected Licensee Commitments. Selected Licensee Commitment 16.9.2 covers spray and/or sprinkler systems listed in Table 16.9.2-1, which includes the Reactor Building Annulus. Testing Requirement 16.9.2.6 requires that every 18 months a visual inspection be performed of each nozzle's spray area to verify the spray pattern is not obstructed. Contrary to the above, prior to March 16, 2005, the licensee failed to adequately implement TR 16.9.2.6 in the annulus area, in that six sprinklers had their spray patterns obstructed by cables in the cable trays because the sprinkler heads were located too close to the tray. Pending completion of the Significance Determination Process, Phase 2, this issue is identified as URI 05000370/2005002-01, Failure to Have Adequate Procedures to Implement SLC Test Requirements for Fire Protection Sprinklers. This issue is captured in the licensee's corrective action program under Problem Investigation Process report (PIP) M-05-01463.

1R07 Heat Sink Performance

a. Inspection Scope

Annual Inspection

The inspectors reviewed the inspection pictures of the 2A and 2B component cooling water heat exchangers taken during the current refueling outage (2EOC16), to verify that the inspection results were appropriately categorized against the pre-established acceptance criteria described in procedure MP/0/A/7700/013, Component Cooling System Heat Exchanger Corrective Maintenance. The inspectors also verified that the frequency of inspection (each refueling outage) was sufficient to detect degradation prior to loss of heat removal capability below design basis values by comparing the current inspection pictures to the previous inspection pictures from refueling outages 2EOC15 (fall 2003) and 2EOC14 (spring 2002).

The inspectors reviewed the following PIP associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- M-05-1364, Tube plugging required in 2B component cooling water and 2B diesel generator cooling water heat exchangers as a result of eddy-current inspections

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems ISI

a. Inspection Scope

From March 8 - 25, 2005, the inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Unit 2. The inspectors selected a sample of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required examinations and Code components in order of risk priority as identified in Section -03 of inspection procedure 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate compliance with TS, ASME Section XI, and ASME Section V requirements, 1998 Edition through 2000 Addenda, and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB-3000 or IWC-3000 acceptance standards.

Specifically, the inspectors observed the following examinations:

Ultrasonic Testing:
- 2NC2FW16-1, Reactor Coolant System (RCS) 10" elbow to nozzle similar metal weld, ASME Class 1

Visual Testing:
- 2PZR-W3SE, Pressurizer Relief Nozzle, ASME Class 1
- 2PZR-W4ASE, Pressurizer Safety Nozzle, ASME Class 1
- 2PZR-W4BSE, Pressurizer Safety Nozzle, ASME Class 1
- 2PZR-W4CSE, Pressurizer Safety Nozzle, ASME Class 1
- 2PZR-W11, Pressurizer Manway Insert, ASME Class 1
- 2MCR-NC-4791, Mechanical Snubber for RCS, ASME Class 1

Liquid Penetrant Testing:
- 2NC2FW16-1, RCS 10" elbow to nozzle weld, ASME Class 1

Specifically, the inspectors reviewed the following examination records:

Ultrasonic Testing:
- NV2FW32-38, 2NV-217 4" Valve Replacement, Chemical and Volume Control System, ASME Class 2
- 2SGC-INLET, SG2C Inlet Nozzle Inner Radius Examination, ASME Class 1
- 2SGC-OUTLET, SG2C Outlet Nozzle Inner Radius, ASME Class 1

Visual Testing (VT):
- 2PZR-W2SE, Pressurizer Spray Nozzle, ASME Class 1

Liquid Penetrant Testing:
- 2NV2FW21-33, 2" Stainless steel tee to reducing insert, ASME Class 2
- 2RPV-CRDM-71, Control Rod Drive Mechanism (CRDM) Adapter to Housing Body, ASME Class 1
- 2PZR-W13C, Pressurizer Lug to Shell, ASME Class 1

The Inspectors reviewed examination records for the following recordable indications to evaluate if the licensee's acceptance was in accordance with acceptance standards contained in Article IWB-3000 of ASME Section XI.

Ultrasonic Testing:
- 2RPV-W01, Reactor Pressure Vessel (RPV) Lower Head to Bottom Head Circumferential Weld, second 10 year In-Vessel ISI (IV-ISI), ASME Class 1
- 2RPV-W18, RPV Outlet Nozzle to Shell Weld @ 338E, second 10 year IV-ISI, ASME Class 1
- 2NI2FW26-15, Elbow to Valve, coverage limitation, ASME Class 2

Liquid Penetrant Testing:
- 2NV2FW21-33, 2" Stainless steel tee to reducing insert, ASME Class 2
- 2PZR-W13C, Pressurizer Lug to Shell, ASME Class 1

The inspectors reviewed the "McGuire Nuclear Station, Unit 2, Inservice Inspection Report, EOC15 Refueling Outage," dated December 15, 2003, which stated that there were no reportable indications from last outage. The report did list the results of examinations and the inspectors reviewed documentation for an ASME Class 2 Hanger, 2MCA-ND-5503, which failed a VT-3 examination. The inspectors verified that the unacceptable condition was adequately repaired and entered into the licensee's corrective action program.

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the above ISI examination activities were reviewed and compared to requirements stated in ASME Section V and Section XI.

A sample of pressure boundary welding activities associated with ASME Class 1 and Class 2 components were reviewed, to verify the welding process and examinations were performed in accordance with the ASME Code Sections III, V, IX, and XI requirements. The inspectors reviewed weld data sheets, the welding procedure specification (WPS), supporting welding procedure qualification records (PQR), welder qualification records, and preservice examination results for the following welds:

- NV2FW32-38, 2NV-217 4" ASME Class 2 Valve Replacement, Butt Weld
- NV2FW80-9, 2NV-217 4" ASME Class 2 Valve Replacement, Socket Weld

The inspectors performed a review of piping system ISI related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From March 8-25, 2005, the inspectors reviewed the licensee's BACC program to ensure compliance with commitments made in response to NRC Generic Letter 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary" and Bulletin 2002-01 "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

The inspectors conducted an on-site record review, as well as an independent walk-down of parts of the reactor building that are not normally accessible during at-power operations to evaluate compliance with licensee BACC program requirements and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In particular, the inspectors verified that the visual examinations focused on locations where boric acid leaks can cause degradation of safety significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed a sample of engineering evaluations completed for boric acid found on reactor coolant system piping and components to verify that the minimum design code required section thickness had been maintained for the affected component(s). The inspectors also reviewed licencee corrective actions implemented for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code and 10 CFR 50 Appendix B Criterion XVI. Specifically, the inspectors reviewed:

- Work Request (WR) 98716766, 2KF-114 Repair leaking valve, Spent Fuel Pool Cooling
- PIP M 04-04091, 1A Residual Heat Removal (ND) Pump mini flow pressure switch has active fitting leak
- PIP M 04-04090, 1A Spent Fuel Cooling Pump has outboard seal and gland leak

b. Findings

No findings of significance were identified.

.3 Steam Generator (SG) Tube ISI

a. Inspection Scope

From March 22 - 25, 2005, the inspectors reviewed the Unit 2 SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements.

The inspectors reviewed the SG examination scope, expansion criteria, eddy current testing (ET) acquisition procedures, ET analysis procedures, the SG Operational Assessment, in-situ tube pressure testing procedures and records and examination reports to confirm that:

- In-situ SG tube pressure testing screening criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620 “Steam Generator In Situ Pressure Test Guidelines,” and the licensee’s screening criteria included allowances for ET probe flaw sizing error bands.

- The numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee’s previous outage Operational Assessment predictions. Also, the SG tube ET examination scope and expansion criteria was sufficient to identify tube degradation based on site and industry operating experience by confirming that the ET scope completed was consistent with the licensee’s procedures and plant TS requirements. Additionally, the inspectors reviewed the SG tube ET examination scope to determine that it was consistent with that recommended in EPRI 1003138 “Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6", and included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, U-bends and support plates.

- The SG tube repair criteria and process (plugging) implemented was consistent with TS requirements and that the licensee was only applying the TS plugging limit at tube wear locations.

- The ET probes and equipment configurations used to acquire ET data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H “Performance Demonstration for Eddy Current Examination” of EPRI 1003138.

- The licensee adequately examined for loose parts indications.

- The licensee adequately evaluated for any contractor deviations from their ET data acquisition or analysis procedures or EPRI 1003138.

The inspectors performed a review of SG ISI related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors’ review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requirements.

b. Findings

No findings of significance were found.

1R11 Licensed Operator Requalification

a. Inspection Scope

On January 13, 2005, the inspectors observed licensed-operator performance during requalification simulator training for shift ‘A’ to verify that operator performance was consistent with expected operator performance, as described in Exercise Guide OP-MC-SRT-02 and 51. This training tested the operators’ ability to perform abnormal and emergency procedures dealing with reactor trip response to a steam leak, a failed safety relief valve, recognition and isolation of a faulted steam generator and the effects of an inadvertent safety injection signal, and loss of residual heat removal while in mid-loop. The inspectors focused on clarity and formality of communication, use of procedures, alarm response, control board manipulations, group dynamics and supervisory oversight. The inspectors observed the post-exercise critique, to verify that the licensee identified deficiencies and discrepancies that occurred during the simulator training.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two degraded SSC/function performance problems or conditions listed below, to verify the licensee’s appropriate handling of these performance problems or condition in accordance with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, and 10 CFR 50.65, Maintenance Rule.

- Moisture Separator Reheater Drain System (HS)
- Flow Accelerated Corrosion Program effectiveness for secondary steam systems

The inspectors focused on the following:

- Appropriate work practices
- Identifying and addressing common cause failures
- Scoping in accordance with 10 CFR 50.65(b)
- Characterizing reliability issues (performance)
- Charging unavailability (performance)
- Trending key parameters (condition monitoring)

- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification, and
- Appropriateness of performance criteria for SSCs/functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified (a)(1)

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 risk assessments and the risk management actions used to manage risk for the plant configurations associated with the five activities listed below. The inspectors assessed whether the licensee performed adequate risk assessments, and implemented appropriate risk management actions when required by 10 CFR 50.65(a)(4). For emergent work, the inspectors also verified that any increase in risk was promptly assessed, and that appropriate risk management actions were promptly implemented. The inspectors also reviewed associated PIPs to verify that the licensee identified and implemented appropriate corrective actions.

- Work for week of January 10, 2005, including emergent work on Unit 1 Halon fire extinguishing system that caused an unplanned Unit 1 letdown isolation.

- Work for week of January 17, 2005, including: emergent work on the Standby Shutdown Facility Diesel Generator due to its output breaker tripping on reverse power resulting in the diesel failing its operability test; and a Unit 1 rod control urgent failure alarm on shutdown bank control cabinet SCDE.

- Work for week of January 30, 2005, including emergent work from the previous week's Unit 1 rod control urgent failure alarm; compensatory measures put in place for an inadequate design of the Unit 2 incore instrument room hatch that allows diversion of emergency core cooling system (ECCS) water into the room that included increasing spent fuel pool level and procedure revisions (PIP M-04-5049), and also resulted in risk management changes and schedule revisions (PIP M-05-0424); and emergent work and risk management due to a rod control non-urgent failure alarm on Unit 2 power cabinet 2BD.

- Work for week of February 20, 2005, including planned work and emergent nuclear service water discharge header air vent problems that caused the inoperability of the Unit 1 A train auxiliary feedwater pump.

- Work for week of February 27, 2005, including: planned work on 1B nuclear service water pump and 1B emergency diesel generator; and emergent work from a Unit 2 steam leak that caused an early entry into a refueling outage, subsequent 2SM-1 MSIV failure to close and resulting forced early mode change.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

During the non-routine evolutions identified below, the inspectors observed plant instruments and operator performance to verify that the operators performed in accordance with the associated procedures and training.

- Unit 2 MSR vent line break caused entry into AP-01, “Steam Leak”, and subsequently AP-04, “Rapid Downpower”

- Unit 2 Shutdown for EOC16 refueling outage

Following the non-routine evolution identified below, the inspectors reviewed operator logs, plant computer data, and other plant records, to determine what occurred and how the operators responded, and to verify that the response was in accordance with the associated procedures and training.

- Unit 1 rod control urgent failure alarm on shutdown bank control cabinet SCDE that caused entry into abnormal procedure AP-14, “Rod Control Malfunction.”

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability determinations the licensee had generated that warranted selection on the basis of risk insights. The selected samples are addressed in the PIPs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of any necessary compensatory measures, and compliance with the TS. The inspectors verified that the operability determinations were made as specified by Nuclear System Directive (NSD) 203, “Operability.” The inspectors compared the arguments made in the determination to the requirements from the TS, the UFSAR, and associated design-basis documents to verify that operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred.

- M-04-05579, 1CA-42B has Slight Leak by the Seat
- M-04-05049, Unit 2 Incore Room Hatch Leakage, Rev. 0
- M-04-5592, Reactor Coolant System Leak Detection Airborne Radioactivity Monitoring

- M-04-05049, Unit 2 Incore Room Hatch Leakage, Rev. 1
- M-04-05611, Incore Room Leakage Detection, Rev. 0

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) of TS 3.4.15 for failing to take the required actions for containment radiation gaseous monitors being inoperable.

Description: On February 10, 2005, the inspectors found that the licensee had identified in PIP M-04-5592 that the containment particulate and gaseous monitors that are used to implement RCS Leakage Detection Instrumentation TS 3.4.15 had alarms set at values that will not indicate a 1 gallon per minute leak in 1 hour. TS 3.4.15 is based on Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, which contains the 1 gpm in 1 hour value, and indicated that in analyzing the sensitivity of leak detection systems using airborne particulate or gaseous radioactivity, a realistic primary coolant radioactivity concentration assumption should be used. The licensee's PIP identified on December 10, 2004, that the current alarm values are based on 0.1% failed fuel and would not detect a 1 gpm leak in 1 hour at current coolant activity concentrations. Consequently, the inspectors concluded that the alarm function was inoperable since it was not based on a realistic current activity value. The alarm function is what the operators use for indication that RCS leakage should be evaluated under TS 3.4.13, RCS Operational Leakage, Limiting Conditions for Operation (LCO) to determine whether unidentified leakage of greater than 1 gpm exists. The inspectors found that the licensee had not taken any actions to compensate for this deficiency and had not taken the LCO actions required by TS 3.4.15 for the gaseous radioactivity monitor.

Analysis: The finding is greater than minor because the containment particulate and gas channel radiation monitors were not capable of performing the design bases function of alerting control room operators of elevated RCS unidentified leakage for an extended period of time. This inoperability resulted in a potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was of very low safety significance (Green) because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1gpm was indicated through the RCS water balance surveillance. The unavailability of the gaseous and particulate channel leak detection alarm functions did not contribute to an increase in core damage sequences when evaluated using the significance determination phase 1 worksheets. This issue contained elements of problem identification and resolution, as well as human performance, in that licensee operations and engineering personnel determined the radiation monitors to be operable without consideration of all available information.

Enforcement: TS 3.4.15, "RCS Leakage Detection Instrumentation", LCO requires that RCS leakage detection instrumentation shall be operable, including one containment atmosphere gaseous activity monitor and the containment atmosphere particulate radioactivity monitor. When the LCO is not met for the containment gaseous radioactivity monitor being operable, actions B.1 or B.2 must be performed. Action B.1

requires that grab samples of containment atmosphere be taken every 24 hours; B.2 requires that SR 3.4.13.1 be performed every 24 hours (RCS water inventory balance). For the particulate monitor, as long as the containment ventilation drain tank level monitor is available, no other actions are required, otherwise the particulate monitor must be restored within 30 days. Contrary to the above, from December 15, 2004, until February 12, 2005, the licensee failed to set the alarm function such that a 1 gpm leak of reactor coolant could be detected in 1 hour and failed to take the actions required by either B.1 or B.2. Because this issue was of very low safety significance and was placed in the corrective action program under PIPs M-05-857 and M-05-902 , this violation is being treated as a NCV in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000369,370/200502-02, Failure to Comply with RCS Leakage Detection TS for Containment Radiation Gaseous Monitors.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the operator work-arounds listed below that warranted selection on the basis of risk insights, to verify that these work-arounds did not affect either the functional capability of the related system in responding to an initiating event, or the operators' ability to implement abnormal or emergency operating procedures. The selected samples are listed below:

- OWA 04-09; Gradual downward trend in service water flow to the diesel generator cooling water heat exchangers during flow balance over the last few years
- OWA 05-02; Minor leakage into 1A cold leg accumulator (CLA) results in slow dilution of the CLA. Operations required to feed and bleed in order to maintain boron concentration
- OWA 05-03; Reactor Operators required to perform NC (reactor coolant system) leakage calculation PT (surveillance test) every 24 hours due to operability questions associated with TS 3.4.15
- OWA 04-11; Ground Water Intrusion causes operator to take preventive measures to avoid equipment damage

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the post-maintenance tests listed below, the inspectors witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s) described in the UFSAR and TS. The tests included the following:

- PT/1/A/4206/002B, Safety Injection (NI) Train B Valve Stroke Timing - Quarterly

(preventive maintenance for 1NI-144B actuator)

- PT/1/A/4403/002A, Nuclear Service Water (RN) Train A Valve Stroke Timing - Quarterly (replacing of pushbutton on main control board panel MC-11)
- PT/1/A/4600/001, Rod Cluster Control Assembly (RCCA) Movement Test (replacement of 5 cards in shutdown bank logic cabinet for banks C, D, and E)
- PT/2/A/4350/004A, Diesel Generator 2A Operability Test; and PT/2/A/4350/002A, 2A Diesel Generator Periodic and Load Sequencer Test (various electrical and mechanical inspections, and maintenance during refueling outage)
- OP/2/A/ 6350/002, Diesel Generator (Run to test synchronization relay replacement)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors evaluated licensee outage activities to verify that the licensee: considered risk in developing outage schedules; adhered to administrative risk reduction methodologies they developed to control plant configuration; adhered to operating license and TS requirements that maintained defense-in-depth; and developed mitigation strategies for losses of the key safety functions identified below:

- Decay heat removal
- Inventory control
- Power availability
- Reactivity control
- Containment

Prior to the outage, the inspectors reviewed the licensee's outage risk control plan to verify that the licensee had performed adequate risk assessments and had implemented appropriate risk management strategies when required by 10 CFR 50.65(a)(4).

The inspectors observed portions of the cool-down process to verify that TS cool-down restrictions were followed. The inspectors observed the items or activities described below, to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan for the key safety functions identified above and applicable TS when taking equipment out of service.

- Clearance Activities
- Reactor Coolant System Instrumentation
- Electrical Power
- Decay Heat Removal
- Spent Fuel Pool Cooling
- Inventory Control

- Reactivity Control
- Containment Closure

The inspectors reviewed the licensee's responses to emergent work and unexpected conditions, to verify that resulting configuration changes were controlled in accordance with the outage risk control plan. The inspectors also observed fuel handling operations to verify that those operations and activities were being performed in accordance with TSs and procedure PT/0/A/4150/037, "Total Core Unloading." Additionally, the inspectors observed refueling activities to verify that the location of the fuel assemblies was tracked, including new fuel, from core offload through core reload. The inspectors also observed the post-fuel load core verification.

Prior to mode changes and on a sampling basis, the inspectors reviewed system lineups and/or control board indications to verify that TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. Also, the inspectors periodically reviewed RCS boundary leakage data, and observed the setting of containment integrity to verify that the RCS and containment boundaries were in place and had integrity when necessary. The inspectors reviewed the licensee's commitments to Generic Letter (GL) 88-17 and verified that the procedural commitments were still in place. The inspectors observed control room activities during mid-loop on March 31, 2005, and April 1, 2005, to verify that GL 88-17 commitments were implemented.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing</u>

a. <u>Inspection Scope</u>

For the surveillance tests identified below, the inspectors witnessed testing and/or reviewed the test data, to verify that the SSCs involved in these tests satisfied the requirements described in the TSs, the UFSAR, and applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions.

- PT/2/A/4350/036B, Diesel Generator 2B 24-Hour Run
- *PT/1/A/4403/001A, 1A RN Pump Test
- *PT/1/A/4403/002D, RN Train A Valve Stroke Timing Test, Quarterly
- *PT/1/A/4252/001A, 1A Auxiliary Feedwater (CA) Pump Performance Test
- PT/2/A/4350/002B, Diesel Generator 2B Operability Test
- TS Surveillance Requirements 3.4.13.1 and 3.4.15 for Unit 1 (U1) and Unit 2 (U2) - [Reference procedures are documented in the Attachment of this report.]
- PT/0/A/4250/001, Main Steam (SM) Safety Valve Setpoint Testing (2SV 8-12)
- **PT/2/A/4255/003C, SM Valve Timing Test at Full Temperature and Pressure
- PT/0/A/4200/032, Periodic Inspection of Ice Condenser Lower Inlet Doors
- *PT/2/A/4204/005A, ND Train A Valve Stroke Timing - Shutdown (valve 2ND-19A)

- *PT/2/A/4204/005B, ND Train B Valve Stroke Timing - Shutdown (valve 2ND-4B)
- PT/1/B/4350/002B, Diesel Generator 1B Operability Test

*(This procedure included inservice testing requirements.)
**(This procedure included testing of a large containment isolation valve.)

The inspectors reviewed PIP M-04-5592, Reactor coolant system leak detection airborne radioactivity monitoring to verify that the licensee identified and implemented appropriate corrective actions. This PIP was associated with TS 3.4.15, Reactor Coolant System Leak Detection. Specifically, the inspectors reviewed surveillance requirements (SR) 3.4.15.2, channel operational test (every 92 days), and SR 3.4.15.4, channel calibration (every 18 months) to determine whether the licensee was adequately accomplishing the surveillance requirement.

b. Findings

(1) Failure to Have Adequate Surveillance Procedures for RCS Leakage Detection Instrumentation

Introduction: The inspectors identified a Green NCV of TS 5.4.1.a for failure to establish and maintain adequate RCS Leakage Detection Instrumentation surveillance procedures for surveillance requirement (SR) 3.4.15.2, channel operational test of containment atmosphere radioactivity monitor; SR 3.4.15.3, channel calibration of containment floor and equipment sump (F&ES) level monitoring system; SR 3.4.15.4, channel calibration of containment atmosphere radioactivity monitor; and SR 3.4.15.5, channel calibration of containment ventilation condensate drain tank (VCDT) level monitor.

Description: On February 28, 2005, the inspectors questioned whether the surveillance procedures used to satisfy SR 3.4.15.2 and SR 3.4.15.4 met the surveillance requirements in relation to the alarm function. 10 CFR 50.36, “Technical Specifications”, states in part that surveillance requirements are to assure that the necessary quality of systems or components are maintained. Technical Specification Section 1.1, Definitions, indicated that channel calibration and channel operational tests set and test the alarms using simulated or actual signals. The licensee’s investigation found that procedures used to satisfy the TS surveillance requirements did not verify that the Operator Aid Computer alarms, credited for all TS 3.4.15, RCS Leakage Detection Instrumentation, were received or actuated when the applicable alarm set points were reached. There were no other procedures that tested these alarms. The licensee initiated PIP M-05-813 and took immediate corrective actions to enter SR 3.0.3, and perform the revised tests to verify the alarm functions worked. All alarm indications were verified to work.

Analysis: The finding is greater than minor because the surveillance procedures had not provided assurance that the necessary quality of systems or components were maintained. Consequently, the reliability and availability of the TS 3.4.15 RCS leakage detection instrumentation in the Barrier Integrity cornerstone were affected. The finding was of very low safety significance (Green) because excessive leakage had not existed based on reactor coolant inventory water balances and that the alarm indication functioned properly when tested. This issue contained elements of problem identification and resolution, in that the licensee's operability determination failed to adequately evaluate whether surveillance requirements had been met and actions to determine the "time to alarm" given current RCS activity levels were not prompt.

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, including Surveillance Procedures. Surveillance Requirement 3.4.15.2 requires the performance of a channel operational test of the required containment atmosphere radioactivity monitor every 92 days. Surveillance Requirements 3.4.15.3, 3.4.15.4, and 3.4.15.5 require the performance of a channel calibration ever 18 months of the required containment F&ES level monitoring system, atmosphere radioactivity monitor, and VCDT level monitor, respectively. Technical Specification Section 1.1, Definitions, indicates that channel calibration and channel operational tests set and test the alarms using simulated or actual signals. This TS is based on Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, position C.5 which indicates that the leakage detectors should be set to detect a leakage rate, or its equivalent, of 1 gallon per minute within 1 hour. Contrary to the above, the licensee failed to have an adequate surveillance procedure for SRs 3.4.15.2, 3.4.15.3, 3.4.15.4, and 3.4.15.5, in that the alarm function credited to alarm in the control room for all required RCS leakage detection systems was not tested and the particulate and gaseous monitor alarm set points were not set at a value equivalent to RCS leakage of 1 gallon per minute in 1 hour for a realistic current activity level. Because this issue was of very low safety significance and was placed in the corrective action program under PIP M-05-0813, this violation is being treated as a NCV in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000369,370/2005002-03, Failure to Have Adequate Surveillance Procedures for RCS Leakage Detection Instrumentation.

(2) MSIV Fails to Close During Surveillance Testing

Introduction: An URI was identified regarding the failure of MSIV 2SM-1 to stroke close during a valve stroke timing surveillance test. This was the fourth degraded MSIV in the past 12 months. Since the last full stroke in October 2003, two subsequent Operability Assessments regarding this valve deemed the valve fully operable. The issue will remain unresolved pending completion of the licensee's root cause investigation.

Description: On March 2, 2005, while performing PT/1/A/4225/003, "SM Valve Timing at Full Temperature and Pressure", during shutdown for Unit 2 EOC16 refueling outage, valve 2SM-1, the "D" loop main steam isolation valve, failed to stroke close. The valve was closed without air assistance following a period of time to allow the valve to cooldown. The inspectors noted during disassembly of the valve, that the atypical use of a hydraulic jack was necessary to separate the stem from the bonnet assembly. Similar to this case was the disassembly of MSIV 1SM-1 in October 2004, when increased friction in the stuffing box area required the use of a hydraulic jack to separate the bonnet from the stem. Main steam isolation valve 2SM-1 did not have any of the MSIV modifications (i.e., anti-vibration kit, additional throat bushing and carbon bushing clearances, and dimensional mapping) at the time of failure, as in the case with MSIVs 1SM-1 and 1SM-7. Markings were observed on the stem for 2SM-1 following its removal. Metallurgical results of samples taken from the stem were consistent with wear debris from the stem. Additionally, the sample flakes analyzed appear to have been sheared. There were also deposits found in the main poppet which were analyzed. These results indicated that metal-oxide corrosion products were present, as well as magnetite and metallic slivers. The licensee is conducting a root cause investigation to determine the failure mechanisms for MSIV 2SM-1.

There were two previous Operability Assessments which evaluated the operability and capability of 2SM-1. One documented in PIP M-04-2109, addressed the operability of 2SM-1 as it related to the failure of MSIV 1SM-7 (thermal expansion, stem guiding friction) in April 2004. The other assessment was documented in PIP M-04-5133 and addressed the transportability of the MSIV 1SM-1 failure (excessive guide body clearance, guide rib wear, misalignment), to the Unit 2 valves. In both cases, 2SM-1 was deemed fully operable and capable of performing its safety function.

Analysis: The significance of this issue is undetermined pending completion of the licensee's root cause investigation.

Enforcement: Pending the licensee's root cause determination, disposition this issue will be tracked via URI 05000370/2005002-04, MSIV Fails to Close During Surveillance Testing.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification listed below, to verify that the modification did not affect the safety functions of important safety systems, and to verify that the modification satisfied the requirements of 10 CFR 50, Appendix B, Criterion III, Design Control.

- MD200238, Remove regulator for pressurized relief tank (PRT) sample vessel 2NCME6240 (Unit 2)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on February 23, 2005, to verify licensee self-assessment of classification, notification, and protective action recommendation development in accordance with 10 CFR 50, Appendix E.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Controls To Radiologically Significant Areas

a. Inspection Scope

Access Controls

Licensee program activities for monitoring workers and controlling access to radiologically significant areas and tasks were inspected. The inspectors evaluated procedural guidance; directly observed implementation of administrative and established physical controls; assessed worker exposures to radiation and radioactive material; and appraised radiation worker and technician knowledge of, and proficiency in, implementing radiation protection program activities.

During the inspection, radiological controls for selected operations and maintenance activities were observed and discussed. The inspectors reviewed on-going tasks including Independent Spent Fuel Storage Installation (ISFSI) cask loading and movement to onsite storage facility, valve maintenance, and ‘at power’ containment surveillance activities. In addition, licensee controls for selected tasks conducted during the previous Unit 1 EOC16 refueling outage were assessed. The evaluations included, as applicable, Radiation Work Permit (RWP) details; use and placement of dosimetry and air sampling equipment; electronic dosimeter (ED) set-points; and monitoring and assessment of worker dose from direct radiation and airborne radioactivity source terms. Effectiveness of established controls were assessed against area radiation and contamination survey results, and occupational doses received. Physical and administrative controls and their implementation for extra-high radiation area (EHRA) locations and for storage of highly activated material within the spent fuel pool (SFP) areas were evaluated through discussions with licensee representatives, direct field observations, and record reviews.

Occupational worker adherence to selected RWPs and Health Physics Technician (HPT) proficiency in providing job coverage were evaluated through direct observations of staff performance during job coverage and routine surveillance activities, review of selected exposure records and investigations, and interviews with licensee staff. Radiological postings and physical controls for access to designated high radiation (HRA) and EHRA locations within Auxiliary Building and SFP areas were evaluated during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for selected Auxiliary Building and SFP locations. Occupational exposures associated with direct radiation, potential radioactive material intakes, and from discrete radioactive particle (DRP) or dispersed skin contamination events for calendar year (CY) 2004 were reviewed and discussed.

Radiation protection program activities were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; UFSAR details in Section 12, Radiation Protection; TS Sections 5.4, Procedures, and 5.7, High Radiation Area; and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in the Attachment to this report.

Problem Identification and Resolution

Licensee Corrective Action Program (CAP) documents associated with access controls to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NSD 208, Problem Investigative Process, Revision (Rev.) 27. Licensee audits, self-assessments and PIPs related to access controls that were reviewed and evaluated in detail during inspection of this program area are identified in the Attachment to this report.

b. Findings

Introduction: A Green NCV of TS 5.4.1 (a) was identified by the inspectors for failure to follow procedural guidance for ISFSI area quarterly radiation surveys conducted to assure compliance with 10 CFR Parts 20 and 72 requirements.

Description. From a review of recent ISFSI radiation dose rate surveillance records, the inspectors noted that for the quarterly surveillances conducted on August 24, 2004, gamma radiation measurements were made using a RO-20 portable ion chamber instrument. Radiation Protection Management Procedure (RPMP) 7-8, "Maintaining Radiation Control Zones Associated with ISFSI," and Health Physics Procedure (HP)/0/B/1003/063, "Routine Surveillance," Enclosure 5.17, Attachment 1, Table 1, define occupancy factor assumptions and establish trending point total dose rate limits for the ISFSI storage pad facility to evaluate compliance with 10 CFR Parts 20 and 72 requirements. The limits range from 0.05 millirem per hour (mrem/hr) at the site's north unrestricted area fence to 2.0 mrem/hr at the ISFSI north protected area fence. Licensee procedure HP/0/B/1003/0063 recommends use of Micro Rem equipment for conducting the quarterly surveillances. The inspectors noted that the RO-20 instrument's minimum sensitivity to gamma radiation, approximately 0.1 mrem/hr, exceeded the dose rate limit (0.05 mrem/hr) at Trend Point Nos. 32 and 33. Thus, actual dose rates measurements between 0.05 and approximately 0.1 mrem/hr would not have been properly evaluated for the subject trend point data locations. Further, the dose rate values documented (<0.1 mrem/hr) for the subject trending points did not allow verification that the established procedural limits used to demonstrate compliance with 10 CFR Parts 20 and 72 requirements were met.

Analysis: The inspectors determined that the failure to use proper instrumentation to monitor ISFSI perimeter dose rates was a performance deficiency in that the licensee is expected to conduct accurate surveys of ISFSI operations to demonstrate compliance with 10 CFR Parts 20 and 72 requirements. This finding was greater than minor because it is associated with the Program and Process attributes of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to protect personnel from unnecessary or unintended exposure to radiation. The finding was evaluated using the Occupational Radiation SDP and was determined to be of very low safety significance (Green) because the established dose rate limits are based on conservative occupancy factors and no unexpected elevated dose rates and resultant compliance issues were identified from reviews of appropriate dose rate surveys conducted prior and subsequent to August 24, 2004. This finding has a cross cutting aspect involving human performance. Specifically the technician performing ISFSI area radiation surveys selected an instrument that did not have the required range of detection and was not in accordance with procedural requirements.

Enforcement: TS 5.4.1(a) requires written procedures to be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. RG 1.33 Appendix A, Section 7.e specifies procedures for radiation surveys. Licensee Shared Health Physics Procedure (SH)0/B/2000/0004, "Taking, Counting, and Recording Surveys," requires radiation surveys to be conducted with portable radiation instrumentation which cover the range of radiation levels expected.

Contrary to TS 5.4.1(a), the licensee failed to follow approved procedures, in that, on August 24, 2004, quarterly ISFSI dose rate surveys were conducted using portable radiation monitoring instrumentation, a RO-20 ion chamber survey meter, which did not cover the lower range of radiation levels expected, less than 0.05 mrem/hr, for two boundary trending points. Further, the dose rate values documented (<0.1 mrem/hr) for the subject trending point locations did not allow verification that procedural dose rate limits used to demonstrate compliance with 10 CFR Parts 20 and 72 requirements were met. Because the failure to follow procedures for the conduct of ISFSI radiation surveys has been determined to be of very low safety significance and has been entered into the licensee's CAP (PIP 05-00361), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000369,370/2005002-05, Failure to Follow Procedural Guidance for Conducting ISFSI Radiation Surveys.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Radiation Monitoring Instrumentation and Post-Accident Sampling Systems

During tours of the auxiliary building, Radiologically Controlled Area (RCA) exit points, and administrative building areas, the inspectors observed installed radiation detection equipment including Area Radiation Monitor (ARM), Personnel Contamination Monitor (PCM), Portal Monitor (PM), and Whole Body Counter (WBC) equipment. During the tours, the adequacy of the equipment's physical location and material condition were evaluated.

From review of selected records and discussions with the system engineer, the inspectors evaluated completion and adequacy of equipment calibrations, and assessed system operability and reliability. The two most recent calibration records for ARM EMF-43A, Control Room Air; ARM 1EMF-9, U1 Reactor Building Incore Room; and ARM 2EMF-51A, U2 Containment High Range ARM were evaluated against required calibration frequencies and technical requirements. In addition, the inspectors reviewed and discussed procedural guidance used to meet post-accident dose-equivalent iodine monitoring requirements against commitments specified in license amendments detailing changes to Post Accident Sampling System (PASS) capabilities.

During equipment walk-downs, the inspectors observed functional checks of various fixed and portable radiation monitoring/detection instruments. The observations included source checks of PCM, PM, and WBC equipment. The inspectors reviewed calibration records and discussed the functional testing and testing intervals for selected PCM and PM equipment located at the RCA and protected area exits. PCM equipment detection capabilities were demonstrated using a low-level mixed radionuclide source that was passed through the equipment. The operability and analysis capabilities of the WBC equipment were evaluated. WBC equipment operations and training of staff were reviewed and discussed with responsible personnel.

For selected portable survey instrumentation used in field tasks, the inspectors observed HPT selection of survey instruments, completion of required performance and/or functional checks, and use of instruments during selected task coverage. Availability of portable instruments for licensee use was evaluated through observation of instruments staged for issue and discussion with licensee personnel. For frisker and portable survey instruments in the field, the inspectors noted calibration sticker data. Calibration data for five portable instruments staged or recently used for coverage of HRA/EHRA field tasks were reviewed.

Operability and reliability of selected radiation detection instruments were reviewed against 10 CFR 20; NUREG-0737, "Clarification of TMI Action Plan Requirements;" License Amendment Number (No.) 9 to NPF and No. 180 to NPF 17; TS Sections 3 and 5.4; Selected Licensee Commitments (SLC) Manual Section 16.7; UFSAR Chapter 12; and applicable licensee procedures. Documents reviewed during the inspection are listed in the Attachment to this report.

<u>Self-Contained Breathing Apparatus (SCBA) and Protective Equipment</u>

Selected SCBA units staged for emergency use in the Control Room and other locations were inspected for material condition and adequate air pressure. The inspectors also reviewed the previous five years of maintenance records for components of three SCBA units. In addition, certification records associated with supplied-air quality were reviewed and discussed.

Qualifications for staff responsible for testing and repairing SCBA equipment were evaluated through review of training records. Two Control Room operators were interviewed to determine their knowledge of available SCBA equipment locations, including corrective lens inserts if needed, and their training on bottle change-out during periods of extended SCBA use. Respirator qualification records were reviewed for several licensed Operations and Maintenance Department personnel designated as emergency responders.

Licensee activities associated with maintenance and use of respiratory protection equipment were reviewed against 10 CFR 20; Regulatory Guide (RG) 8.15, "Acceptable Programs for Respiratory Protection;" American National Standards Institute (ANSI)-

Z88.2-1992, “American National Standard for Respiratory Protection;” and applicable licensee procedures. Documents reviewed during the inspection are listed in the Attachment to this report.

Problem Identification and Resolution

Four PIPs and one corporate audit associated with instrumentation and protective equipment were reviewed and assessed. Inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NSD 208, PIP, Rev. 27. Documents reviewed are listed in the Attachment of this report.

b. Findings

Introduction: A self-revealing, Green NCV was identified for the use of supplied-air respiratory protective equipment without adequate breathing air (VB) system capacity as required by 10 CFR 20.1703(e).

Description: During the licensee’s 1EOC16 refueling outage, supplied-air ‘Delta’ suits were used as respiratory protective equipment on a large scale for the first time. The Delta suits were supplied with filtered air from the plant’s VB system which consisted of two independent air compressors each rated at a maximum capacity of 140 standard cubic feet per minute (scfm). Each Delta suit required a nominal airflow of 36 scfm to perform it’s function.

On March 25, 2005, a VB system Lo Pressure alarm was received in the Control Room. Per licensee procedure, the HP staff was contacted and the six workers currently inside ‘Delta’ supplied-air suits were taken off the VB system and exited from their work areas. Shortly thereafter, the VB system pressure returned to normal. At the time of the annunciator alarm, the VB system ‘A’ train was inoperable and the demand for supplied air (216 scfm for the six Delta suits in use) exceeded the 140 scfm capacity of the in-service ‘B’ train VB compressor. This resulted in decreasing system capacity and subsequent annunciation of the control room alarm setpoint.

Capacity limitations of the VB system were not taken into account before use of the new Delta suits. Previously, the licensee had used traditional bubble hoods as their supplied-air respirator. However, these bubble hoods had a lower demand for air than the new Delta suits. Licensee personnel improperly assumed that the VB system would supply adequate air without completion of an engineering evaluation to establish limits for the new equipment use.

Analysis: The inspectors determined that the licensee’s failure to provide adequate breathing air capacity to supplied-air respirators is a performance deficiency because the licensee is expected to follow the requirements of 10 CFR 20.1703(e) and the incident was reasonably within the licensee’s ability to foresee and correct. This finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of Plant Equipment and Instrumentation and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Using supplied-air respirators without establishing equipment limitations could result in low air flow negatively impacting use of the equipment to protect workers for exposure to radioactive materials. The finding was evaluated using the Occupational Radiation Safety SDP. This issue was not related to ALARA planning, nor did it involve an overexposure or substantial potential thereof, and the ability to assess dose was not compromised. For these reasons, the SDP evaluation concluded that the issue is of very low safety significance (Green). Also personnel using the Delta suits at the time of the incident were not in danger of completely losing air since the VB system low air pressure annunciator alerted operators before loss of air flow became critical. In addition, the Delta suit features a “self-escape” capability that could be utilized if air completely ran out.

Enforcement: 10 CFR 20.1703(e) states, in part, for use of respiratory protective equipment, licensees shall consider limitations appropriate to the type and mode of use, and shall use equipment in such a way as not to interfere with the proper operation of the respirator. Contrary to the above, on March 25, 2004, the licensee failed to consider limitations of the VB system and the number of Delta suit supplied-air respirators in use, which could have resulted in improper operation of the respiratory protective equipment. Because the failure to comply with 10 CFR 20.1703(e) is of very low safety significance and has been entered into the licensee’s corrective action program (PIP 04-01757), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000369,370/2005002-06, Failure to Provide Adequate Breathing Air Capacity for Supplied-Air Respiratory Equipment.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Monitoring and Radwaste Equipment

During inspector walk-downs, accessible sections of the U1 and U2 liquid radioactive waste (radwaste) system including Waste Monitor Tank (WMT) A and WMT B, system piping and valves, and Waste Liquid (1-EMF-49) and Containment Ventilation Unit Condensate (1-EMF-44) monitors were assessed for material condition and conformance with current system design diagrams. Inspected components of the gaseous effluent process and release system included Waste Gas Decay Tanks, U1/U2 unit vent air particulate/noble gas/iodine monitor (1/2-EMF-35,36,37) system, condensate air ejector monitor (1-EMF-33), Waste Gas Discharge monitor (1-EMF-50), U1/U2 containment air particulate monitors (1/2-EMF-38), and associated effluent sample lines. The inspectors interviewed chemistry supervision regarding radwaste equipment configuration, effluent monitor operation, and system modifications.

The operability, availability, and reliability of selected effluent process sampling and detection equipment used for routine and accident monitoring activities were reviewed and evaluated. The inspectors reviewed results of calibrations and/or performance surveillances for selected process monitors, flowmeters, and air filtration systems. For effluent monitors 1/2-EMF-33, 1/2-EMF-35 and 1-EMF-50, the inspectors reviewed technical bases for Offsite Dose Calculation Manual (ODCM) and/or design-related changes. For select control room effluent monitor read-outs, the inspectors independently verified established alarm set-points against procedural/release requirements. The two most recent surveillances on the Auxiliary Building Ventilation (VA) High Efficiency Particulate Air (HEPA)/charcoal air treatment system also were reviewed. The inspectors evaluated out-of-service (OOS) effluent monitors and compensatory action data for the period January 2003, through December 2004. In addition, unit vent isokinetic design documents and unit vent versus sample line flow rates (velocities) were reviewed and discussed with chemistry and engineering staff.

Installed configuration, material condition, operability, and reliability of selected effluent sampling and monitoring equipment were reviewed against details documented in the following: 10 CFR 20; RG 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials In Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants," June 1974; ANSI-N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities;" TS Section 5; the ODCM, Rev. 45; Selected Licensee Commitment (SLC) Manual, Section 16.11, Rev. 23; and UFSAR, Chapter 11. Procedures and records reviewed during the inspection are listed in the Attachment to this report.

Effluent Release Processing and Quality Control (QC) Activities

The inspectors directly observed the weekly collection of airborne effluent samples from the U2 vent continuous iodine sampler, the U1 vent particulate sampler (1-EMF-35), and U2 vent gas and tritium grab samples. Selected activities associated with a liquid waste tank 'A' release were observed and radwaste operators were interviewed regarding valve lineups and effluent monitor operation. Chemistry technician proficiency in collecting, processing, and counting the samples, as well as preparing the applicable release permits was evaluated.

QC activities regarding gamma spectroscopy and beta-emitter detection were discussed with count room technicians and Chemistry supervision. The inspectors reviewed the daily QC check data of January 25, 2004, for the four High Purity Germanium (HPGe) detectors and the two liquid scintillation detectors. For HPGe detector No. 4, the inspectors reviewed CY 2003 through 2004 daily QC quarterly summaries. The inspectors also reviewed the most recent calibration records for HPGe detectors Nos. 2 and 4, and liquid scintillation counter No. 1. In addition, results of the radiochemistry cross-check program were reviewed from the second quarter of CY 2003 through the third quarter of CY 2004.

Four procedures for effluent sampling, processing, and release were evaluated for consistency with licensee actions. Four liquid and three gaseous release permits were reviewed against ODCM specifications for pre-release sampling and effluent monitor setpoints. The ODCM was reviewed and discussed with responsible licensee representatives to identify and evaluate any changes made since January 1, 2002. The inspectors also reviewed CY2002 and CY2003 annual effluent reports for effluent release data trends and anomalous releases.

Observed task evolutions, count room activities, and offsite dose results were evaluated against details and guidance documented in the following: 10 CFR 20 and Appendix I to 10 CFR 50; ODCM; RG 1.21; RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50 Appendix I," October 1977; and TS Section 5. Procedures and records reviewed during the inspection are listed in the Attachment to this report.

Problem Identification and Resolution

Five licensee PIPs and three audits associated with effluent release activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve selected issues in accordance with NSD 208, Revision 27. Reviewed documents are listed in the Attachment to this report.

b. Findings

Introduction: An URI was identified regarding the adequacy of the Unit Vent Monitors (1/2-EMF-35) to collect representative particulate samples. This is an URI pending determination of whether observed long-term changes in unit vent operating conditions, which differ from both UFSAR and the U1 or U2 Plant Vent Effluent Monitor (1/2-EMF-35) sampling system design specifications, were evaluated previously with respect to maintaining isokinetic sampling and if observed operations of the unit vent and 1/2-EMF-35 sampling system support representative effluent measurement particulate sampling.

Description: From discussions with the system engineer and review of the manufacturer design drawing MCM 1346.05-0075.001, the inspectors determined the nozzles (unit vent sample probes) were designed to ensure isokinetic (representative) sampling when the unit vent effluent velocities were 3100 and 2200 feet per minute (fpm) for U1 and U2, respectively. These values correspond with the maximum effluent velocities specified in Table 11-25 of the UFSAR for each vent. The inspectors further noted that velocity tolerances to maintain isokinetic sampling conditions were not specified within the subject design document nor available from the manufacturer.

From review of current operations, the inspectors noted that the unit vent effluent velocities did not correspond with those referenced in the UFSAR or applicable design document for the 1/2-EMF-35 equipment. Specifically, the U1 and U2 vent volume flow rates on January 26, 2005, were observed to be approximately 98,000 cubic feet per minute (cfm), corresponding to vent effluent velocities of approximately 2,550 fpm. Discussions with licensee personnel indicated that the observed volume flow rate was typical of operations for a number of years, following a reduction in operational vent velocities related to safety concerns associated with negative pressure gradients produced in the auxiliary building. In addition, routine decreases in the vent volume flow rates are know to occur based on certain ventilation systems being out of service for maintenance activities. Review of the U1 unit vent stack flow data from December 27, 2004, through January 26, 2005, and discussions with the system engineer, indicated that removal of the Fuel Pool Ventilation (VF) system input can reduce the unit vent flow volume flow rate to approximately 70,000 cfm (1,820 fpm) for periods of a day or longer and removal of the VA system input can reduce the unit vent volume flow rate to 40,000 cfm (1,040 fpm). From discussions with licensee personnel and review of applicable records, the inspectors were unable to determine if the identified changes in the unit vent effluent velocities were evaluated for their potential influence on maintaining representative sampling conditions.

The inspectors also observed that the current sample volume flow rate for the unit vent effluent monitors was 4.7 scfm (1/2-EMF-35 at 4 scfm plus parallel sampler at 0.7 scfm), rather than the 5 scfm specified in the manufacturer's design drawing. A review of applicable PIP documents indicated that an evaluation had been performed in 1997 with respect to decreasing the 1/2-EMF-35 flow rate from 5 to 4 scfm, with concurrence from the manufacturer. During discussions with licensee personnel, they indicated that consideration was being given to further decreasing the sampler volume flow rate but had not been evaluated to ensure representative sampling of airborne particulates.

Analysis: The inspectors noted that a failure to evaluate the effect of significantly varying the unit vent effluent velocity or 1/2-EMF-35 sampler velocities from the isokinetic probe design specifications is a performance deficiency because the licensee is expected to establish, implement and maintain the quality assurance requirements for effluent measurements. The failure to complete the required evaluation is more than minor because it is associated with the program and process attribute of the Public Radiation Safety Cornerstone and it affected the cornerstone objective to assure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The failure to maintain isokinetic sampling conditions and conduct appropriate evaluations to assure representative sample collection from the U1/U2 plant ventilation effluent streams could result in inaccurate measurement and reporting of airborne particulate radionuclides in samples and resultant dose estimates.

Enforcement: TS 5.4.1(c) requires written procedures for quality assurance for effluent and environmental monitoring and all programs specified in TS 5.5, which includes the Radioactive Effluent Controls Program as specified in TS 5.5.5. Selected Licensee Commitment (SLC) Manual Table 16.11.7-1 specifies instrumentation used for radioactive effluent monitoring. UFSAR section 11.4.2 specifies that operating limits for continuous monitoring are based on RG 1.21. Section C.6 of RG 1.21 details that the guides for representative sampling from ducts and stacks contained in ANSI N13.1-1969, are generally acceptable and provide adequate bases for the design and conduct of monitoring programs for airborne effluents including velocity criteria for isokinetic sampling. UFSAR Section 11.4.2.2.3, Unit Vent Monitor, specifies that an isokinetic sample probe is located within the unit vent. As specified in drawing MCM1346.05-0075.001, the U1 and U2 vent sample probes supplying 1/2-EMF-35 are designed to provide isokinetic sampling when the unit vent effluent velocities are 3,100 fpm and 2,200 fpm for U1 and U2 respectively, with no error bars specified.

At the time of the onsite inspection the licensee was unable to determine if the observed changes in unit vent volume flow rate had been evaluated properly to assure isokinetic conditions. In addition, the licensee was working with the vendor's staff to determine if the observed differences between the effluent monitor design document specifications and the current plant vent and 1/2-EMF-35 operations maintained isokinetic sampling conditions. Pending NRC review of the licensee's determination whether the observed changes had been previously evaluated, and the vendor's review of the 1/2-EMF-35 system design operating specifications and associated operational tolerances, this issue will be identified as URI 05000369,370/2005002-07, **Review Licensee Assessments and Vendor Evaluations for Observed U1/U2 Unit Vent Effluent Velocity Changes to Assure Representative Effluent Sampling**.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

REMP Implementation

During the accompaniment of environmental technicians, the inspectors observed collection of environmental samples and surveillance of sampling instruments. The inspectors noted the material condition and flowmeter operability of airborne particulate filter and iodine cartridge sample stations at monitoring location Nos. 120, 121, 125, 133, 134C, 139, 141 (control), 192, and 195. Environmental thermoluminescent dosimeter (TLD) Nos. 145, 182SI, 191SI, and 187SI were checked for material condition and appropriate identification. Collection of milk samples was observed at sample locations 139 and 141 (control). The inspectors determined the current location of selected air samplers, TLDs, and dairy farms using NRC global positioning system instrumentation. Land use census results and sample collection/processing activities were discussed with environmental technicians.

The inspectors reviewed the two most recent calibration records for 14 environmental air samplers and two water samplers. The inspectors also reviewed the 2003 Radiological Environmental Operating Report, results of the 2003 and 2004 interlaboratory cross-check program, and three procedures for environmental sample collection, processing, and instrument calibration. Selected environmental measurements were reviewed for consistency with licensee effluent data, evaluated for radionuclide concentration trends, and compared with detection level sensitivity requirements.

Procedural guidance, program implementation, and environmental monitoring results were reviewed against: 10 CFR 20; Appendix I to 10 CFR 50; SLC 16; TS 5.0; ODCM; RG 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment; and the Branch Technical Position, An Acceptable Radiological Environmental Monitoring Program - 1979. Documents reviewed are listed in the Attachment to this report.

Meteorological Monitoring Program

During a weekly surveillance of the meteorological tower, the inspectors observed the physical condition of the tower and discussed equipment operability and maintenance history with a tower technician. The inspectors compared locally generated meteorological data with information available to control room operators. For the primary meteorological measurements of wind speed, wind direction, and temperature, the inspectors reviewed calibration records for applicable tower instrumentation and evaluated measurement data recovery for CY 2004.

Licensee procedures and activities related to meteorological monitoring were evaluated against: ODCM; UFSAR Section 2.3; ANSI/ANS-2.5-1984, Standard for Determining Meteorological Information at Nuclear Power Sites; and Safety Guide 23, Onsite Meteorological Programs. Documents reviewed are listed in the Attachment to this report.

Unrestricted Release of Materials from the Radiologically Controlled Area (RCA)

The inspectors observed surveys of material and personnel being released from the RCA using Small Article Monitor (SAM), PCM, and PM instruments. The inspectors also observed source checks of these instruments and discussed equipment sensitivity and release program guidance with licensee staff scientists.

To evaluate the appropriateness and accuracy of release survey instrumentation, radionuclides identified within recent waste stream analyses were compared against the radionuclides used in current calibration sources and performance check sources. The inspectors also reviewed the two most recent calibration records for selected SAM instruments.

Licensee programs for monitoring materials and personnel released from the RCA were evaluated against: 10 CFR 20; UFSAR Section12; and IE Circular 81-07, Control of Radioactively Contaminated Material, 5/14/81. Documents reviewed are listed in the Attachment to this report.

Problem Identification and Resolution

The inspectors reviewed six PIPs involving environmental monitoring, meteorological monitoring, and release of radioactive materials. The inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NSD 208, Revision 27. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified. However, the following minor violations are being documented as items of NRC interest as required by Inspection Manual Chapter 0612.

A minor violation of 10 CFR 20.1802 was identified for failure to maintain control over licensed material. Shortly after the McGuire Nuclear Station (MNS) 2EOC15 refueling outage ended, the licensee was informed that small quantities of licensed radioactive material were discovered on workers clothing during in-processing worker whole body counts being conducted at both the Turkey Point (October 8, 2003) and VC Summer (October 13, 2003) nuclear plants. The contaminated individuals, most recently, had previously worked for the licensee during the 2EOC15 outage. The approximate levels of clothing contamination ranged from 4 - 11 nanocuries (nCi) of Cobalt (Co)-58. Subsequently, following the 1EOC16 outage, additional instances of small quantities of licensed radioactive material were discovered during incoming whole body counts of personnel at the Robinson (April 13, 2004), VC Summer (April 2, 2004), and Vogtle

(May 3, 2004) nuclear plants. These contaminated individuals also had worked at MNS during the recent 1EOC16 outage. The levels of contamination ranged from 4 - 21 nCi of Co-58. For all cases, the inspectors verified that the quantities of radioactive material were below the detection limits of the licensee's personnel contamination monitoring equipment.

The inspectors reviewed the licensee's program for release of potentially contaminated personnel and equipment, and identified no concerns associated with the technical bases for the monitoring equipment used nor found any performance deficiencies regarding program implementation. These issues have been entered into the licensee's CAP as PIPs M-03-05010, M-03-05078, M-04-02255 and M-04-02535.

4. OTHER ACTIVITIES

40A1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data to verify the accuracy of reported performance indicator (PI) data for the periods listed below. To verify the accuracy of the reported PI elements, the reviewed data were assessed against PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2.

Occupational Radiation Safety Cornerstone

For the period January 2004 through December 2004, the inspectors assessed CAP documents associated with the Occupational Exposure Control Effectiveness PI to determine whether HRA, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred. The inspectors evaluated data reported to the NRC, and subsequently sampled and assessed applicable CAP documents and selected Health Physics Program records. The reviewed records included selected personnel contamination event data, internal dose assessments, and Unusual Dosimetry Occurrence (UDO) records. Reviewed documents relative to this PI are listed in Sections 2OS1, 2OS3, and 4OA5.2 of the Attachment to this report.

Public Radiation Safety Cornerstone

The inspectors reviewed the Radiological Effluent Technical Specifications (RETS)/ ODCM Effluent Occurrence PI results for the period January 2003 through December 2004. The inspectors reviewed selected OOS effluent radiation monitor and compensatory sampling data, abnormal release results as reported in the 2003 Annual Effluent Report, selected PIP documents related to RETS/ODCM issues. In addition, the inspectors reviewed cumulative and projected doses to the public for the period January 2003 through December 31, 2004. Documents reviewed are listed in sections 2PS1 and 2PS3 of the Attachment to this report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (PI&R)

.1 Daily Screening of Items Entered Into the Corrective Action Program

As required by Inspection Procedure 71152, "Identification and Resolution of Problems", and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing hard copies of condition reports, attending daily screening meetings, and accessing the licensee's computerized database.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected PIP M-05-0871 for detailed review. This PIP was associated with discovering that the actuator switch mechanism for 2NI-76A, 2C Cold Leg Accumulator discharge valve, which was originally thought to be a Pre-78 design, was actually a Post-78 design and not exempt from a previous Part 21 notice. The inspectors reviewed this report to verify that the licensee identified the full extent of the issue, performed an appropriate evaluation, and specified and prioritized appropriate corrective actions. The inspectors evaluated the report against the requirements of the licensee's corrective action program as delineated in corporate procedure NSD 208, "Problem Identification Process," and 10 CFR 50, Appendix B .

b. Observations and Findings

From the review of PIP M-05-0871, no findings of significance were identified. However, the inspectors identified that other valves may exist that were previously evaluated and assumed to be exempt from the Part 21 notice relating to the Post-78 switch mechanisms. The licensee has discovered cases in which the records show a particular valve as a Pre-78 design (exempt from the Part 21 Notice); however, through field inspection it is determined to be a Post-78 design and susceptible to the non-conforming condition. The licensee has initiated PIP M-05-1082 to evaluate and analyze valves that were previously assumed to be Pre-78 design based on records, and perform an extent of condition as if they were Post-78 design.

.3 Summary of PI&R Cross-Cutting Findings Documented Elsewhere

The RCS leakage detection instrumentation issues (Section 1R15 and 1R22b.(1)) contained elements of PI&R. During review of the operability evaluation for PIP-04-5592, the inspectors identified that the licensee's investigation failed to consider all of the information in the UFSAR for the RCS leakage detection gaseous and particulate radiation monitors. In addition, the licensee had failed to adequately evaluate whether surveillance requirements had been met. The licensee's corrective action due date for determination of how long it would take at current RCS activity levels to receive an alarm due to 1 gpm leakage was not consistent with the necessity of making prompt operability determinations.

The Unit 1 MSIV issues (section 4OA5.4) contained elements of PI&R. The licensee had evaluated both 1SM-1 and 1SM-3 prior to their failures and determined that they were operable. In both cases, the failure to take prompt corrective action had resulted in the valves becoming inoperable.

The Unit 2 containment annulus fire protection sprinkler issue (section 1R05) had elements of PI&R. A trend was identified in 2004 where the licensee was failing to identifying fire protection issues. This is another example.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000369/2004-002, Main Steam Isolation Valve Inoperable

a. Inspection Scope

On November 4, 2004, the licensee identified MSIV 1SM-3 was not completing the closing function associated with the pilot poppet. Upon disassembly of the valve, the licensee discovered the MSIV to be improperly reassembled from an October 2002 disassembly. The LER addressed two 10 CFR 50.73 reporting requirements: (1) MSIV 1SM-3 inoperable longer than allowed by TSs (from October 2002 to November 2004); and (2) Concurrent inoperability of MSIV 1SM-1 caused a loss of safety function for the main steam system. The inoperability of MSIV 1SM-3 was caused by improper reassembly due to a deficient corrective maintenance procedure. Additionally, inadequate and untimely corrective actions attributed to the valve degradation. The inspectors reviewed the subject LER and associated PIP M-04-5315, which documented this event in the corrective action program, to verify that the cause of the MSIV 1SM-3 inoperability was identified and proper corrective actions were addressed. The technical aspects related to the inoperability of MSIV 1SM-3 are discussed in Section 4OA5.4 of this report under the closure of URI 05000369/2004008-02, Potential for Multiple MSIV Inoperability.

b. Findings

Introduction: A Severity Level (SL) IV NCV was identified for failure to report a condition prohibited by TSs related to past inoperability for MSIV 1SM-1, as required by 10 CFR 50.73.

Description: On December 29, 2004, the licensee issued LER 05000369/2004-02, Main Steam Isolation Valve Inoperable, regarding the inoperability of MSIV 1SM-3 from October 2002 until November 2004, a period longer than allowed by TS. The LER also cited a loss of safety function for the main steam isolation system due to the simultaneous inoperability of MSIV 1SM-1. Main steam isolation valve 1SM-1 was not considered to have been in a condition prohibited by TSs, and therefore not considered reportable at the time. In the LER, the licensee indicated that upon further evaluation of results of the safety significance, any conclusions different from those stated in the LER would be revised.

The inspectors questioned the licensee on the content of the LER with regards to 1SM-1 and the guidance provided by NUREG-1022, "Event Reporting Guidelines," and 10 CFR 50.73. Per 10 CFR 50.73, the LER must include the cause of each component or system failure if known, and the failure mode, mechanism, and the effect of the failure. The LER only stated that 1SM-1 was inoperable during the time 1SM-3 was inoperable. The licensee stated that the root cause for 1SM-1 was not complete. The root cause report for 1SM-1 was approved on January 18, 2005. Reportability guidance provided by NUREG-1022 states that based on the best information available, determine the most likely time the condition was introduced to the component. The licensee is required to report past TS inoperabilities within 60 days of discovery. From the root cause investigation, it was determined that indications of the valve failure mechanisms were present in April 2004. Scoring marks on the stem were present in April 2004 that were attributed to the main poppet tipping and actuator misalignment. Subsequently, these conditions were also attributed as the primary failure contributors. On October 18, 2004, the next stroke for 1SM-1 following the April 2004 indications, the valve failed to close. The licensee had not reported this condition by the end of the period on March 31, 2005, and the reportability evaluations in PIPs M-04-5043 (1SM-1) and M-04-5315 (1SM-3) had been closed.

Analysis: The significance of this issue was not formally evaluated under the Reactor Oversight Process per the Enforcement Policy. In accordance with the Enforcement Policy, the regulatory significance of the issue is determined by the severity of the associated technical issue. Based on the very low safety significance of the technical issue as described by NCV 05000369/2005002-09 in section 4OA5.4 of this report, this violation is categorized as a SL IV violation under Enforcement Policy, Supplement I.

Enforcement: Technical Specification 3.7.2 states that if one Main Steam Isolation Valve is inoperable in Mode 1, the affected MSIV shall be operable within 8 hours. If this action is not met, the affected unit shall be in Mode 2 within six hours. Reportable events under 10 CFR 50.73 include those events in which the operation or condition is prohibited by Technical Specifications (e.g. inoperable greater than the allowed outage time). Specifically, 10 CFR 50.73(a)(2)(i)(B) states that any operation or condition prohibited by TSs must be reported within 60 days of discovery of the event. Contrary to the above, from March 20, 2005 to present, the licensee failed to report a condition prohibited by TS within the 60 day time frame, in that, during the root cause investigation (completed on January 15, 2005), the licensee discovered that the failure mechanisms for MSIV 1SM-1 were present in April 2004 and the unit had been in Mode 1 until October 2004; valve 1SM-1 was corrected at this time. This failure to report a condition prohibited by TSs as required by 10 CFR 50.73 is being treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000369/2005002-08, Failure to Report a Condition Prohibited by Technical Specifications. This issue is captured in the licensee's corrective program under PIP M-05-2000.

.2 Auxiliary Feedwater Actuation

a. Inspection Scope

The inspectors responded to a safety actuation of the auxiliary feedwater system in Mode 3 on March 2, 2005. The actuation was due to a low suction pressure trip of the last running feedwater pump which caused both motor-driven auxiliary feedwater pumps to start. The inspectors reviewed the event to determine if all required safety equipment had responded as required. The licensee reported this event per the requirements of 10 CFR 50.72.

b. Findings

No findings of significance were identified.

4OA4 Summary of Human Performance Cross-Cutting Findings Documented Elsewhere

The RCS leakage detection instrumentation issues in section 1R15 contained elements of human performance. Although the licensee was aware of NRC violations at four other sites and was aware that their position was contrary to the NRC's, licensee operations and engineering personnel determined the radiation monitors to be operable.

The MSIV issues in section 4OA5.4 contained elements of human performance. Each involved failures to properly evaluate data and deficiencies associated with the valves.

The Unit 2 containment annulus sprinkler issue in section 1R05 contained elements of human performance. The sprinklers were required to be checked every 18 months, yet these checks were ineffective in identifying that the sprinklers were obstructed.

4OA5 Other Activities

.1 Initial Cask Loading and Storage Observation

a. Inspection Scope

The inspectors reviewed the documentation package for Cask NAC-UMS TSC-002 (Document Control No. MCEI 0400-147) created using procedure XSM-006, "Workplace Procedure for Selecting Spent Fuel for Use of NAC-UMS System at McGuire," and Regulatory Guide 3.54, "Spent Fuel Heat Generation," to verify that the selected fuel assemblies and burnable poison inserts met the requirements for insertion in dry cask storage. The inspectors reviewed the Cask loading verification video tape 98629411-01, Cask (NAC) #9 Load Verification 1/15/05, to verify that the alpha-numeric identification numbers stamped on the loaded fuel assemblies and burnable poison assemblies matched the identification numbers used in the MCEI 0400-147 evaluation documents. The inspectors observed portions of initial cask loading and moving per procedure OP/0/A/6550/028, NAC UMS Fuel Assembly Loading/Unloading Procedure, to verify that actual activities were accomplished in accordance with the procedures.

b. Observations and Findings

No findings of significance were identified. Overall, the licensee established and maintained adequate oversight for the dry cask storage evolution. TS requirements and acceptance criteria as outlined in the FSAR for the NAC-UMS casks and the procedures were followed appropriately. The licensee encountered a recurring problem in completing the vacuum drying process of the spent fuel storage cask within the NAC-UMS technical specification allowed time of 52 hours following fuel loading. The previous occurrence was discussed in Inspection Report 05000370/ 2004009. After returning the cask to in-pool cooling for 24 hours, the licensee reestablished vacuum drying and was able to meet the TS requirements.

.2 Independent Spent Fuel Storage Installation

a. Inspection Scope

Access controls and surveillance results for the licensee's ISFSI activities were evaluated. The evaluation included review of ISFSI radiation control surveillance procedures and assessment of ISFSI radiological surveillance data. During tours of the ISFSI cask loading areas and storage facilities, the inspectors observed access controls, TLD locations, material condition, and radiological postings on the perimeter security fence. Radiation controls for the initial loading of the NAC-UMS storage casks and resultant occupation doses were reviewed and discussed in detail. The inspectors conducted independent radiation surveys of the general areas and selected casks currently maintained within the established ISFSI Storage Pad area. Survey results were compared to licensee survey data and established postings.

Program guidance, access controls, postings, equipment material condition and surveillance data results were reviewed against details documented in applicable sections of the UFSAR; 10 CFR Parts 20 and 72; applicable Certificates of Compliance and TS details; and licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in the Attachment to this report.

b. Findings

Section 2OS1 discusses a finding associated with an ISFSI area quarterly radiation survey.

.3 (Closed) Temporary Instruction (TI) 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) - Unit 2 only

The inspectors reviewed the licensee's 60-day response to NRC Bulletin 2004-01, dated July 27, 2004 and the licensee's supplemental response to NRC Bulletin 2004-01, dated September 21, 2004. The inspectors verified that the licensee's inspection activities conducted during this outage were consistent with the their response.

The inspectors conducted an independent walk-down of the top of the pressurizer to ensure that the physical conditions of the pressurizer penetrations and welds were clean and accessible for the prescribed inspections, and that there were no problems with debris, insulation, dirt, boron from other sources, physical layout, or viewing obstructions, which could have interfered with the identification of relevant indications. Specifically, the inspectors observed or reviewed documentation for the following components:

d. 2PZR-W3SE, Pressurizer Relief Nozzle, ASME Class 1
e. 2PZR-W4ASE, Pressurizer Safety Nozzle, ASME Class 1
f. 2PZR-W4BSE, Pressurizer Safety Nozzle, ASME Class 1
g. 2PZR-W4CSE, Pressurizer Safety Nozzle, ASME Class 1
h. 2PZR-W11, Pressurizer Manway Insert, ASME Class 1
i. 2PZR-W2SE, Pressurizer Spray Nozzle, ASME Class 1

Reporting Requirements are as follows:

a. For each of the examination methods used during the outage, was the examination:

1. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes. The licensee used knowledgeable staff members certified as Level II, VT-2 examiners in accordance with procedure NDE-B, "Training, Qualification, and Certification of NDE Personnel," to conduct a direct visual examination of the bare metal surface of the above components. This qualification and certification procedure referenced the industry standard ANSI/ANST CP-189, "Standard for Qualification and Certification of Nondestructive Testing Personnel."

2. Performed in accordance with demonstrated procedures?

   Yes. The inspectors observed the licensee performing the bare metal inspection of the pressurizer penetrations in accordance with procedure QAL-15, "Inservice Inspection (ISI) Visual Examination, VT-2, Pressure Test." The licensee considered this procedure to be demonstrated because examination personnel could resolve a specific size of lower case alpha numeric characters at the actual visual examination distance. The inspectors observed the inspection personnel performing this visual quality check.

3. Able to identify, disposition, and resolve deficiencies?

   Yes. The inspectors concluded that the licensee's direct visual examinations were capable of detecting leakage from cracking in pressurizer penetrations if it had existed. This conclusion was based upon the inspectors direct observations of pressurizer penetration locations, which were free of debris or deposits that could mask evidence of leakage in the areas examined. The inspectors also verified that the licensee's procedures included guidance for proper disposition and investigation of any identified deficiencies.

4. Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01?

The inspectors verified that the licensee's examination personnel were capable of identifying any leakage in pressurizer penetration nozzles or steam space piping components.

b. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

Through discussions with licensee personnel, the inspectors verified that the metal reflective insulation had been removed with extreme caution so as not to disrupt any potential indications of boric acid leakage from the pressurizer at these penetration locations. The licensee personnel performed a direct visual inspection of these pressurizer penetrations. Based on this examination, the area examined was clean and free of debris or deposits or other obstructions which could mask evidence of leakage.

c. How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)?

The licensee's inspection personnel used the direct visual examination technique along with a handheld mirror.

d. How complete was the coverage (e.g., 360° around the circumference of all the nozzles)?

The licensee was able to view the entire circumference, 360°, around each penetration.

e. Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized?

The examination personnel were appropriately trained and qualified to identify small boron deposits as described in the bulletin.

f. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

There were no deficiencies identified that required repair.

g. What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)?

There were no impediments for an effective examination.

h. If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations?

Not Applicable. No augmented surface or volumetric examinations were performed. In accordance with the licensee's response, only a BMV examination was conducted this outage, and there were no indications identified that required further examination.

i. Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system?

Not Applicable. There were no indications of boric acid leaks from pressure-retaining components in the pressurizer system.

.4 (Closed) URI 05000369/2004008-002, Potential for Multiple MSIV Inoperability

This URI, which was opened in PI&R Inspection Report 05000369,370/2004008, was discussed further and left open in Integrated Inspection Report 05000369,370/2004006 pending further investigation into the root causes for the degradation of MSIVs 1SM-1 and 1SM-3. The root causes have been completed and this URI is now considered closed.

Introduction: A Green NCV was identified for failing to take timely and adequate corrective actions to resolve adverse conditions that resulted in two MSIVs being inoperable beyond their TS 3.7.2 LCO Allowed Outage Time (AOT). Specifically, prompt corrective actions in regard to inadequacies with the corrective maintenance procedure and deficiencies identified in engineering evaluations, were not taken to preclude degraded conditions for MSIVs 1SM-1 and 1SM-3. The conditions prevented the main poppet for 1SM-1 from fully stroking closed during a valve stroke time surveillance test and the pilot poppet for 1SM-3 from performing its safety function.

Description: The licensee issued LER 05000369/2004-02 to address two 10 CFR 50.73 reporting requirements: (1) MSIV 1SM-3 inoperable longer than allowed by Technical Specifications (from October 2002 to November 2004); and (2) concurrent inoperability of MSIV 1SM-1, which caused a loss of safety function for the main steam system. The period of concurrent inoperability was from the time 1SM-1 failed its hot stroke test at 10:28 a.m., on October 18, 2004, until when Unit 1 entered Mode 4 at 9:53 p.m. on October 18, 2004.

MSIV 1SM-1

The LER provided no information about the cause for SM-1 failing its hot stroke test. The root cause, identified in PIP M-04-5043 (Rev 0), was that a pre-existing oversized body guide rib (at the 0600 position) combined with flow induced vibration and normal stroking wear of the body guide rib and main poppet guide pad, caused the main poppet to tip and bind two inches from the valve's full open position. Three contributing causes were also identified:

(1) The actuator to valve stem coupling alignment caused scoring at the packing gland and additional stuffing box friction

(2) The side loading of the valve stem by the main poppet caused scoring of the throat bushing and an adverse off center loading condition on the main poppet adding to the main poppet tipping and increased friction affect

(3) Lack of critical maintenance information from the valve vendor

The root cause investigation identified that guide rib wear and inadequate stem guiding on 1SM-1 had been previously identified and documented in the licensee's corrective action program (PIP M-01-1322 , M-01-4120) in March 2001. Inspection of the guide ribs had required engineering evaluation for acceptance of the results since procedure MP/0/A/7200/011, "MSIV and Valve Actuator Corrective Maintenance", had no acceptance criteria for that attribute. The licensee scheduled a detailed inspection of the guide ribs and the installation of several recent MSIV modifications for 1SM-1 for late 2005 (4 ½ years later). On April 27, 2004, during an engineering walk-down, scoring was identified on the stem of 1SM-1. An operability assessment, which was performed by Engineering as a result of the stem scoring, determined the valve was operable. On October 18, 2004, the first stroke after identification of the scoring, the valve failed to stroke. The root cause investigation identified that there were two indicative scoring marks on the stem, one at the 0700 position and one at the 0630 position. The marking at the 0700 position was identified in April 2004 and evaluated. This scoring mark was later attributed to the misalignment of the valve stem and actuator. The licensee failed to identify the 0630 marking during the April 2004 inspection; consequently, it was not evaluated. Later it was determined that the 0630 scoring was caused by stem side loading. Additionally, the improper misalignment between the valve stem and the actuator caused the stem to bear against the stem guiding set and cause additional scoring. Both of these conditions were the result of inadequate guidance from the maintenance procedure, MP/0/A/7200/011, "MSIV and Valve Actuator Corrective Maintenance".

The inspectors' review of the LER, the root cause investigation, and associated historical documents, revealed that the April 2004 operability assessment had not adequately evaluated the scoring marks on the stem, particularly in relation to the pre-existing deficiencies of valve 1SM-1 and historical valve failures of similar MSIVs in the industry. The inspectors' interview of licensee personnel confirmed that the 0630 scoring mark was observable in April 2004. Consequently, the conditions that were attributed in the root and contributing causes were present in April 2004 and, therefore, the inspector concluded that the valve was inoperable in April 2004. The failure to properly evaluate valve operability in April 2004, was not addressed in the licensee's root cause assessment nor were there corrective actions identified to address that inadequacy.

It was clear to the inspectors from the root cause that procedure MP/0/A/7200/011 did not contain the appropriate acceptance criteria to identify and correct adverse conditions with respect to the valve body guide ribs. Additionally, the procedure lacked the guidance for properly assembling and aligning the valve stem and actuator.

<u>MSIV 1SM-3</u>

The LER attributed the valve failure to improper reassembly during a 2002 refueling outage maintenance activity caused by deficiencies in instructions provided in the procedure used to maintain, re-assemble, and test MSIVs. The failure to recognize that the abnormal stroke length identified during the October 7, 2002, stroke test rendered 1SM-3 inoperable, was attributed to inadequate stroke length acceptance criteria in the procedure used to maintain, reassemble, and test the MSIVs. In the LER, the licensee indicated that upon further evaluation of results of the safety significance, any conclusions different from those stated in the LER would be revised.

The inspectors' review of the LER, the root cause investigation contained in PIP M-04-5315, Revision 1, and associated documents, revealed that the test performed in 2002 had identified a deficiency in the stroke of valve 1SM-3 that was inappropriately accepted. The engineering acceptance of this deficiency was attributed in the LER to inadequate acceptance criteria in the procedure. The inspectors found that the procedure acceptance criteria had identified an equipment performance deficiency, and although enhancement of the acceptance criteria, based on current knowledge, would be useful, it does not explain why the equipment performance deficiency was inappropriately determined to be insignificant. The inspectors recognized that the inappropriate determination was a human performance deficiency. In addition, the corrective actions identified in the LER and root cause did not address corrective actions for the human performance deficiency, although contributing cause #1 in the root cause identified a lack of understanding of the pilot stroke mechanism.

As a result of the review of the MSIV failures and other performance issues associated with the determinations of operability issues brought up by the inspectors, the licensee issued a trend PIP M-05-5615 to address this issue.

<u>Analysis</u>: The concurrent inoperability of valves 1SM-1 and 1SM-3 is considered a performance deficiency because inadequate maintenance practices and procedures, and inadequate and untimely corrective actions by the licensee resulted in the failure of two main steam isolation valves during their respective full temperature and pressure valve stroke timing test. The finding is considered greater than minor because it had a direct impact on the MSIVs to perform their safety function, which is to close during a high energy line break or steam generator tube rupture. The finding affects both the Mitigating Systems and Barrier Integrity cornerstones, in that the failure to close impacts the equipment performance (reliability, availability) attribute and containment isolation (minimization of radiological releases) attribute, respectively.

The as-found configuration of valve 1SM-3 precluded the pilot poppet from being capable of performing its pressure equalization function. The pilot poppet typically acts as a "check valve" to balance a significant differential pressure present across the main poppet. This function is significant when a break occurs between the valve and the steam generator and increased downstream steam pressure exists from one of the other MSIVs not shutting, such as 1SM-1. In this case, the pilot poppet would not work, building a significant differential pressure, greater than the actuator closing springs were sized to mitigate, resulting in the main poppet essentially being lifted off the valve seat.

Due to the concurrent failure of main steam isolation valve 1SM-1 and 1SM-3, the significance of this finding was evaluated per IMC 0609 Appendix A, Section IV, "Treatment of Concurrent Multiple Equipment or Functional Degradations," to determine whether the failures were related to a common cause. Based on the inspection findings, the root cause investigations, and additional relevant documents, the inspectors determined the two concurrent degradations of MSIVs 1SM-1 and 1SM-3 were due to a common deficiency (i.e., an inadequate maintenance procedure due to lack of appropriate qualitative and quantitative acceptance criteria and proper assembly instructions). Albeit, different sections of the maintenance procedure caused the affects, the same inadequate procedure led to improper personnel/equipment interface, causing the degradation of two identical safety-related components (i.e., MSIVs) during overlapping time periods. The exposure time for this concurrent condition was greater than 30 days (from April 2004 until October 2004) which encompasses the overlapping time periods both MSIVs were inoperable. As a result of the Phase 2 SDP Sheets, three sequences appear to be greater than green: Sequence 1 (High Pressure Injection Termination: E-6) and Sequence 6 (Overcooling Transient: E-4) under "Main Steam Line Break"; and Sequence 7 (High Pressure Injection with Pressure Equalization: E-6) under "Steam Generator Tube Rupture." Accordingly, a Regional Senior Reactor Analyst performed an SDP phase 3 assessment. Because the NRC's understanding of the Pressurized Thermal Shock issue has changed since the plant SDP notebooks have been issued, the high value sequences that caused the finding to be greater than Green in the Phase 2 SDP have decreased in importance. The change in core damage probability due to the finding was evaluated to be within the Green band, with the Steam Generator Tube Rupture initiator contributing the most. Large Early Release frequency was considered to be Green, due to the timing of the release during the tube rupture, if secondary side cooling is available, and due to the low worth of the sequences containing failure of the secondary side cooling. Accordingly, the finding was determined to be of very low safety significance (Green). This issue contained elements of problem identification and resolution, as well as human performance, as it involved failures to properly evaluate data and deficiencies associated with the MSIVs; therefore, failing to take prompt corrective action to preclude the valves from becoming inoperable.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, from March 2001 until November 4, 2004, the licensee failed to adequately resolve significant conditions adverse to quality related to MSIVs to preclude the valves from becoming inoperable as evidenced by MSIV 1SM-1 failing to fully close on October 18, 2004 and MSIV 1SM-3 failing to fully close its pilot poppet on November 4, 2004. The degraded conditions caused both MSIVs 1SM-1 and 1SM-3 to be inoperable beyond the TS AOT limits.

(1) In March 2001, the licensee was aware valve body guide rib wear was occurring and valve misalignment existed on 1SM-1, but actions to resolve the conditions

and preclude repetition were scheduled for four years later (October 2005). The corrective maintenance procedure was not modified to include any additional inspection guidance for the body guide ribs. When stem scoring reappeared on 1SM-1 in April 2004 (3 years later), the licensee failed to consider the previously known deficiencies with the MSIV, failed to identify the occurrence of indications that the previous condition (stem side loading) had recurred, and consequently, failed to promptly test the valve to verify operability.

(2) In October 2002, despite diagnostic testing results indicating a short pilot poppet stroke, the licensee failed to identify and correct the fact that MSIV 1SM-3 had been rendered inoperable due to improper post-modification reassembly and testing from a deficient procedure. This was later confirmed through subsequent diagnostic testing and valve disassembly in November 2004.

Because this issue was of very low safety significance and was placed in the corrective action program as PIPs M-04-5043, M-04-5315, and M-05-5615 this violation is being treated as a non-cited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000369/2005002-09, Multiple MSIV Inoperability.

4OA6 Meetings, including Exit

Exit Meetings

On April 5, 6, and 19, 2005, the resident inspectors presented the inspection results to Mr. G. Peterson and other members of his staff. The inspectors noted that any proprietary information reviewed during the course of the inspection would not be included in the documented report.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for disposition as an NCV.

- TS 5.7.2 requires persons entering areas with general area dose rates exceeding 1000 mrem/hr to be under continuous surveillance of personnel qualified in radiation exposure to provide positive exposure control over activities being performed in the area. Contrary to the above, on March 30, 2004, two individuals entered the U1 Lower Containment Pipe Chase ‘A’ Sump EHRA boundary, a location where general area dose rates exceeded 1000 mrem/hr, without continuous surveillance by qualified personnel. This item is documented in the licensee’s CAP under PIP M-04-1945. This event is of very low safety significance because it did not involve entry into a very high radiation area nor the substantial potential for worker overexposure.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

Black, D., Security Manager
Bradshaw, S., Superintendent, Plant Operations
Nolin, J., Chemistry Manager
Brown, S., Manager, Engineering
Crane, K., Technical Specialist
Evans, K., Manager, Mechanical and Civil Engineering (MCE)
Harrall, T., Station Manager, McGuire Nuclear Station
Kammer, J., Manager, Safety Assurance
Loucks L., Radiation Protection Manager
Parker, R., Superintendent, Maintenance
Peterson, G.., Site Vice President, McGuire Nuclear Station
Thomas, J., Manager, Regulatory Compliance
Thomas, K., Manager, RES Engineering
Travis, B., Superintendent, Work Control
Hatley, M, SG Project Manager
Mayes, D., SG Engineer Level III
Branch, R., ISI Nuclear Assurance
Grass, L., QA/LWS

NRC personnel

M. Ernstes, Chief, Reactor Projects Branch 1
J, Shea, Project Manager, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000370/2005002-01	URI	Failure to Have Adequate Procedures to Implement SLC Test Requirements for Fire Protection Sprinklers (Section 1R05)
05000369,370/2005002-04	URI	2SM-1 MSIV Fails to Close During Surveillance Testing (Section 1R22b.(2))
05000369,370/2005002-07	URI	Review Licensee Assessments and Vendor Evaluations for Observed U1/U2 Unit Vent Volume Flow Rate Changes to Assure Representative Sampling (Section 2PS1).

Opened and Closed

05000369,370/2005002-02	NCV	Failure to Comply With RCS Leakage Detection TS for Containment Radiation Gaseous Monitors (Section 1R15)
05000369,370/2005002-03	NCV	Failure to Have Adequate Surveillance Procedures for RCS Leakage Detection Instrumentation (Section 1R22b.(1))
05000369,370/2005002-05	NCV	Failure to Follow Procedural Guidance for Conducting ISFSI Radiation Surveys (Section 2OS1)
05000369,370/2005002-06	NCV	Failure to Provide Adequate Breathing Air Capacity for Supplied-Air Respiratory Equipment (Section 2OS3).
05000369/2005002-08	NCV	Failure to Report a Condition Prohibited by Technical Specifications (Section 4OA3.1)
05000369/2005002-09	NCV	Multiple MSIV Inoperability (Section 4OA5.4)

Closed

05000369/2004008-02	URI	Potential for Multiple MSIV Inoperability (Section 4OA5.4)
05000369/2004-002	LER	Main Steam Isolation Valve Inoperable (Section 4OA3.1)
2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) - Unit 2 only (Section 4OA5.3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

1A KC
OP/1/A/6800/016, 1A KC System Block Tagout, Rev. 7
OP/1/A/6400/005A, Component Cooling Water System Valve and Power Supply Checklists, Rev. 16
MCFD-1573-01.00, Flow Diagram of Component Cooling System, Rev. 8

2B CVCS and HHSI
MCFD-2554-03.00, Flow Diagram of Chemical and Volume Control System (NV), Rev. 6
MCFD-2554-03.01, Flow Diagram of Chemical and Volume Control System (NV), Rev. 16
MCFD-2562-01.00, Flow Diagram of Safety Injection System (NI), Rev. 2

1B EDG
MCFD-1609-04.00, Flow Diagram of the Diesel Generator Starting Air System
MCFD-1609-03.00, Flow Diagram of the Diesel Generator Engine 1A Fuel Oil System
MCFD-1609-02.00, Flow Diagram of the Diesel Generator Engine Lube Oil System
MCFD-1609-01.00, Flow Diagram of the Diesel Generator Engine Cooling Water System

2A & B MDCA
OP/2/A/6250/002, Auxiliary Feedwater System, Enclosure 4.8, Rev. 66
MCFD-1592-01.01, Flow Diagram of Auxiliary Feedwater System, Rev. 14

Section 1R05: Fire Protection

McGuire Nuclear Station IPEEE Submittal Report dated June 1, 1994
McGuire Nuclear Station Supplemental IPEEE Fire Analysis Report dated August 1, 1996
MCS-1465.00-00-0008, R4, Design Basis Specification for Fire Protection

Section 1R08: Inservice Inspection and Activities

Nondestructive Examination
NDE-35, Liquid Penetrant Examination, Rev. 20
ESD Boric Acid Corrosion Control Program, Rev. 1
NSD 413, Fluid Leak Management Program, Rev. 3
WPS GTSM0808-01, GTAW and SMAW Welding Procedure
QAL-15, Inservice Inspection (ISI) Visual Examination, VT-2, Pressure Test, Rev. 20
NDE-600, Ultrasonic Examination of Similar Metal Welds in Ferritic and Austenitic Piping, Rev. 15

Steam Generator
Condition Monitoring and Operational Assessment for McGuire Unit 2 EOC14
SGMEP 103, Operational Assessment, Rev. 3
SGMEP 104, Condition Monitoring, Rev. 4
SGMEP 105, CFR80 Specific Assessment of Potential Degradation Mechanisms, Rev. 5

CFR80 Steam Generator Site Technique Validation for Catawba Nuclear Station Unit 1 and McGuire Nuclear Station Units 1 & 2, Rev. 5
Eddy Current Analysis Guidelines for Duke Power Company's CFR80 Steam Generators, Rev. 6
Field Procedure for Remote Rolled Plugging Utilizing the LAN SAP Box, Rev. 10
Field Procedure and Operating Instructions for Installation of a Flexible Stabilizer in a Recirculating Steam Generator, Rev. 17

Other Documents
Boric Acid Corrosion Program Health Report, 2004T1

PIPs
M-05-00186, Failure to locate loose part in SG A during 1EOC16.
M-05-01322, Loose parts found in the 2A S/G secondary side FOSAR.
M-05-01372, Eddy current examination of the 2C steam generator tubing has identified potential loose parts in contact with 13 tubes.
M-05-01431, 9 tubes require plugging in the 2A SG
G-05-00098, Inservice Inspection percentage of coverage documentation
M-05-00913, Results of 2EOC16 Reactor Vessel Bottom Head Inspection
M-05-01488, Review of work scope in the ISI Plan for 2EOC16
M-05-01389, 2EOC16 Reactor Vessel ISI Re-Inspection of Indentation in Cladding
M-03-04270, An indication was found during the PT inspection of weld 2NI2FW24-6

Self-Assessments and Audits
SA-ISIM-03-001 (PIP-G-03-00492), Self Assessment for Inservice Inspection Pressure Test Plan Reporting.
GO-03-07, Fluid Leak Management Program Assessment Report

Section1R12: Maintenance Effectiveness

Flow Accelerated Corrosion Program, Unit 2 EOC-16
MCC-1206.00-05-0068, Pipe Erosion /Corrosion Control Program - System Susceptibility Analysis, Rev. 1
MCFD-2594-01.01, Moisture Separator Reheater Bleed Steam System (HM), Heater Bleed System (HA, HB), Rev. 5
PIP M-04-5393, The 1.5" pipe upstream of valve 1HS083 is leaking
PIP M-03-3960, 2HM-23 did not close as required
PIP M-98-4159, Steam Leak on HV system elbow
PIP M-03-3272, 2HM-113 has a steam leak due to apparent corrosion
PIP M-04-3350, A through wall leak discovered on HM piping downstream of 2HM-106
PIP M-04-0005, Computer alarms fro 2B1 MSR drain tank high level
Metallurgy File 3390- MNS 2: Ruptured Extraction Line at 2HM-23

Section1R14: Personnel Performance During Nonroutine Plant Evolutions

AP/2/A/5500/001, Steam Leak, Rev. 13
AP/2/A/5500/004, Rapid Downpower, Rev. 15
PT/0/4700/045, Reactor Trip Investigation

Section1R15: Operability Evaluations

Standard Review Plan, Sec.6.2.4 Containment Isolation System, Rev. 2, dated July 1981
Updated Final Safety Analysis Report Section 6.2.4, Containment Isolation System

Section1R20: Refueling and Outage Activities

MCEI-0400-41, “McGuire 2 Cycle 16 Final Core Map”, Rev. 11
PT/0/A/4550/003C, “Core Verification”, Rev. 15
PT/0/A/4150/033, “Total Core Reloading”, Rev. 43
MP/2/A/7150/073, “Rod Cluster Control Assembly Heavy Drive Rod Unlatching and Latching”, Rev. 14
OP/2/A/6100/003, Controlling Procedure For Unit Operation
PT/0/A/4150/021, Post Refueling Controlling procedure for Criticality, Zero Power Physics, & Power Escalation Testing
PT/0/A/4150/028, Criticality Following a Change in Core Nuclear Characteristics
PT/0/A/4150/013, Boron Endpoint, Dynamic Rod Worth and Isothermal Temperature Coefficient Measurement
MCEI-0400-47, Unit 2 Cycle 16 Core Operating Limits Report

<u>Mid-Loop</u>
GL 88-17 response
OP-MC-CP-PRO, R22, Pre-Refueling Outage(PRO) Lesson Plan
SLCs 16.5.1, 16.5.2, 16.5.3, 16.5.4, and 16.5.5
OP/2/A/6100/SO-1, Maintaining NC System Level

Section 1R22: Surveillance Testing

PT/1/A/4150/001B, Reactor Coolant Leakage Calculation, Rev. 50
PT/2/A/4150/001B, Reactor Coolant Leakage Calculation, Rev. 45
PT/1/A/4600/003A, Semi-Daily Surveillance Items, Rev. 108
PT/2/A/4600/003A, Semi-Daily Surveillance Items Rev. 81
Work Order (WO) 98700424, PT 1EMF-39L, Containment Gas Radiation Monitor Lo Range, 1/19/05
WO 98691893, PT 1EMF-38L, Containment Particulate Radiation Monitor Lo Range, 12/6/04
WO 98690669, PT 2EMF-39L, Containment Gas Radiation Monitor Lo Range, 11/30/04
WO 98690668, PT 2EMF-38L, Containment Particulate Radiation Monitor Lo Range, 11/31/04
Analysis of MNS 2SM-1 MSIV Debris Samples, dated March 11, 2005
PIP M-05-0841, 2SM-1 fails to stroke during shutdown VST

Section 1R23: Temporary Plant Modifications

Engineering Change Package MD200238 (Unit 2)
Drawing MCFD-2553-02.01, Flow Diagram of Reactor Coolant System (NC)
Work Order: 98715827 01 (Unit 2)
PIP M-05-00833

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals
Health Physics Procedure (HP)/0/B/1003/063, Routine Surveillance, Rev. 14
HP/1/B/1006/012, Controls for Reactor Building Entry Under Power, Rev. 003
HP/0/B/1005/061, Operation of Radiation Protection Portable Survey Instruments, Rev. 001
HP/0/B/1005/079, Source Check of Inservice Radiation Protection Portable Survey Instruments, Rev. 003
Shared Health Physics Procedure (SH)/0/B/2000/004, Taking, Counting and Recording Surveys, Rev. 006
SH/0/B/2000/007, Placement of Personnel Dosimetry for Non-Uniform Radiation Fields, Rev. 001
SH/0/B/2000/009 Neutron Dose Tracking, Rev. 002
SH/0/B/2000/012, Access Controls for High, Extra High, and Very High Radiation Areas, Rev. 3
SH/0/B/2001/003, Investigation of Skin and Clothing Contamination, Rev. 006
SH/0/B/2002/001, Multiple Dosimetry, Rev. 4
Radiation Protection Management Procedure (RPMP) 3-2, Electronic Dosimeter Alarms, Rev. 000
RPMP 7-1, Key Control, Rev. 000
RPMP 7-6, Administrative Controls of Yellow Flashing Light Process, Rev. 0
RPMP 7-13, Supplemental Guidelines For Controls to Extra High Radiation Areas Greater Than 10 Rem Per Hour, Rev. 000
RP-0414, ETQS, Taking, Counting and Recording Surveys, Rev. 6
Radiation Work Permit (RWP) 18, Miscellaneous Valve Maintenance, Rev. 18
RWP 26, Reactor Building Pipe Chase and Seal Table Entry During Power Operations MNS/CNS Only, Rev. 17
RWP 1829, Unit 1 (U1) Reactor Building: Miscellaneous Testing by Operations Test Group, Rev. 8

Records and Data
Active Radiological Key Roster, 12/27/2004
McGuire Nuclear Station Calendar Year (CY) 2004 Maximum Occupational Workers Doses including Total Effective Dose Equivalent, Lens of Eye, Committed Dose Equivalent; Committed Effective Dose Equivalent; and Skin Dose Equivalent
RWP 26, Job Dose Card Report Data January 1 - 28, 2005
Unusual Dosimeter Occurrence (UDO) Log CY 2004
UDO Nos. 04-63, dated 07/01/04; 04-67, dated 08/03/04, ; 04-71, dated 09/14/04; 04-74, dated 10/18/04; and 04-78, dated 12/14/04
Personnel Contamination Event (PCE) Summary Data: Nos 04-68, completed 03/18/04; 04-129, completed 03/31/04; 04-144, completed 04/01/04; 04-156, completed 08/04/04; and 04-165, completed 12/14/04
Clean Area Surveys : 774 foot (‘) elevation hallways outside of RCA 3/1/04 and 4/1/04; Room 1106-1107 Primary Chemistry Staff Offices, 3/1/04
MNS Survey Data Sheets: Unit 1 725 foot (‘) elevation for Containment Pipe Chase ‘A’ Sump conducted 03/10/04 @ 05:20, 12:45, and 20:00 hours (hrs); 03/11/04 @ 04:20 hrs; and 03/31/04 @ 19:25 hrs.
MNS Survey Data Sheets for: U1 Lower Containment Pipechase conducted 03/16/04 @ 01:30 hrs; 03/11/04 @ 01:30 and 09:00 hrs; 03/14/04 @ 19:00 hrs; 03/15/04 @ 09:30 hrs;

03/18/04 @ 07:00 hrs; 03/20/04 @ 07:30 hrs; 03/29/04 @ 23:00 hrs; and 04/01/04 @ 04:30 hrs

MNS Survey Data Sheets for NAC Cask Loading Activities Including: Survey Number (No.) M-011605-1, dated 01/15/05; Nos. M-011605-2, M-011605-3, M-011605-4, and M-011605-5, dated 01/16/05; and Nos. M-011705-4, and M-011605-7, dated 01/17/05

PIPs

PIP M-02-01023, Radiography Boundary Crossed by Non-Radiographic Person, 03/02/02
PIP M-04-01495, TLD and SRD Lost Within Radiologically Controlled Area (RCA) and Radiation Control Zone (RCZ) and not Found, 03/17/04
PIP M-04-1496, Electronic Dosimeters not Turned On Prior to Use in Multipacks, 3/17/04
PIP M-04-01819, Event Investigation Team (EIT) Identified Items Associated with 1EOC16 Significant Dose and Contamination Challenges, 03/26/04
PIP M-04-01862, Dosimetry Cannot Read Multi-pack, Vendor Did Not Log Out of Multipack on the EDC System, 03/28/2004
PIP M-04-01945, Two Individual Observed Inside of Posted Extra High Radiation Area, 03/30/2004
PIP M-04-02027, Evaluation of Multiple Issues Associated with Multiple Dosimetry Use, 04/01/04
PIP M-04-02428, Assess Radiation Protection Challenges Related to Personnel Contaminations Associated with U1 EOC16
PIP M-04-03142, Monthly TLD Changeout for Pregnant Worker was not Performed for June 2004, 06/15/04
PIP M-04-03598, Results of Radiation Protection Outage Assessment (RP-SA-04-06), U1 End of Cycle 16 Cavity Decon & Upper Containment Contamination Control, 7/15/04
PIP M-04-04394, Pre-Job Survey Did Not Match As Found Conditions
PIP M-04-05347, Flashing Lights at Pressurizer Not Working, 11/08/04
PIP M-04-05533, RWP Number Corrected in Radiation and Control Computer Data Base by Unqualified Technician, 11/23/04

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment

Procedures, Guidance Documents, and Manuals

License Amendments Nos. 199 to NPF-9 and No. 180 to NPF-17, Elimination of Post Accident Sampling Requirements, Issued September 17, 2001
Performance Test (PT)/0/A/4457/010, Sampling the Ingersoll-Rand SCBA Air Compressor for Grade “D” Quality Air
SH/0/B/2008/001, Calibration and Quality Assurance of Canberra ARGOS-4AB Contamination Monitors, Rev. 1
HP/0/B/1005/066, Response Checks of Personnel Monitoring Equipment, Rev. 13
HP/0/B/1008/006, Respiratory Protective Equipment Maintenance and Storage, Rev. 11
HP/0/B/1008/011, Respiratory Equipment Use, Rev. 9
NSD 208, PIP, Rev. 27
RP/0/A/5700/019, Core Damage Assessment, Rev. 004
RP/0/A/5700/000, Classification of Emergency, Rev. 011

Records
U1 Reactor Building Incore Room ARM, 1-EMF-9, Calibrations, 03/13/01 and 10/05/02
Control Room Air Radiation Monitor, EMF-43A, Calibrations, 11/02/02 and 06/02/04
U2 Containment Hi-Range ARM, 2-EMF-51A, Calibrations, 03/06/02 and 09/23/03
10 CFR Part 61 Analysis, Dry Active Waste Smear Sample, 02/05/04
SCBA Breathing Air Quality Analysis, 11/09/04
SCBA Air Bottle No. 3260 and SCBA Regulator Nos. 27412, 27048, and 27049, Maintenance History, January, 2000 - January 2005
SCBA Qualification Records, 21 Operations and 6 Maintenance Personnel, Shift Crews On-Duty 01/27/05
MSA Certifications, 3 Individuals Qualified to Maintain/Repair SCBA Components, 03/01/04
Certificates of Calibrations for RO-20 Ion Chamber Serial Number (S/N) 1373, dated 10/26/04; RO-20 Ion Chamber S/N 1447, dated 11/30/04; 6112B Teletector, S/N 14375, dated 01/17/05; REM500 S/N 102, dated 09/01/04; RM-14 Frisker S/N 5286, dated 10/26/04, and 3090-3 GM Gamma-Alarm S/N133-892, dated 08/04/04

PIPs

PIP M-01-01949, Tracking of PORC Action Items Regarding Proposed Elimination of Post Accident Sampling System, 04/16/01
PIP 04-01757, Breathing Air Compressor Did Not Have Adequate Capacity for the Number of ‘Delta’ Suits in Use, 03/25/04
PIP 04-05592, EMF-38L and EMF-39L (Containment Air Radiation Monitors) Alarm Setpoint Change, 12/01/04
PIP 04-05770, 2EMF-38 Spiked > 700 cpm, 12/13/04
PIP 04-04394, A Pre-job Survey Using a Teletector Did Not Reflect Actual Work Conditions, 09/02/04
PIP 04-00507, Documentation of Comments from 2004 RP Functional Area Evaluation

Section 2PS1: Radioactive Gaseous & Liquid Effluent Treatment and Monitoring Systems

HP/0/B/1001/022, Calibration of the Gamma Spectroscopy System, Rev. 5
HP/0/B/1001/024, Quality Assurance for the Gamma Spectroscopy System, Rev. 6
HP/0/B/1001/038, Quality Assurance of the Count Room Equipment, Rev. 8
HP/0/B/1001/039, Manual Calculations of Automatic Count Room Reports, Rev. 7
HP/0/B/1001/045, Calibration of Packard Tri-Carb 2900 TR Series Liquid Scintillation Counter, Rev. 1
HP/0/B/1003/049, Waste Monitor Tank Release, Rev. 11
HP/0/B/1003/050, Waste Gas Decay Tank Sampling and Release, Rev. 11
HP/0/B/1003/036, Unit Vent, Rev. 16
Chemistry Procedure (CP)/0/B/8600/001, Radwaste Sampling, Rev. 26
Operations Procedure (OP)/0/B/6200/106, Liquid Waste Release - WMT A with WMT Pump A, Rev. 14

Records and Data Reviewed
Gamma Spectroscopy System Calibrations for Detectors Nos. 2 and 4, conducted 09/03
Gamma Spectroscopy System Quality Assurance (QA) Data for Detectors Nos. 1, 2, 3, and 4, conducted 1/25/05; Quarterly QA Summary Data for Detector No. 4 for 2002-2003.

Liquid Scintillation Detector Calibrations, Semiannual IPA Reviews, and Annual Quench Curve Verifications for Detector No. 1 dated 1/19/05, 9/7/04, 3/11/04, and 9/16/03.
Liquid Scintillation Detector QA Data for Detectors Nos. 1, 2, conducted 1/25/05
Counting Room Inter-Laboratory Cross-Check Results, 2nd Quarter 2003 - 3rd Quarter 2004
Liquid Waste Release (LWR) Permit 2005005, Waste Monitor Tank “A”, 1/25/05
LWR Permit 2004211, Conventional Waste Water Treatment, 11/1/04-12/1/04
LWR Permit 2004212, Waste Water Collection Basin, 11/1/04-12/1/04
LWR Permit 2004188, Waste Monitor Tank “B”, 11/11/04
Gaseous Waste Release (GWR) Permit 2004005, “B” Waste Gas Decay Tank, 1/15/04
GWR Permit 2004086, U1 Vent, 11/1/04-12/1/04
GWR Permit 2004087, U2 Vent, 11/1/04-12/1/04
Primary-to-Secondary Calibration Documentation and Calibration Source Certificates for effluent monitors 1/2-EMF-31, 1/2-EMF-33, 1/2-EMF-35L, 0-EMF-41, and 0-EMF-50
2-EMF-35L, U2 Unit Vent Particulate Monitor Calibration Records, Conducted 6/27/02 and 12/15/03
2-EMF-36, U2 Unit Vent Gas Radiation Monitor Calibration Records, Conducted 4/02/02 and 8/11/03
2-EMF-37, U2 Unit Vent Iodine Monitor Calibration Records, Conducted 6/26/02 and 2/24/04
0-EMF-41, Auxiliary Building Ventilation Radiation Monitor Calibration Records, Conducted 2/17/03, 4/19/04, 7/6/04, and 9/27/04
1-EMF-44, U1 Containment Vent Drains Radiation Monitor Calibration Records, Conducted 11/28/01 and 10/28/03
0-EMF-49, Waste Liquid Radiation Monitor Calibration Records, Conducted 11/28/01 and 1/28/04
0-EMF-50L, Waste Gas Radiation Monitor Calibration Records, Conducted 11/27/01 and 8/28/03
Minor Modification Notice MGMM14236, Perform an Alternate Transfer Calibration for 0EMF50L, 8/18/03
U1 Unit Vent Exhaust Air Flow Monitor Analog Channel Operational Tests, Conducted 7/20/04 and 11/02/04
U1 Auxiliary Building Ventilation Hepa/damper and Carbon Adsorber Filters In-place Leak Test Results, Conducted 5/5/03 and 9/22/04
MCM 1346.05-0075.001, Installation Isokinetic Nozzle, Unit Vent Particulate, Iodine, Gas Monitor drawing, Rev. D1
Cumulative Offsite Dose from Liquid and Gaseous Effluents, November 2004, Conducted 1/4/05

PIPs

M-97-0384, Evaluation Needed for Optimum Sampling Rate for Unit Vent Isokinetic Sampling, 2/3/97
M-02-5770, 1-EMF-31 Was Isolated and Inoperable on 11/13/02 10:02 but Not Declared in TSAIL until 11/14/02 04:47
M-03-1348, Sample Connections from Various Radiation Monitors Obtained from “Tee” Connection, 3/24/03
M-03-1798, While aligning WG Compressor A for Service, Received Call from RP about EMF-41 alarm. Resulted in an Unplanned Waste Gas Release, 4/21/03
M-04-4329, 1-EMF-35L Sample Flow Rate Appears Non-conservative for Isokinetic Sampling, 8/31/04

M-03-2830, Documentation of Self-Assessment RES-SA03-09, McGuire Radiation Monitoring System Improvement, 6/26/03
M-04-0507, PIP to Address 9 NPA Areas for Improvement Identified During the RP Functional Area Audit, 12/27/04
M-04-3041, Chemistry Group Self Assessment in Response to CA #17 of PIP M-03-1798, 6/8/04

Annual Reports
2002 Annual Radioactive Effluent Release Report, dated April 10, 2003
2003 Annual Radioactive Effluent Release Report, dated April 29,2004

Section 2PS3: REMP and Radioactive Material Control Program

Procedures and Guidance Documents
EnRad Laboratories 317, Low Volume Air Sampler Calibration Procedure, Rev. 2
EnRad laboratories 730, Airborne Radioiodine and Airborne Particulate Sampling at McGuire Nuclear Station, Rev. 0
EnRad Laboratories 207, Configuration and Calibration of the ISCO 3710 Water Sampler, Rev. 0
IP/0/B/3260/023, Meteorological Monitoring (EEB) System Weekly Channel Verification, Rev. 24
SH/0/B/2008/001, Calibration and Quality Assurance of Canberra ARGOS-4AB Contamination Monitors, Rev. 1
HP/0/B/1005/066, Response Checks of Personnel Monitoring Equipment, Rev. 13
SH/0/B/2000/006, Removal of Items from RCA/RCZs and Use of Release/Radioactive Material Tags, Rev. 1
NSD 208, PIP, Rev. 27.

Instrument Calibration and Environmental Data Records
2003 Radiological Environmental Operating Report Meteorological Data Recovery Summary, January 2004 - December 2004
Meteorological Tower Weekly Surveillance Records, 12/21/04, 12/28/04, and 2/25/05
Meteorological Tower Calibration Records including 60m/10m Wind Speed, 3/2/04 and 8/5/04; 60m/10m Wind Direction, 3/2/04 and 8/5/04; and Delta Temperature, 3/2/04 and 8/5/04; Environmental Air Sampler Calibration Records including S/N 00289, 5/6/03 and 7/7/04; S/N 00292, 12/3/01 and 4/13/04; S/N 00307, 3/12/03 and 4/15/04; S/N 00311, 5/6/03 and 7/9/04; S/N 00314, 9/16/03 and 9/15/04; S/N 00320, 11/27/01 and 2/24/04; S/N 00325, 9/2/03 and 9/14/04; S/N 00333, 9/3/03 and 9/2/04; S/N 00344, 10/11/02 and 7/7/04; S/N 00349, 10/11/02 and 4/16/04; S/N 00353, 10/10/02 and 6/4/04; S/N 00356, 10/11/02 and 6/30/04; S/N 00358, 1/8/01 and 2/25/04; and S/N 00359, 10/11/02 and 7/7/04;
Environmental Water Sampler Calibration Records including S/N 00276, 1/5/04 and 12/22/04; and S/N 00277, 1/12/04 and 12/22/04
EnRad Laboratories Interlaboratory Cross-check Results, CY 2003 and 1st - 3rd Quarters 2004
SAM-9, Serial No. 27342, Calibration Records, 6/27/02 and 1/19/04
10 CFR Part 61 Analysis, Dry Active Waste Smear Sample, 02/05/04

PIPs
PIP 03-02828, Low levels of tritium identified on the weekly chemistry sample of WZ sump ‘C’, 6/25/03
PIP 04-04161, Unplanned entry into TS due to inoperability of meteorological system, 8/22/04
PIP 03-05010, Turkey Point WBC results for vendors previously working at MNS indicated low-levels of Co-58, 10/8/03
PIP 03-05078, Low levels of Co-58 detected on individual’s shirt during entrance WBC at VC Summer, 10/14/03
PIP 04-02255, Low levels of Co-58 contamination found during WBCs at Robinson and VC Summer, 4/14/04
PIP 04-02535, During entrance WBC at Vogtle, low levels of Co-58 detected on individual’s lanyard, 5/3/04

Section 4OA2: Identification and Resolution of Problems

PIP M-05-0871, During inspection of 2NI-76A, it was determined switch mechanism is Post -78 design
PIP M-05-1082, During inspection of 2KC0413B, it was determined switch mechanism is Post-78 design
PIP M-05-1343, During inspection of 2NI-54A, it was determined switch mechanism is Post -78 design
PIP M-04-4837, 2NI-76A indicates intermediate. Unplanned entry into Ts
PIP M-04-2943, 10 CFR Part 21 Notification for Rotork Controls regarding potential manufacturing concern with plastic PPS components fitted in the switch mechanism assembly, manufactured between 1978 and November 2001
PIP M-04-4890, INOT monitoring of Operability Assessment for 2NI-76A
MGTM-0320,Temp mod to allow stem mounted limit switch to provide remote position indication

Section 4OA5.2: Independent Spent Fuel Storage Installation

Procedures, Guidance Documents
RWP 2202, Unit 2 Auxiliary Building/Yard Dry Cask Storage ISFSI Work, Rev. 10
RWP 5035, Unit-0 Yard: All Work Associated With Independent Spent Fuel Storage Installation Dry Cask Storage Pad, Rev. 2
ALARA Pre-planning Worksheet, Load NAC Dry Storage Cask (#OFTCKN004) 12/04,
HP/0/B/1003/063, Routine Surveillance, Rev. 14
HP/2/B/1006/027, Radiation Protection Controls For Loading Spent Fuel Assemblies Into NAC-UMS Dry Storage Casks, Rev. 002
RPMP 7-8, Maintaining Radiation Control Zones (RCZs) Associated with Independent Spent Fuel Storage Installation (ISFSI), Rev. 001
Certificate of Compliance (CoC) No. 1201, Amendment No. 1, For The Transnuclear, Inc., TN-32 Dry Storage Cask, Effective 2/20/2001
COC No. 1015, Amendment 3, For The NAC International UMS Universal Storage System, Effective 3/31/04

Records
MNS Radiation Survey No. M-122204-5, ISFSI Pad Area, conducted 12/22/2004
Procedures Process Record HP/2/B/1006/027, Radiation Protection Controls For Loading Spent Fuel Assemblies Into NAC-UMS Dry Storage Casks Enclosures 5.1, 5.2, 5.3, 5.4, 5.5, and 5.6, conducted 12/13/04
Procedures Process Record HP/0/B/1003/063, Enclosure 5.17, Routine Surveillance, conducted 01/11/04, 05/25/04, 08/24/04, and 12/13/04
First NAC-UMS Cask Load (McGuire Nuclear Station) Dose Summary (December 2004)

LIST OF ACRONYMS

1EOC	-	Unit 1 End-of-Cycle
2EOC	-	Unit 2 End-of-Cycle
AB	-	Auxiliary Building
ALARA	-	As Low As Reasonably Achievable
ANSI	-	American National Standards Institute
AOT	-	Allowable Outage Time
AP	-	Abnormal Procedure
ARM	-	Area Radiation Monitor
ASME	-	American Society of Mechanical Engineers
BACC	-	Boric Acid Corrosion Control
CA	-	Auxiliary Feedwater
CAP	-	Corrective Action Program
CAST	-	Auxiliary Feedwater Storage Tanks
cfm	-	Cubic Feet Per Minute
CLA	-	Cold Leg Accumulator
CFR	-	Code of Federal Regulations
COT	-	Channel Operational Test
CP	-	Chemistry Procedure
CY	-	Calendar Year
DPC	-	Duke Power Company
DRP	-	Discrete Radioactive Particle
ECCS	-	Emergency Core Cooling System
ED	-	Electronic Dosimeter
EDG	-	Emergency Diesel Generator
EHRA	-	Extra-High Radiation Area
EMF	-	Effluent Monitoring
EnRad	-	Environmental Radiation
EP	-	Emergency Procedure
EPRI	-	Electric Power Research Institute
ET	-	Eddy Current Testing
fpm	-	Feet Per Minute
F&ES	-	Floor & Equipment Sump
FSAR	-	Final Safety Analysis Report
GL	-	Generic Letter
GWR	-	Gaseous Waste Release
HEPA	-	High Efficiency Particulate Air
HHSI	-	High Head Safety Injection

HP	-	Health Physics Procedure
HPGe	-	High Purity Germanium
HPT	-	Health Physics Technician
HRA	-	High Radiation Area
HS	-	Moisture Separator Reheater Drain System
IR	-	Inspection Report
ISFSI	-	Independent Spent Fuel Storage Installation
ISI	-	In Service Inspection
KC	-	Component Cooling Water
LCO	-	Limiting Condition of Operation
LER	-	Licensee Event Report
LLD	-	Lower Limit of Detection
LWR	-	Liquid Waste Release
MDCA	-	Motor Driven Auxiliary Feedwater
MGTM	-	Temporary Modifications
MNS	-	McGuire Nuclear Station
MSIV	-	Main Stem Isolation Valve
MSR	-	Moisture Separator Reheater
NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NDE	-	Non-Destructive Examination
NEI	-	Nuclear Energy Institute
NI	-	Safety Injection
No.	-	Number
NSD	-	Nuclear System Directive
NV	-	Chemical and Volume Control
OA	-	Other Activities
ODCM	-	Offsite Dose Calculation Manual
OOS	-	Out-of-Service
OS	-	Occupational Radiation Safety
PASS	-	Post-Accident Sampling System
PCM	-	Personnel Contamination Monitor
PI	-	Performance Indicator
PIP	-	Problem Investigation Process report
PM	-	Portal Monitor
PMT	-	Post-Maintenance Testing
PRT	-	Pressurized Relief Tank
PS	-	Public Radiation Safety
PT	-	Surveillance Test
QA	-	quality assurance
QC	-	Quality Control
RCA	-	Radiologically Controlled Area
RCCA	-	Rod Cluster Control Assembly
RCS	-	Reactor Coolant System
RCZ	-	Radiation Control Zone
REMP	-	Radiological Environmental Monitoring Program
Rev	-	Revision
RF	-	Fire System
RN	-	Nuclear Service Water

RP	-	radiation protection
RTP	-	Rated Thermal Power
RWP	-	radiation work permit
SAM	-	Small Article Monitor
SCBA	-	Self-contained Breathing Apparatus
scfm	-	Standard Cubic Feet per Minute
SDP	-	Significance Determination Process
SFP	-	Spent Fuel Pool
SG	-	Steam Generator
SH	-	Shared Health Physics Procedure
SLC	-	Selected Licensee Commitment
SM	-	Main Stem
SR	-	Surveillance Request
SSC	-	Structures, Systems, and Components
SSF	-	Standby Shutdown Facility
TH	-	Temporary Health Physics Procedure
TLD	-	Thermoluminescent Dosimeter
TR	-	Testing Requirement
TS	-	Technical Specification
U1	-	Unit 1
U2	-	Unit 2
UDO	-	Unusual Dosimetry Occurrence
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
VA	-	Auxiliary Building Ventilation System
VB	-	Breathing Air System
VT	-	Visual Testing
WBC	-	Whole-Body Counter
WMT	-	Waste Monitor Tank
WO	-	Work Order
WR	-	Work Request