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

March 11, 2004

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 SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION REPORT NOS. 05000369/2004002 AND 05000370/2004002

Dear Mr. Peterson:

On February 13, 2004, the Nuclear Regulatory Commission (NRC) completed a safety system design and performance capability team inspection at your McGuire Nuclear Station. The enclosed report documents the inspection findings which were discussed on February 12, 2004, with Mr. T. Harrall and other members of your staff. Following completion of additional review in the Region II office, a final exit was held by phone with Mr. T. Harrall and other members of your staff on March 3, 2004.

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 inspection team reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents one NRC-identified finding of very low safety significance (Green). The finding was determined to be a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you deny this non-cited violation, you should provide a response 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, DC 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 the

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/

Charles R. Ogle, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: NRC Inspection Report 05000369/2004002 and 05000370/2004002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-369, 50-370
License Nos.:	NPF-9, NPF-17
Report Nos.:	05000369/2004002 and 05000370/2004002
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station, Units 1 & 2
Location:	12700 Hagers Ferry Road Huntersville, NC 28078
Dates:	January 26 - 30, 2004 February 9 - 13, 2004
Inspectors:	N. Merriweather, Senior Reactor Inspector (Team Lead) J. Leivo, Instrumentation and Control Systems Contractor J. Fuller, Reactor Safety Intern R. Moore, Senior Reactor Inspector R. Rodriguez, Reactor Safety Intern M. Thomas, Senior Reactor Inspector
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000369/2004-002, 05000370/2004-002; 01/26-30/2004 and 02/09-13/2004; McGuire Nuclear Station, Units 1 and 2; Safety System Design and Performance Capability Inspection.

This inspection was conducted by a team of inspectors from the NRC Region II office with assistance from a contractor specializing in instrumentation and controls. The team identified one Green non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control requirements. The licensee had failed to identify and evaluate the impact on design of sloping the impulse lines for the containment pressure transmitters downward from the containment towards the transmitters without low point drain legs installed. This configuration was a deviation from the licensee's design requirements, and introduced the potential for water intrusion in the instrument impulse lines during normal operation and accident conditions. In response to this condition, the licensee performed an operability evaluation and entered the finding into their corrective program (Problem Investigation Process (PIP) Report No. M-04-00713).

The finding is greater than minor because it affects the design control attribute of the mitigating systems cornerstone objective, in that the formation of a loop seal would have the potential to affect the performance capability of instruments used for automatic initiation of engineered safety features, containment pressure control, and post-accident monitoring. The finding was determined to be of very low safety significance (Green) because it is a design deficiency that will not result in loss of automatic initiation of engineered safety features, containment pressure control, or post-accident monitoring capability (loss of function). (Section 1R21.21. b).

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

1R21 Safety System Design and Performance Capability (71111.21)

Except as noted below, the inspection team focused on Unit 2 systems and components and operator actions required to prevent or mitigate the consequences of a steam generator tube rupture (SGTR) event. The inspection team also examined the licensee's steam generator (SG) surveillance program, secondary water chemistry controls, foreign material exclusion controls, loose parts monitoring, and steam generator leakage monitoring controls and procedures. Additionally, the team reviewed the licensee's applicability evaluations and corrective actions for industry operating experience issues related to SGTR events.

.1 System Needs

.11 Process Medium

a. Inspection Scope

The team reviewed the availability, reliability, and adequacy of water sources required for the mitigation of an SGTR event. These included the refueling water storage tank (RWST), volume control tank (VCT), condensate storage tank (CACST), and the assured service water (RN) suction source. Design criteria information, drawings, vendor manuals, and calculations to determine the pump net positive suction head (NPSH) and tank volume were reviewed to verify that design and Updated Final Safety Analysis (UFSAR) accident analysis assumptions were consistent with system and equipment capabilities. The team evaluated the available primary and secondary water source volumes with respect to the anticipated water source requirements for the SGTR event.

b. Findings

No findings of significance were identified.

.12 Energy Sources

a. Inspection Scope

The team reviewed appropriate test and design documents to verify that the 4160 volt and 600 volt alternating current (VAC) power sources, as well as the 125 volt direct current (VDC) power sources, were adequate to meet minimum voltage specifications for electrical equipment during and following an SGTR event. The Unit 2 safety related 4160 VAC pump motors reviewed for voltage adequacy were safety injection (NI), chemical and volume control (NV), residual heat removal (ND), and RN. Specific valves reviewed for adequacy of voltage, included the Unit 2 steam supply air operated valves (AOVs) to turbine driven auxiliary feedwater pump (TDCAP) and SG auxiliary feedwater

(CA) isolation motor operated valves (MOVs), as well as, the standby nuclear service water pond (SNSWP) supply and discharge “B” isolation MOVs and ND inlet isolation MOVs.

The team reviewed the 125 VDC battery sizing calculation, and voltage drop study to verify that the 125 VDC electrical distribution system was capable of providing reliable power during an SGTR event.

In addition to the above, the team reviewed the quality controls for the air systems required for operation of Unit 2 AOVs (i.e., the TDCAP steam supply valves 2SA-48ABC, -49ABC; and SG PORVs 2SV-1AB, -7ABC, -19AB). Specifically, valve design drawings, vendor manuals, and periodic air quality test results were reviewed to verify that plant air quality standards were consistent with vendor recommendations, regulatory guidance, and industry standards. Alternate air and gas sources for operation of AOVs were also assessed, including the capacity and availability of the diesel air compressors.

b. Findings

No findings of significance were identified.

.13 Instrumentation and Controls

a. Inspection Scope

The team reviewed the Unit 2 instrumentation that is used by operators for detection of primary to secondary leakage and an SGTR event, as well as selected control circuits used for SGTR mitigation. Instrumentation used for detection included the main steam (SM) line N-16 radiation monitors, SM line radiation monitors (area monitors strapped to the four main steam lines), condenser air ejector radiation monitor, SG blowdown sample isolation radiation monitor, and SG narrow range level instruments. For these instruments, the team reviewed the SGTR accident analysis, design basis documents, calculations, design documents, and vendor documents establishing the basis for calibration and alarm setpoints, to confirm that the calibration, setpoints, and emergency operating procedure (EP) action levels were consistent with the design and licensing basis.

For controls used in SGTR mitigation, the team reviewed the instrument detail drawings and elementary diagrams for the control circuits for the steam generator power operated relief valves (PORVs), the pressurizer PORVs, and the automatic isolation of the steam generator blowdown sample line. The objective of the drawing review was to confirm that the circuits implemented the functional requirements levied by the design and licensing basis.

The specific documents reviewed are included in the Attachment to this report.

b. Findings

No findings of significance were identified.

.14 Operator Actions

a. Inspection Scope

The team reviewed plant operating procedures (OPs), including EPs, abnormal operating procedures (APs), and annunciator response procedures that would be used in the identification and mitigation of an SGTR event. Specific procedures reviewed included:

- AP/2/A/5500/010, NC System Leakage Within Capacity of Both NV Pumps, Rev. 14
- EP/1/A/5000/E-3, Steam Generator Tube Rupture, Rev. 12
- EP/2/A/5000/E-0, Reactor Trip or Safety Injection, Rev. 18
- EP/2/A/5000/E-3, Steam Generator Tube Rupture, Rev. 10
- OP/2/A/6100/010Q, Annunciator Response for Panel 2RAD-1, Rev. 34
- OP/2/A/6100/010R, Annunciator Response for Panel 2RAD-2, Rev. 27
- OP/2/A/6100/010S, Annunciator Response for Panel 2RAD-3, Rev. 11

The review was done to verify that the procedures were consistent with the UFSAR description of an SGTR event and with the Westinghouse Owners Group Emergency Response Guidelines, any step deviations were justified and reasonable, and the procedures were written clearly and unambiguously. The team conducted discussions with licensed operators and reviewed job performance measures and training lesson plans pertaining to an SGTR event to ensure that training was consistent with the procedures.

In addition, the team observed a simulation of an SGTR event on the plant simulator and walked down portions of applicable procedures to verify that operator training, procedural guidance, and instrumentation were adequate to identify an SGTR event and implement post-event mitigation strategies. The local manual operator action times for performance of SGTR event mitigation activities were reviewed for consistency with accident analyses and operator training.

b. Findings

No findings of significance were identified.

.15 Heat Removal

a. Inspection Scope

The team reviewed the reliability and availability of cooling for equipment required to mitigate the SGTR event. This included cooling water to the CA, NI, and NV Pumps.

Vendor manuals, design documentation, drawings, and surveillance and test procedures, were reviewed to verify the vendor recommendations for equipment operation were satisfied.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

.21 Installed Configuration

a. Inspection Scope

The team performed field walk downs of accessible SGTR mitigation equipment in the CA, NV, SM, NI, instrument air (VI), and steam generator blowdown systems to assess general material condition, identify degraded conditions, and verify the installed configuration was consistent with design drawings and design inputs to calculations. Additionally, the team assessed potential flooding and missile impact on SGTR mitigation equipment. The team performed a walkdown of the discharge side of the 2A motor driven CA pump to ensure that pipe supports and snubbers were installed per design drawings.

The team walked down portions of the 125V DC and 4160V AC electrical distribution systems to verify that the installed configurations were consistent with design basis information. The team visually inspected the 4160V AC vital switchgears and panels, and the 125V DC batteries and battery chargers and distribution panels to evaluate material condition.

The team walked down the Unit 2 main steam line N-16 radiation monitors, Unit 1 & 2 main steam line radiation monitors, Unit 1 & 2 condenser air ejector radiation monitors, and Unit 2 steam generator blowdown sample isolation radiation monitor and the associated control room indicators and setpoints, to confirm that the instrument configurations were consistent with the design and licensing basis. The team also confirmed that the taps for the RWST level instruments were located so as to preclude adverse velocity effects on the measurement.

The team reviewed the installation detail drawings for containment pressure and containment sump level field instrumentation. These instruments are relied upon, in part, to determine whether the event is a loss of coolant accident (LOCA) or an SGTR event. The objective of the review was to confirm that design basis failure modes and effects on the installed configurations had been adequately considered with respect to normal and accident conditions. For both units, the team visually inspected all of the containment pressure transmitters and impulse line tubing from the reactor building/containment annulus penetrations to the transmitters, to confirm that the impulse lines were adequately separated, and were sloped or drained to preclude loop seals or blockage of the lines.

b. Findings

Introduction: The team identified a Green, non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control requirements. Prior to the inspection, the licensee had not identified and evaluated the potential impact of sloping of the impulse lines of the containment pressure transmitters downward from the containment towards the transmitters without providing drain legs. This configuration was a deviation from the licensee's design requirements, and introduced the potential for water intrusion during normal operation and accident conditions.

Description: During the inspection, prior to an announced NRC walkdown, the licensee identified that the impulse line tubing for the four redundant channels of the Unit 1 & 2 containment pressure transmitters was sloped downward, rather than upward, from the containment to transmitters. For Unit 2, no drain legs were provided, as required by the licensee's design drawings when the upward slope requirements could not be satisfied. The licensee had not previously analyzed the potential effects of this deviating configuration on the performance of the instruments.

Licensee Drawing No. MCID-2499-NS.01, Revision 1, "Instrument Detail - Reactor Building Containment Pressure," contains detailed design requirements for all of the 14 safety-related containment pressure transmitters. Note 11 of the drawing stipulates:

> "Tubing should be sloped upward from tap to transmitters. If this is not possible, a drain leg should be provided by lengthening the test tee and plug at the transmitter."

During the inspection, the team requested the licensee show how this design requirement had been satisfied, and to identify what measures the licensee used to ensure that the impulse lines would not be blocked or sealed. In response, the licensee determined from walkdowns during the inspection that the slope requirements had not been satisfied, and that for Unit 2, the required drain legs had not been installed. The licensee also determined that a low point sag existed for one channel of instrument tubing within the containment annulus of Unit 1.

The licensee entered this issue into their corrective action program as Problem Investigation Process report (PIP) M-04-00713, dated 2/12/2004, and performed an operability evaluation. The evaluation analyzed the potential for water intrusion during normal and accident operation. The evaluation was supported by detailed walkdowns of all containment pressure instrument lines and by two calculations.

Calculation MCC-1223.13-00-0021, "Containment Pressure Instrument Sensing Line Condensation during a Loss of Coolant Accident," Revision 1, dated 2/17/2004, determined that the total moisture in the tubing from the initial air and steam introduced into the line following a large break LOCA would be 1.9 inches of water per 100 feet of impulse line tubing. From the results of this calculation and detailed walkdowns, the licensee prepared and issued calculation MCC-1381.17-00-0120, "PIP-M-04-0713 NS Containment Pressure Instrumentation Line Operability Evaluation," Revision 1, dated

2/23/2004, to determine the approximate column of water expected in the impulse lines during a LOCA, and determined, based on the as-found impulse line geometry, that the volumes of water introduced into the containment pressure sensing lines during a LOCA would be insufficient to cause blockage in any of the associated channels. This result supported the licensee's conclusion that the Unit 1 & 2 containment pressure transmitters and associated sensing lines were operable, and were degraded/non-conforming.

To assess the licensee's operability evaluation, the team performed a walkdown of all containment pressure transmitters and associated impulse lines outside of the reactor building wall, and subsequently performed an in-office review of calculations MCC-1223.13-00-0021 and MCC-1381.17-00-0120 cited above. Based on this review, the team agreed with the licensee's conclusion that the Unit 1 & 2 containment pressure transmitters and associated sensing lines were not vulnerable to the effects of a loop seal, and therefore were operable and degraded/non-conforming.

Analysis: The finding is greater than minor because the degraded/non-conforming condition affected the design control attribute of the mitigating systems cornerstone objective. Specifically, for engineered safety features actuation and containment pressure control and integrity, the containment pressure instrumentation and associated tubing that supports these cornerstone objectives was in a configuration that deviated from the design requirements.

The finding was determined to be of very low safety significance (Green) because it is a design deficiency that will not result in loss of automatic initiation of engineered safety features, containment pressure control, or post-accident monitoring capability. This determination was based on review of the licensee's operability evaluation, NRC Generic Letter 91-18 (Rev. 1) guidance on degraded and nonconforming conditions, and the Significance Determination Process (SDP) Phase 1 Screening Worksheet question number 1 under Mitigating Systems.

Enforcement: 10 CFR 50, Appendix B, Criterion III, Design Control, requires in part, that design control measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Licensee Drawing No. MCID-2499-NS.01, Revision 1, "Instrument Detail - Reactor Building Containment Pressure," contains detailed design requirements for all of the 14 safety-related containment pressure transmitters. Note 11 of the drawing stipulates:

> "Tubing should be sloped upward from tap to transmitters. If this is not possible, a drain leg should be provided by lengthening the test tee and plug at the transmitter."

Contrary to the above requirements, the NRC identified on February 12, 2004, that since initial startup of the units, the design requirements for slope and drain legs for the safety-related containment pressure impulse lines had not been fully satisfied, and the licensee had not evaluated the impact of this deviation on the performance of the containment pressure channels. Because this violation was of very low safety significance and the licensee entered the finding into their corrective action program as PIP No. M-04-00713, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. This finding is identified as NCV 05000369,370/2004002-01, Deviation from Design Requirements for Line Slope and Drain Legs for Containment Pressure Transmitter Impulse Lines Was not Identified or Evaluated.

.22 Operation

a. Inspection Scope

The team walked down accessible portions of CA and SM systems (which included the steam supply valves 2SA-48ABC and 2SA-49ABC to the TDCAP; and SM valves 2SV-1AB, 2SV-7ABC, 2SV-13AB, 2SV-19AB, 2SV-25, 2SV-26, 2SV-27, 2SV-28) to verify that the system alignments were consistent with design and licensing basis assumptions, and to verify that the systems would be available for operators to mitigate an SGTR event. During those walkdowns, the team compared valve positions with those specified in the CA and SM systems' piping and instrumentation drawings and operating procedure lineups; and observed the equipment material condition to determine if it would be adequate to support operator actions to mitigate an SGTR event.

The team walked through, with an operator, Unit 1 and Unit 2 EP actions to locally operate the SM power operated relief valves (PORV) and PORV isolation valves and isolation of the steam supply from the affected SG to the TDCAP. The team reviewed the EP actions to verify that human factors in the procedures and in the plant (e.g., clarity, lighting, accessibility, labeling) were adequate to support effective use of the procedures.

b. Findings

No findings of significance were identified.

.23 Design

a. Inspection Scope

Mechanical Design Review

The team reviewed design calculations, specifications, and UFSAR accident analysis to identify the design criteria which defined the required capacity and capability of SGTR mitigation equipment. Surveillance test procedures and equipment monitoring activities were reviewed to verify the design criteria was appropriately translated into acceptance criteria. The team reviewed NPSH calculations for the NI and centrifugal charging pumps from the RWST to verify that adequate NPSH was available. Design changes were reviewed to verify that system and equipment design functions were appropriately evaluated and maintained. Design changes reviewed included replacement of actuators on TDCAP steam supply valves, as well as replacement of actuator springs on main steam isolation valves and steam generator blowdown isolation valves. The team also reviewed a design changes associated with replacement of a failed mechanical snubber and hydraulic snubber.

Electrical, Instrumentation and Controls Design Review

The team reviewed the battery sizing calculation for the Unit 1 Class 1E 125 VDC electrical distribution system to assess the adequacy of the batteries to provide power for selected components required to mitigate a SGTR event. Additionally, the team examined the voltage adequacy calculation for the Unit 2 steam supply air operated valves (AOVs) to TDCAP and SG CA isolation MOVs, as well as, the ND inlet isolation MOVs.

The team reviewed a sample of temporary and permanent plant design changes impacting instrumentation and control systems to confirm that the design changes were in accordance with the design and licensing basis. Design changes reviewed included replacement of the Unit 2 condenser air ejector radiation monitor, revision of the SG level operator aide computer setpoint, as well as removal of the CA speed controls on both units. A list of the modifications reviewed are included in the Attachment to this report.

b. Findings

No findings of significance were identified.

.24 Testing and Inspection

a. Inspection Scope

The team reviewed valve stroke time testing of MOVs 1SM-5AB, 2SM-1AB, 2SV-1AB, and 2SV-7ABC, as well as, MOV torque and limit switch settings and post-maintenance testing to verify that the tests and inspections were appropriately verifying that the assumptions of the licensing and design bases were being maintained and that performance degradation would be identified.

The team evaluated the 125 VDC batteries (i.e., EVCA, EVCB, EVCC, and EVCD) surveillance test records, including preventive maintenance and performance tests results, to verify that the batteries were capable of meeting design basis load requirements. The team reviewed calibration records for the overcurrent and undervoltage protective relays on safety buses 2ETA and 2ETB to verify that the relays were set in accordance with calibration procedures and setpoint documents.

The team reviewed surveillance test procedures and completed test results related to reactor coolant system leakage, primary to secondary leakage, and operator time critical task verification to verify that testing was being performed in accordance with applicable

Technical Specification requirements, UFSAR commitments, and the McGuire Nuclear Station Selected Licensee Commitments Manual (SLC).

The team reviewed a sample of the records of the last two calibrations and channel operational tests for selected SGTR related instruments, as well as containment sump level instruments, to confirm that acceptance criteria were satisfied or that appropriate corrective actions had been taken. The team observed the operation of these instruments during an SGTR simulator exercise to qualitatively confirm simulator fidelity.

b. Findings

No findings of significance were identified.

.3 Selected Components

.31 Component Degradation

a. Inspection Scope

The team reviewed maintenance and testing documentation, performance trending, and equipment history as identified by In-Service Test program trending, work orders, and PIP reports, to assess the licensee’s actions to verify and maintain the safety function, reliability and availability of selected components. Additionally reviewed were potential common cause failure mechanisms due to flooding, maintenance, parts replacement and modifications. The Maintenance Rule Functional Failures for the selected components for the past 5 years were reviewed. The selected components included MOVs, air operated valves, relief valves, check valves, solenoid valves and pumps. A specific list of components reviewed is included in the Attachment to this report.

The team reviewed the licensee’s secondary water chemistry control program, to ensure that the critical secondary water chemistry parameters, as stated in Chapter 10.4 of the UFSAR, are monitored and controlled to inhibit secondary side steam generator corrosion and associated tube degradation. The team assessed the licensee’s program for adherence to appropriate vendor recommendations and industry guidance in this area.

The team reviewed preventive maintenance records for 125 VDC batteries and chargers, as well as CA pump motors, to assess the licensee’s actions to verify and maintain the safety function, reliability, and availability of the components in the system.

The team visually inspected the Unit 2 main steam line N-16 radiation monitors, Unit 1 & 2 main steam line radiation monitors, Unit 1 & 2 condenser air ejector radiation monitors, Unit 2 steam generator blowdown sample isolation radiation monitor, and all of the containment pressure transmitters and impulse line tubing from the reactor building/containment annulus penetrations to the transmitters, to confirm that the visible material condition of the impulse lines, instruments, supports, and connections was adequate.

b. Findings

No findings of significance were identified.

.32 Equipment/Environmental Qualification

a. Inspection Scope

The team reviewed preventive maintenance records for selected Class 1E electrical equipment to verify that environmental qualification requirements were being implemented during mentioned activities. Specifically, while reviewing calibration procedures for steam generator level transmitters included in the licensee's environmental qualification program, the team confirmed that appropriate requirements were included for replacement of O-ring seals as required to maintain qualification.

b. Findings

No findings of significance were identified.

.33 Equipment Protection

a. Inspection Scope

For both units, the team visually inspected the SM line N-16 radiation monitors, SM line radiation monitors, condenser air ejector radiation monitor, and steam generator blowdown sample isolation radiation monitors to confirm that the instruments and connections were not vulnerable to the effects of design basis events for which they were credited to be functional, including the effects of extreme ambient temperatures and background dose rates.

In addition to the above, the team reviewed the equipment specifications for the SG PORVs, main steam safety valves, TDCAPs, and NV pumps to verify the design was adequate for anticipated ambient conditions and system application.

b. Findings

No findings of significance were identified.

.34 Component Inputs/Outputs

a. Inspection Scope

The team reviewed selected CA and RN MOV operator requirements calculations and evaluated the capability of the MOVs to perform their design function under degraded voltage conditions. The specific MOVs reviewed are discussed in Section 1R21.12.a.

b. Findings

No findings of significance were identified.

.35 Operating Experience

a. Inspection Scope

The team reviewed the licensee's applicability evaluations and corrective actions for industry experience issues related to TDCAPs, VI system failures, CA re-circulation flow orifice fouling, fouling of Catawba's component cooling water heat exchangers, ND heat exchanger partition plate design, failures of safety-related valves due to linestarter relay degradation, as well as transformer reliability issues and other SGTR events to verify that applicable insights from those reports had been applied to plant procedures and operator training. The specific documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.36 Steam Generator Inspection

a. Inspection Scope

The team performed a limited-scope review of the SG surveillance program in order to ensure that SG tubes were being inspected as required by Technical Specifications and plant procedures.

b. Findings

No findings of significance were identified.

.37 Foreign Material Exclusion (FME) Control Program And Loose Parts Monitoring

a. Inspection Scope

The team reviewed procedures and performance records for the Loose Parts Monitoring System (LPMS) to verify that the system was operational and was being used to monitor for loose parts in the reactor coolant system and steam generators. The team assessed the licensee's program for adherence to appropriate industry recommendations, and selective licensing commitments. The team reviewed completed test procedures for channel operability, channel calibration, and daily operations surveillance tests. In addition, the team held discussions with appropriate engineering personnel, and reviewed procedures and PIPs regarding FME program controls, cleanliness requirements, and material accountability.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The team screened and reviewed a sample of Unit 2 PIPs initiated over the past three years, to confirm that the licensee was adequately identifying, evaluating, and dispositioning adverse conditions regarding the main steam line N-16 monitors, main steam line monitors, condenser air ejector monitor, steam generator blowdown sample isolation monitor, steam generator narrow range level instruments, control circuits for the SG PORVs, and control circuits for the pressurizer PORVs. The team reviewed selected system health reports, maintenance records, surveillance test records, calibration test records, and PIPs to verify that design problems were identified and entered into the corrective action program. The team also reviewed PIPs related to other selected SGTR mitigation equipment to assess the scope of the licensee's extent-of-condition reviews and the adequacy of the corrective actions. The specific documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

The lead inspector presented the inspection results on February 12, 2004, and in a subsequent conference call on March 3, 2004, to Mr. T. Harrall, and other members of the licensee staff. The licensee acknowledged the findings presented. Proprietary information is not included in this inspection report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

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S. Bradshaw, Superintendent, Plant Operations
J. Bramblett, Chemistry Manager
S. Brown, Manager, Engineering
M. Costella, Radiological Engineering
K. Crane, Regulatory Compliance
S. Hackney, Operations Procedures Group
T. Harrall, Station Manager
T. Henrickson, Secondary Chemistry
D. Homce, Reactor Engineering
R. Johansen, System Engineer
J. Kammer, Safety Assurance Manager
P. Keller, Radiation Monitoring System Engineer
J. Nolin, Operations Support
T. Pederson, System Engineer
R. Pope, Operations Training
S. Schultz, Manager, Radiological Engineering
F. Setzer, Air Operated Valve Engineer
J. Thomas, Manager, Regulatory Compliance
K. Thomas, Manager, RES Engineering
B. Travis, Superintendent, Work Control
M. Vanhoy, Instrumentation and Controls Engineer
M. Weiner, Operations Procedures Group
T. Welch, Valves/Heat Exchangers Engineering Supervisor
M. Wilder, Regulatory Compliance

NRC (attended exit meeting)

J. Brady, Senior Resident Inspector
C. Ogle, Chief, Engineering Branch 1, NRC Region II
S. Walker, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000369,370/2004002-01	NCV	Deviation from Design Requirements for Line Slope and Drain Legs for Containment Pressure Transmitter Impulse Lines Was not Identified or Evaluated (Section 1R21.21. b)

LIST OF DOCUMENTS REVIEWED

Procedures

AP/2/A/5500/010, NC System Leakage Within Capacity of Both NV Pumps, Rev. 14
EP/1/A/5000/E-3, Steam Generator Tube Rupture, Rev. 12.
EP/2/A/5000/E-0, Reactor Trip or Safety Injection, Rev. 18
EP/2/A/5000/E-3, Steam Generator Tube Rupture, Rev. 10
HP/0/B/1003/008, Determination of Radiation Monitor Setpoints (EMFs), Enclosure 5.1, Rev. 33
IP/0/A/3050/004D, Containment Pressure Protection IV Calibration, Rev. 7.
IP/0/A/3190/003B, Starter And Contactor Preventive Maintenance, Rev. 5
PT/2/A/4350/026C, Auxiliary Shutdown Panel Controls Verification for #2 TDCAP Controls, Rev. 7
PT/2/A/4600/003A, Enclosure 13.1, Semi-Daily Surveillance Items Checklist [containment pressure instruments].
OP/2/A/6100/010Q, Annunciator Response for Panel 2RAD-1, Rev. 34
OP/2/A/6100/010R, Annunciator Response for Panel 2RAD-2, Rev. 27
OP/2/A/6100/010S, Annunciator Response for Panel 2RAD-3, Rev. 11
PT/1/B/4700/087, Unit 1 Secondary Chemistry Periodic Surveillance for Primary to Secondary Leakage (SGs), Rev. 0
PT/2/A/4150/001C, Primary to Secondary Leakage Monitoring, Rev. 2
PT/2/B/4700/087, Unit 2 Secondary Chemistry Periodic Surveillance for Primary to Secondary Leakage (SGs), Rev. 0
CP/0/B/8100/045, Primary to Secondary Leakrate Calculation and Response, Rev. 24
NSD-513, Primary-to-Secondary Leak Monitoring Program, Rev. 2
NSD-514, Control of Time Critical Tasks, Rev. 0
OMP 4-3, Use of Abnormal and Emergency Procedures, Rev. 22
PT/0/B/4453/004, VI Air Quality Test - Dew point Temperature, Rev. 3
PT/1/B/4453/005, VI Air Quality Test - Oil and Particulate, Rev. 4
PT/2/A/4200/043B, Flushing of Unit 2 RN Makeup Line to CA Pumps (B train), Rev. 4
PT/2/A/4251/002C, BB Valve Stroke Timing - Shutdown, Rev. 10
PT/2A/4600/030, Cycling Time Critical Manually Operated Valves, Rev. 6
MP/0/A/7600/167, Measuring AOV Thrust Loads Using Valve Vision, Rev. 6
PT/0/A/4600/113, Operator Time Critical Task Verification, Rev. 5
NSD 514, Nuclear System Directive, Control of Time Critical Tasks, Rev. 0
MP/0/A/7600/100, Pacific Stop Check Valve Corrective Maintenance, Rev. 8
NSD 104, Material Condition / Housekeeping, Cleanliness / Foreign Material Exclusion and Seismic Concerns, 09/13/02, Rev. 22
OP/2/B/6150/016, Loose Parts Monitoring System, Rev. 12
OP/2/A/6100/010 N, Annunciator Response for Panel 2AD-13, Loose Part Panel Trouble
PT/2/B/4600/098, Loose Parts Monitor (LPMS) Functional Test, Rev. 6
IP/0/B/3050/012, Loose Parts Monitor Calibration, Rev. 14
IP/0/B/3050/012 A, Loose Parts Monitoring System Field Maintenance Procedure, Rev.1
NSD 204, Operating Experience Program (OEP) Description, Rev. 8

Drawings

MCCD-1700-00.00,Unit 1 Configuration One Line Diagram Unit Essential Power System, Rev.4
MCCD-1705-01.00, 125 VDC / 120 VAC Vital Instrument & Control Power System, Rev.79
MCCD-1702-02.00, 4160 V Essential Auxiliary Power System, Rev.0

MCM 1205.00-1252 001, 4 CL 1500 Parallel Disc Gate Valve (Atwood & Morril Co.) W/Rotork 30 NAI Operator, Rev. 6
MCFD 1593-01.02, Flow Diagram (FD), SM Supply to Auxiliary Equipment System (SA) Turbine Exhaust (TE), Rev. 4
MCM 1205.09-0027 001, Actuator Assembly, 200 inch 6 Stroke, Normally Retracted Control Components, Rev. 0
MCM 1205.01-0618 001, 4 inch-900 Automatic Re-circulation Valve with 2 inch Bypass Valve, Rev. 9
MCM 1201.04–0137 001, Refueling Water Tank, Rev. D8
MCM 1205.09–0010 001, Valve Installation, SG Relief Power Operated, McGuire 1&2, Rev. D4
MCM 1205.09-0011 001, Body Assembly, Relief Valve Power Operated Steam Generator, McGuire 1&2, Rev. D10
MCM 1205.09-0002 001, Nozzle Type Safety Valve, Crosby, (MSSV), Rev. DH
MCM 1205.12-0001 001, 34 inch Main Steam Isolation Valve, Cylinder Assembly, Rev. D17
MCM 1201.04-100, Volume Control Tank - 400 FT, Rev. 0
2-CA-354, MCSRD-CA System Auxiliary Building, Sheet 1, Rev. 2
2-CA-354, MCSRD-CA System Auxiliary Building, Sheet 2, Rev. 4
2-CA-354, MCSRD-CA System Auxiliary Building, Sheet 3, Rev. 0
MC-1041-14, Reactor Building Unit 2 General Arrangement - Plan at Elevation 738+3, Rev. 26.
MCFD-1580-01.00, FD of SG Blowdown Recycle System, Rev. 20.
MCFD-2591-01.00, Unit 2 FD of Feedwater System, Rev. 2
MCFD-2591-01.01, Unit 2 FD of Feedwater System, Rev. 13
MCFD-2593-01.00, FD of SM System and SM Vent to Atmosphere, Rev. 15.
MCFD-2593-01.01, FD of SM System and SM Vent to Atmosphere, Rev. 12.
MCFD-2593-01.03, FD of SM System and SM Vent to Atmosphere, Rev. 15.
MC-2499-NI.48, Instrument Detail (ID) - Containment Sump Level, Rev. 0.
MC-2499-NI.48.02, ID - Containment Sump Level Transmitter Guard Bracket, Rev. 0.
MCID-1499-BB.22, ID - SG Flow to SG Blowoff Tank Control, Rev. 3.
MCID-1499-CF.10, ID - SG Feedwater Inlet Bypass Control, Rev. 1
MCID-1499-FW.02-01, ID - Refueling Water Storage Tank Level Transmitter, Rev. 6
MCID-2499-FW.02-01, ID - Refueling Water Storage Tank Level Transmitter, Rev. 6
MCID-2499-NC.10, ID - Pressurizer Power Relief Valve Control, Rev. 3
MCID-2499-NS.01, ID - Reactor Building Containment Pressure, Rev. 1.
MCID-2499-SM.09, ID - SM PORV Control, Rev. 8.
MCEE-145-00.03, Elementary Diagram (ED), SG 1A, 1B, 1C, 1D Feedwater Control Bypass Control Valves 1CF0104, 1CF0105, 1CF0106, 1CF 0107 (Train A), Rev. 1.
MCEE-145-00.06, ED, SG 1A, 1B, 1C, 1D Feedwater Control Bypass Control Valves 1CF0104, 1CF0105, 1CF0106, 1CF 0107 (Train B), Rev. A.
MCEE-171-00.01 , ED, SG 1A Blowdown Containment Isolation Valve 1BB1B, Rev. 12.
MCEE-171-00.02 , ED, SG 1A Blowdown Containment Isolation Valve 1BB5A, Rev. 16.
MCEE-171-00.16 , ED, SG Blowdown Blowoff Tank Sample Control Valves 1BB123, 1BB124, 1BB125, 1BB126, Rev. 9.
MCEE-250-00.02, ED, Pressurizer PORV 2NC32B, Rev. 13.
MCEE-250-00.02-01, ED, Pressurizer PORV 2NC32B, Rev. 9.
MCEE-261-00.26, ED, Nuclear Sampling System, 2NM267SG Sample Header Radiation Monitor Inlet Isolation, Rev. 5.

MCEE-270-05.09, ED, Main Steam Vent to Atmosphere System (SV) MS 2D, Atmospheric Dump PORV 2SV1, Rev. 3.
MCEE-270-05.09-01, ED, SM PORVs Safety Controls, Rev. 6.
MCEE-270-05.09-02, ED, SM PORVs Safety Controls, Rev. 5.
MCEE-251-00.74, 2NI430A-Accu.2A Vent to 2NC34A for Blackout, Rev.6
MCEE-251-00.74-01, 2NI430A-Accu.2A Vent to 2NC34A for Blackout, Rev.6
MCEE-251-00.73, 2NI431B-Accu.2B Vent to 2NC32B for Blackout, Rev.9
MCEE-251-00.73-01, 2NI431B-Accu.2B Vent to 2NC32B for Blackout, Rev.7
MCEE-138-00.07, SNSWP Supply B Shutoff Valve 0RN9B, Rev.6
MCEE-138-00.07-01, SNSWP Supply B Shutoff Valve 0RN9B, Rev.5
MCEE-138-00.07-02, SNSWP Supply B Shutoff Valve 0RN9B, Rev.6
MCEE-138-00.44, SNSWP Discharge B Isolation Valve 0RN152B, Rev.6
MCEE-138-00.44-01, SNSWP Discharge B Isolation Valve 0RN152B, Rev.5
MCEE-138-00.44-02, SNSWP Discharge B Isolation Valve 0RN152B, Rev.6

Calculations

MCC-1223.13-00-0021, Containment Pressure Instrument Sensing Line Condensation during a Loss of Coolant Accident, Rev.1
MCC-1381.17-00-0120, PIP-M-04-0713 NS Containment Pressure Instrumentation Line Operability Evaluation, Rev. 0
MCC-1381.17-00-0120, PIP-M-04-0713 NS Containment Pressure Instrumentation Line Operability Evaluation, Rev. 1
DPND-DPC-NE-3002-A, UFSAR Chapter 15 System Transient Analysis Methodology, Section 7.2, Steam Generator Tube Rupture, Rev. 3
DPC-1552.08-00-150, RSG FSAR Analyses - 15.6.3 - SG Tube Failure Sections Describing Operator Actions, MNS SGTR Simulation Sequence of Events, and Figure 15.6.3.28, Rev. 8
MCC-1205.06-00-0015, AOV Capability Evaluation for 1(2)NC-0032,-0034,-0036. Rev. 2
MCC-1205.19-00-003, Electric Motor Operator Sizing Guidelines per GL 89-10 for Gate Valves, Rev. 21
MCC-1223.21-00-0003, FWST Capacity, Rev. 0
MCC-1223.12-00-0010, Verification of Minimum Available NPSH for ECCS Pumps, Rev. 3
MCC-1223.43-02-0006, Design Inputs for NSM MG-1/2 2556/0, ½ SA-48 & 49 Actuators, Rev.0
MCC-1201.06-00-0007, Qualification of RHR HX Partition Plate Design, 3/20/1998
MCC-1381.05-00-0263, McGuire Unit 2 ETAP Auxiliary System Voltage and Transformer Tap Study, Rev. 1 (specifically Appendices B, D, U, V and W)
MCC-1381.05-00-0230, Voltage Drop on the 125 VDC Vital Instrumentation and Control Power System (EPL), Rev. 1
MCC-1381.05-00-0200,125 VDC Vital Instrumentation and Control Power System Battery and Battery Charger Calculation, Rev. 5
Calculation MCC-1503.13-00-0103, USQ Review of NSMs MG-12062/00 and MG-22062/00: Feedwater Isolation on Low Tavg, dated 2/12/1988
Calculation DPC-1552.08-00-0150; MCC-1552.08-00-0239; CNC-1552.08-00-0228, RSG FSAR Analyses - 15.6.3 - SG Tube Failure, Rev. 8
Calculation MCC-1552.08-00-0208, Emergency Procedure Setpoints, Rev. 17

Design Basis Documents

MCS-1591.CF-00-0001, Design Basis Specification for the CF System, Rev. 7
MCS-1593.SM-00-0001, Design Basis Specification for the SM, SV, and SB Systems, Rev. 11
MCS-1605.VI-00-0001, Design Basis Specificaiton for the VI System, Rev. 12
MCS-346.05-EMF-0001, Design Basis Specification for the EMF [Radiation Monitoring] System, Rev. 13

Updated Final Safety Analysis Report

UFSAR Sections 7.2.2.3.5, 10.4.7, 10.4.8, 15.6.3,
UFSAR Section 1, Table 1-6, Regulatory Guide 1.97 Rev. 2 Review
UFSAR Figure 7-1, Instrumentation and Control System Logic Diagram
UFSAR Section 11.4.2.1.3, Steam Generator Sample Monitor
UFSAR Section 11.4.2.2.2, Condenser Air Ejector Monitor
UFSAR Section 11.4.2.2.11, Steam Line Monitor
UFSAR Section 11.4.2.2.14, Main Steam Line N-16 Monitors
UFSAR Section 15.6.3, SG Tube Failure
McGuire Selected Licensee Commitments, Section 16.11.7, Radioactive Gaseous Effluent Monitoring Instrumentation

Lesson Plans/Job Performance Measures (JPM)

ASE-11, Active Simulator Exam, Rev. 18
OP-MC-JPM-STM-SG:173T, Locally Isolate #1 TDCAP Steam Supply from Ruptured S/G 1C, Rev. 01
OP-MC-JPM-STM-SM:189T, Locally Close 1SV-25 (1D S/G PORV Motor Operated Isolation Valve), Rev. 00
OP-MC-JPM-STM-SM:202T, Manually Open 1SV-19 (1A S/G PORV) During SGTR Event, Rev. 00

Technical Specifications

Section 16.11.7, Radioactive Gaseous Effluent Monitoring Instrumentation [condenser air ejector monitor]
Section 3.3.1, Reactor Trip System Instrumentation
Section 3.3.2, Engineered Safety Feature Actuation System
Section 3.4.13, RCS Operational Leakage
Section 3.7.3, Main Feedwater Isolation Valves, Main Feedwater Control Valves (MFCVs),
MFCV’s Bypass Valves and Main Feedwater to Auxiliary Feedwater Nozzle Bypass Valves
Section 5.4, Procedures

Modifications

NSM-22549/00, Replace 2EMF33 Radiation Monitor.
MM-13025, Revise Steam Generator Level OAC Points - Unit 2.
TM-0147, Isolate CA Turbine Driven Pump Pushbutton Nuisance Ground

TM-0158, ‘B’ wide range containment sump level bypassed to range of 0 to 17 feet
MG-22556, Replace the Actuators on 2SA-48ABC & 49AB, 7/21/03
MGMM-11644, Replace Actuator Springs on 2BB–1B, -2b, -3B, -4B, 5A, -6A, -7A, and 8A, 10/1/03
MGMM-140141, Replace Springs on valve 2SM-003, 3/27/03
MGMM-11970, Replace Degraded Mechanical Snubbers on Supports 2MCR-NC-4011, 2MCR-NI, 4122, and 2MCR-S-NI-155-04-W with New Lisega Hydraulic Drop-in Replacement Snubbers, 10/2/2000
MGMM-11060, Remove TDCAP Speed Controls from TDCA Panel and CR on Unit 2, 10/4/00
MGMM-11059, Remove TDCAP Speed Controls from TDCA Panel and CR on Unit 1, 4/23/01

Completed Work Orders (WOs) and Work Requests (WRs)

98409843-01, PT 2CFLP5510, S/G ‘A’ Narrow Range (NR) Level Channel #2, 02/27/02.
98531034-01, PT 2CFLP6000, S/G ‘A’ NR Level Channel #1, 09/15/03.
98531034-02, PT 2CFLP6000, S/G ‘A’ NR Level Channel #1, 09/2203.
98243832, PM 2SA-6, IMV Check Valve Inspection, 10/24/01
98410290, PM 2SA-6 Disassemble and Inspect, 01/7/03
98044142, PT 2SA-5, Disassemble and Inspect, 04/20/00
98629096 01, PT 0EPLBAEVCA 1Q Battery Inspection,12/15/03
98612723 01, PT 0EPLBAEVCA 1Q Battery Inspection, 10/21/03
98621437 01, PT 0EPLBAEVCB 1Q Battery Inspection, 12/9/03
98606845 01, PT 0EPLBAEVCB 1Q Battery Inspection, 09/14/03
98620144 01, PT 0EPLBAEVCC 1Q Battery Inspection, 11/24/03
98604612 01, PT 0EPLBAEVCC 1Q Battery Inspection, 09/3/03
98626969 01, PT 0EPLBAEVCD 1Q Battery Inspection, 12/30/03
98611599 01, PT 0EPLBAEVCD 1Q Battery Inspection, 10/9/03
98595901 01, PM 2EPCRL2ETA02 Protective Relaying, 7/1/02
98409416 01, PM 2EPCRL2ETA02 Protective Relaying, 10/15/01
98422558 01, PM 2EPCRL2ETB02 Protective Relaying, 11/26/01
98187102 01, PM 2EPCRL2ETB02 Protective Relaying, 2/3/01
98605852 01, PM 2EPCRL2ETA15 Protective Relaying, 1/12/04
98409417 01, PM 2EPCRL2ETA15 Protective Relaying, 12/19/01
98598511 01, PM 2EPCRL2ETB15 Protective Relaying, 12/4/03
98405163 01, PM 2EPCRL2ETB15 Protective Relaying, 11/6/01
98410649 01, PM 2EPCRL2ETA16 Protective Relaying, 2/27/02
98557023 01, PM 2EQBRL27AX Station Blackout Relay 227AX, 4/9/03
98454882 01, PM 2EQBRL27AX Station Blackout Relay 227AX, 5/8/02
98553555 01, PM 2EQBRL27BX Station Blackout Relay 227BX, 3/26/03
98449855 01, PM 2EQBRL27BX Station Blackout Relay 227BX, 4/24/02
98051018 01, PM 2EQBLP27A Blackout Relaying Train A, 4/2/99
97015243 01, PM 2EQBLP27A Blackout Relaying Train A, 4/29/97
98043068 01, PM 2EQBLP27B Blackout Relaying Train B, 10/13/98
97011059 01, PM 2EQBLP27B & 27DB Blackout Relaying Train B, 4/30/97
98531888 01, PT 2EQBLP27DA, 2ETA Degraded Voltage Relaying Train A, 9/23/03
98410666 01, PT 2EQBLP27DA, 2ETA Degraded Voltage Relaying Train A, 3/1/02
98531889 01, PT 2EQBLP27DB, 2ETB Degraded Voltage Relaying Train B, 9/10/03

98410667 01, PT 2EQBLP27DB, 2ETB Degraded Voltage Relaying Train B, 3/7/02

Completed Surveillance Procedures, Preventive Maintenance (PM), Calibration and Test Records

IP/0/A/3190/001, PM Inspection and Cleaning, 2EMF-34L per WO 98366233-01,
IP/0/B/3006/003C, RMS W&T 5510 Flow Switch Calibration, 2EMF-34L per WOs 98366233-02 & 98531281-02,
WOs 98266781-01 & 98483722-01, IP/0/B/3006/007A, RMS Liquid Monitor Transfer Calibration, 2EMF-34L per WOs 98266781-01 & 98483722-01,
IP/2/A/3002/003A, SG Level Protection Calibration, Channel I, Loop A, B, C, & D, last two records for: Loop 2A Channel 1 per WO #98531034, WO #98409847; Loop 2A Channel 2 per WO# 98531030, WO #98409843; Loop 2B Channel 1 per WO #98531033, WO #98409846; Loop 2B Channel 2 per WO #98531035, WO #98409848.
IP/2/B/3005/007A, RMS RP-86A Low Range Area Channel Calibration, last two records for steam line radiation monitor 2EMF-10, per WO #98382326-01, WO #98216968-01.
IP/2/B/3005/007A, RMS RP-86A Low Range Area Channel Calibration, last two records for steam line radiation monitor 2EMF-11, per WO #98395331-01, WO #98241946-01.
IP/2/B/3006/009, RMS RP-86A Process Monitor Analog Channel Operational Test, last record for (new) condenser air ejector monitor 2EMF-33, per WO #98623821-01.
IP/2/B/3006/009, RMS RP-86A Process Monitor Analog Channel Operational Test, last two records for 2EMF-34L, per WO #98366233-01, #98531281-01.
IP/2/B/3006/009A, RMS RP-86A Low Range Process Channel Maintenance, last record for (new) condenser air ejector monitor 2EMF-33, per WO #98585933-10.
IP/2/B/3006/009A, RMS RP-86A Low Range Process Channel Maintenance, last two records for 2EMF-34L, per WO #98266781-01, WO #98483722-01.
IP/2/B/3006/009B, RMS Nitrogen-16 Monitor Calibration, last two records for 2EMF-71, per WO #98536830-01, WO #98466352-01.
IP/2/B/3006/009C, RMS Nitrogen-16 Monitor Analog Channel Operational Test, last three records for 2EMF-71, per WO #98611635-01, WO #98579652-01, WO #98536830-01.
IP/2/B/3006/029, RMS RD26 Inline Detector Transfer Calibration, last record for (new) condenser air ejector monitor 2EMF-33 per WO #98585933-10.
IP/0/A/3009/004, Containment Level Loop Calibration, Loop 2, last two records per WO #98531298, WO #98410121.
PT/0/A/4600/113, Operator Time Critical Task Verification, completed 11/21/02
PT/1/B/4700/087, Unit 1 Secondary Chemistry Periodic Surveillance for Primary to Secondary Leakage (SGs), completed 10/22/03; 11/18/03; 12/9/03; 1/6/04
PT/2/A/4150/001B, RC Leakage Calculation, completed ½/04;1/4/04;1/5/04;1/7/04;1/11-12/04; 1/12/04;1/16/04;1/18/04;1/19/04;1/21/04;1/23/04;1/25/04;1/26/04;1/29/04
PT/2/A/4250/033, SM PORV and PORV Isolation Valve Movement Test, completed 10/1/03
PT/2/A/4600/030, Cycling Time Critical Manually Operated Valves, completed 9/10/03
PT/2/B/4700/087, Unit 2 Secondary Chemistry Periodic Surveillance for Primary to Secondary Leakage (SGs), completed 10/22/03;11/18/03;12/9/03;1/6/04
PM 2EMDLP9100, Vibration/Loose Parts Monitor Channel Calibration, completed 9/13/2003
PM 2EMDLP9100, Vibration/Loose Parts Monitor Channel Calibration, completed 10/2/2003
PM 2EMDLP9100, Loose Parts Analog Channel Test, completed 12/18/2003
PM 2EMDLP9100, Loose Parts Analog Channel Test, completed 1/15/2003

PT/1/A/4600/003 B, Loose Part Detection System Daily Surveillance, completed 2/9/2004

Problem Investigation Process (PIP) Reports

M-97-00015, Evaluate operator response capabilities for required actions in doghouses
M-99-00701, TDCAP Steam Supply Check Valves Not Oriented per Manufacturer Recommendation
M-99-03034, Evaluate INPO Significant Event Notification 198
M-99-03239, Evaluate need to test valves associated with operator time critical tasks
M-00-02028, Review INPO SEN 213, SGTR at Indian Point Unit 2, for applicability at McGuire
M-01-04108, 2SA-77 Cannot be Closed in Timely Manner
M-01-04665, Dresser-Rand 10 CFR Part 21 on TDCAP Trip & Throttle Valve Spindle
M-02-02039, VI Air Quality PTs Failed Particulate Acceptance Criteria
M-02-02239, Contamination Found in Fuel Samples Taken From Bottom of G and H Diesel Air Compressors Fuel Tanks
M-02-05061, Low Margin on valve 2SM-3 (MSIV)
M-03-04000, Time critical valve 2CA-39 will not operate freely
M-03-05207, Failure of radiation monitor 2EMF-33
M-03-06021, Evaluate NRC event report from Surry
M-04-00357, Evaluate McGuire difference from other Westinghouse Owners Group plants in main feedwater isolation and main feedwater regulation bypass valve control
M-03-00543, 1CA42B Limit Switch Failed During Maintenance PMT Causing Stem Failure
M-00-01900, Unit 1 CA Pumps Normal Suction Sources Were Inadvertently Isolated Following a Reactor Trip and Automatically Aligned to RN, 5/26/2000
M-01-00951, CA Flow Control Valves Could Become Clogged When Aligned to RN
M-97-03989, Several Seismic Concerns Were Noted During Inspection of Control Cabinets in the Auxiliary Feedwater Pump Room
M-97-03161, Cable Tray did not appear to be supported above the Unit 1 and Unit 2 TDCAP
M-00-03261, PSA ½ kip Mechanical Snubber Would not Stroke Through It’s Full Range (2EOC13)
M-03-03784, Self Assessment on Chemistry Excursion Shutdown Limits
M-01-04115, Safety Review Assessment of Service Water Pipe Corrosion Program SA-01-35
M-03-05897, Wall Thickness Margin for 3 RN Pipe Corrosion Program Test Locations Are Being Reduced by Routine Corrosion
G-01-00169, Program Review Meeting for CFR-80 Model Steam Generators
G-02-00353, New Revision to “EPRI Pressurized Water Reactor Steam Generator Examination Guidelines,”
M-01-01387, Foreign Material Found On The Secondary Side of Steam Generator 1C During Secondary Side Inspection (SSI)
M-01-01330, Weld Slag Identified in Secondary Side of Steam Generator 1A During Post Sludge Lance Inspection
M-02-01204, Foreign Object Found and Removed from Steam Generator 2A during EOC14
M-02-00734, Portion of 4" NI Piping is Below Allowable Minimum Wall Thickness
M-97-04524, Applicability review of corrective actions taken at Cawtaba Nuclear Station concerning a CCW Pump breaker that failed to close on demand.
M-02-06094, WR#98248775 documents the failure of the 2B CA Pump Control switch.
M-02-04560, SOER 02-3, Large Power Transformer Reliability

M-03-03305, OEDB 03-33912, NRC IN 2003-06, Failures of Safety-related Valves Due To Linestarter Relay Degradation

Problem Investigation Process Reports (PIPs) Written Due to this Inspection

M-04-0509, Improvements are needed in the UFSAR descriptions provided for the steam line radiation monitors.
M-04-00644, VI System DBD incorrectly references 5 micron particulate limit. Criteria has been changed to 40 micron.
M-04-00667, NRC identified typographical error in Instrument Procedure, IP/0/B/3050/012, “Signal Processor Voltages,” - the tolerance section showed a tolerance range of 14.5 to 5.5. [this should indicate 14.5 to 15.5.]
M-04-00673, Abandoned sway strut hanging on steel immediately above the NE of the 2B motor driven CA pump motor. WR 98304305 written to remove the strut.
M-04-00678, Instrument Details for 1/2NSSV5560, 5570, 5580 & 5590 are misleading. MOVs are incorrectly shown on detail with a symbol of “SV.”
M-04-0710, Lack of Documentation To Verify Performance of Corrective Action on Steam Supply Check Valve Problem.
M-04-00713, NS pressure transmitter impulse tubing is not in compliance with MCID-2499-NS.01 Note 11, which states that tubing should be sloped upward from the tap to the transmitters and if this is not possible, a drain leg should be provided.

Vendor Documents and Technical Manuals

Sorrento Electronics Report E-115-1101, Review of Algorithms Used in General Atomic Company Radiation Monitoring Equipment, October 1980.
Sorrento Electronics Report No. 03608939, Expected Response Report for Model RD-26-32 In-Line Noble Gas Gamma Detector, February 19, 2003.
TM-8810B-001, RP86A Technical Manual, Rev. 2.
MCM 1205.05-0280 0001, JHF Pacific Pumps Operating and Maintenance Manual (SI Pump 1B), dated 1/24/03
MCM 1201.05-0249 001, Bingham Pumps Installation, Operation and Maintenance Manual - Multistage Horizontal Pumps, dated 9/12/00
MCM 1201/05-0228 001, Westinghouse/Pacific Pumps 1J Operating and Maintenance Instruction, dated 7/31/03
MCM 1205.01-0641 001, Installation, Operation Instruction Manual for 4" and 6" Automatic Recirculation Valves, Rev. 4
MCM 1205.09-0020 001, Maintenance Manual for SM PORV 1/2-SV-1, -7, -13, -19, Rev. 4
MCM 1205.12-0003 001, Instruction Manual for MSIVs, Rev. D23
MCM 1213.00-0207 002, Service Manual, Caterpillar 3406C Industrial Engine, Rev. 0
Information Bulletin (IB) 7.4.1.7-7, Type 27N High Accuracy Undervoltage Relay, Issue D
IB 18.4.7-2, Type 27D Single Phase Undervoltage Relay, Issue E
IB 7.1.1.7-2, Type 50G Ground Fault Protection relay, Issue A
IB 18.2.7-1, Type 51G Solid-State Overcurrent Relay, Issue F

Miscellaneous Documents:

Licensee Event Report (LER) 89-004 Rev.0, “A Steam Generator Tube Rupture Occurred on March 7, 1989 and Resulted in an Alert Being Declared and an Unplanned Release of Radioactivity”
Steam Generator Tube Rupture Analysis Notebook, SAAG File No. 624, Rev. 3
70016-COM, Software and Data Quality Assurance Document, RP86A Radiation Readout Module Firmware, Rev. 2, and Attachment 3 (Coding Description), Attachment 4 (Software Users Manual, Attachment 6 (Verification & Validation of Version 4.1), Attachment 7 (Verification & Validation of Version 4.1A), Attachment 8 (Verification & Validation of Version 4.2X2), and Attachment 9 (Verification & Validation of Version 4.2)
Letter from R. W. Eaker to G. E. Singletary, Subject: N-16 Leak Rate Calculations, January 1996
NRC Information Notice 93-56, Weakness in EOPs Found as a Result of SGTR
Background Document for AP/1 & 2/A/5500/010, (NC System Leakage Within the Capacity of Both NV Pumps), Rev. 1
Westinghouse Owners Group Abnormal Response Guidelines, Steam Generator Tube Leak (ARG-3), Rev. 0, and Associated Background Document
Westinghouse Owners Group Emergency Response Guidelines E-3, Steam Generator Tube Rupture, HP Rev. 1C, and Associated Background Document
McGuire Specific Differences From the Westinghouse E-3 Generic Guidelines, Rev. 11
SCM-8, Appendix B of McGuire Station Chemistry Manual, Secondary Chemistry Optimization Plan, Rev 5, 6/10/2002
Memo to B.H. Hamilton from R.W. Eaker, APE Meeting Action Item 88-13.04, 12/12/1988
MCM 3.2, Secondary Systems Analytical Requirements and Corrective Actions, Rev. 26
MTP 3103.0, Chemistry Training Program, Rev. 8, 7/28/2003
SLC 16.5.7, Chemistry, Rev. 0
Letter DAP-87-555 from Westinghouse Electric Corporation to Duke Power Company, McGuire and Catawba Nuclear Stations Feedwater Isolation Setpoint Study, dated 4/13/1987
SGMEP 105, CFR80 Specific Assessment of Potential Degradation Mechanisms, McGuire Unit 2 EOC 14, Rev. 3
SGMEP 105, CFR80 Specific Assessment of Potential Degradation Mechanisms, McGuire Unit 2 EOC 15, Rev. 4
MCC-2201.37-00-0005, ‘Appendix A’ Condition Monitoring and Operational Assessment for McGuire Unit 2 EOC 14, Rev. 0
NEI 97-06, SG Program Guidelines, Rev. 1
EPRI TR1003138, Pressurized Water Reactor SG Examination Guidelines, Rev. 6
EPRI TR-102134, Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) Secondary Water Chemistry Guidelines, Rev. 6
Regulatory Guide 1.83, Inservice Inspection of PWR SG Tubes
Regulatory Guide 1.133, Loose-Part Detection Program For the Primary System of Light-Water-Cooled Reactors, May 1981
Information Notice 2004-01, AFW Pump Recirculation Line Orifice Fouling - Potential Common Cause Failure, 1/21/2004

Section 1R21.31.a.: List of Components Reviewed

2SV-1AB,-7ABC,-19AB (SG PORVs)
2SA-48ABC, -49ABC (TDCA pump steam supply)
2SV-25,-26,-27,-28 (SG PORV block valves))
2SM-89, (SM drain line isolation)
2RN-103A, -204B (RN cooling supply to NV pump)
2RN-114A,-215B (RN cooling supply to NI pump motor)
2NV-141A,-142B (VCT outlet)
2NV-221A,-222B (FWST outlet)
2CA-22,-26,-3 (CA pumps' automatic re-circulation valves)
2NC-34A,-34B (N2 supply to pressurizer PORV)
2SMSV-0010,-0011,-0012 (SM isolation valve solenoids)
2SVSV-0011,-0012,-0071,-0072 (SG PORV solenoids)
2CASV-0600,-0601,-0560 (motor driven CA pump discharge flow solenoid)
2SV 8-12, 14-18, 20-24 (SM safety valves)
2NI-15,-17,-19,-22 (NI discharge check valve)
2CA-45,-49,-57,-61 (CA discharge check valve)
2NV-15,-18 (NV discharge check valve)
pumps (NI, NV, and CA)
VI system diesel air compressors

Section 1R21.35.a.: List of Operating Experience Items Reviewed

OEDB-93-006083, INPO SOER 93-1, Diagnosis and Mitigation of RCS Leakage, Including SG Tube Ruptures (Palo Verde Unit 2 SGTR event)
OEDB 99-020643, Flooding of main steam lines due to bypassing reactor feed pump high level trip
OEDB-00-024862, INPO SEN 213, SG Tube Failure at Indian Point Unit 2
OEDB Number 04-035581, Screen of IN 2004-01, 1/26/2004
M-03-03305, OEDB 03-33912, NRC IN 2003-06, Failures of Safety-related Valves Due To Linestarter Relay Degradation
M-01-04665, Dresser-Rand 10 CFR Part 21 on TDCAP Trip & Throttle Valve Spindle
M-00-02028, Review INPO SEN 213, SGTR at Indian Point Unit 2, for applicability at McGuire
M-03-06021, Evaluate NRC event report from Surry
M-99-03034, Evaluate INPO Significant Event Notification 198
M-02-04560, SOER 02-3, Large Power Transformer Reliability

1R21.4.a.: List of PIPs Screened and Sampled

PIPs Involving Steam Generator Narrow Range Level Instrumentation:

M-01-00400	M-01-01477	M-01-05053
M-02-01004	M-02-02290	M-02-01885
M-02-03235	M-02-04211	M-02-05261
M-03-00198	M-03-02777	M-03-02536

M-03-02126	M-03-03442	M-03-03948
M-03-04984	M-04-00352	M-04-00357

PIPs Involving Radiation Monitors Used for Detection of Primary/Secondary Leaks or SGTR:

M-01-01425	M-02-00747	M-02-01711
M-02-00174	M-02-03812	M-01-02252
M-01-03679	M-01-04180	M-01-04181
M-01-04285	M-01-04544	M-01-05382
M-02-00077	M-02-00691	M-02-00693
M-02-02373	M-02-02553	M-02-02649
M-02-04708	M-02-04914	M-02-05585
M-02-05734	M-02-05973	M-03-02830
M-03-05207	M-03-05274	M-03-06115
M-01-00120	M-01-01568	M-01-01641
M-01-02167	M-01-02682	M-01-02689
M-01-03763	M-01-04403	M-01-04568
M-01-04676	M-02-03169	M-02-03330
M-02-03353	M-02-03662	M-02-04249
M-02-04902	M-02-05476	M-02-06155
M-03-00184	M-03-01080	M-03-01081
M-03-01142	M-03-01630	M-03-02244
M-03-03529	M-03-04198	M-03-04631
M-03-04736	M-03-04880	M-03-05481
M-03-05994	M-01-00393	M-01-00572
M-01-00610	M-01-00677	M-01-00841
M-01-02296	M-01-02875	M-01-04660
M-03-01091	M-03-02618	M-01-02383
M-01-02633	M-01-02799	M-02-03243
M-02-03412	M-03-01979	M-01-00723
M-02-01910	M-02-03260	

PIPs Involving Pressurizer PORV Control:

M-01-00536	M-02-00805	M-02-00975
M-02-00315	M-02-04703	M-03-03089
M-03-06049	M-04-00245	

PIPs Involving Steam Generator PORV Control:

M-01-02044	M-01-03139	M-01-03224
M-03-01104	M-03-03883	