UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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Duke Energy Corporation
ATTN: Mr. D. Jamil
Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
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SUBJECT: MCGUIRE NUCLEAR STATION - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 50-369/03-07 AND 50-370/03-07

Dear Mr. Jamil:

On May 23, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your McGuire Nuclear Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on May 22, 2003, with you and other members of your staff.

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

This report documents three findings that have potential safety significance greater than very low significance, however, a safety significance determination has not been completed. These findings did not present an immediate safety concern, however, a fire watch was initiated on June 10, 2003, as a compensatory measure for one of the findings.

If you contest any violation 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 2.790 of the NRC's "Rules of Practice," a copy of this letter and 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

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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,

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

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

Enclosure: Inspection Report 50-369, 370/03-07
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 No.:	50-369/03-07 and 50-370/03-07
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station
Location:	12700 Hagers Ferry Road Huntersville, NC 28078
Dates:	May 5 - 9, 2003 (Week 1) May 19 - 23, 2003 (Week 2)
Inspectors:	P. Fillion, Reactor Inspector R. Maxey, Reactor Inspector B. Melly, Fire Protection Engineer (Consultant) R. Schin, Senior Reactor Inspector (April 14-17, 2003) M. Thomas, Senior Reactor Inspector (Lead Inspector)
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR05000369/03-07, IR05000370/03-07; Duke Energy Corporation; 05/05-09/2003 and 05/19-23/2003; McGuire Nuclear Station, Units 1 and 2; Triennial Fire Protection

The report covered a two-week period of inspection by regional inspectors and a consultant. Three unresolved items with potential safety significance greater than Green were identified. 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. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- TBD. The team identified a violation because Train A and Train B cables associated with the reactor protection system were located in the same fire area and were not protected from fire damage, as required by McGuire's fire protection program.

  This finding is unresolved pending determination of the systems affected and completion of a significance determination. This finding is greater than minor because it was associated with the equipment performance attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events in that instrumentation important for post-fire safe shutdown could be lost. This finding did not present an immediate safety concern, however, a fire watch was initiated on June 10, 2003, as a compensatory measure. When assessed in combination with the finding related to inadequate protection of auxiliary feedwater system cables and equipment required for safe shutdown in Fire Area 16/18 (also discussed in this inspection report), this finding may have potential safety significance greater than very low significance. (Section 1R05.03.b.1)

- TBD. The team identified a violation in that the turbine driven auxiliary feedwater (TDAFW) pump suction supply valve 2CA0007A was not evaluated in the licensee's safe shutdown analysis for potential impact on safe shutdown in the event of a fire where the TDAFW pump is required for safe shutdown. The valve could spuriously operate due to fire damage and adversely affect the TDAFW pump.

  The finding is unresolved pending completion of a significance determination. The finding is greater than minor because it was associated with the equipment performance attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events in that spurious closure of the valve could damage the TDAFW pump and seriously degrade the decay heat removal function. This finding may have potential safety significance greater than very low significance. (Section 1R05.04.b.2)

B. Licensee Identified Violations

- TBD. The physical protection of cables and equipment relied upon for safe shutdown (SSD) of Unit 2 during a fire in the Train A Electrical Penetration Room (Fire Area 16/18) was not adequate. Train B electrical cables, associated with the 2B motor driven auxiliary feedwater pump discharge valve 2CA0042B to steam generator 2D, were located in the Train A Electrical Penetration Room (Fire Area 16/18) without adequate spatial separation or fire barriers as required by the McGuire fire protection program. Local, manual operator actions (which had not been reviewed and approved by NRC) would be used to achieve and maintain SSD of Unit 2 in lieu of providing adequate physical protection for the electrical cables associated with valve 2CA0042B.

  This finding is unresolved pending completion of a significance determination. The finding is greater than minor because it was associated with the equipment performance attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events in that fire damage to the unprotected cables could prevent operation of SSD equipment from the main control room. When assessed in combination with the inadequate reactor protection system cable separation finding (also discussed in this inspection report), this finding may have potential safety significance greater than very low significance. (Section 1R05.03.b.2)

Report Details

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R05 FIRE PROTECTION

The purpose of this inspection was to review the McGuire Nuclear Station (MNS) fire protection program (FPP) for selected risk-significant fire areas. Emphasis was placed on verification that the post-fire safe shutdown (SSD) capability and the fire protection features provided for ensuring that at least one redundant train of safe shutdown systems is maintained free of fire damage. The inspection was performed in accordance with the Nuclear Regulatory Commission (NRC) Reactor Oversight Program using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team used the licensee's Individual Plant Examination for External Events (IPEEE) and performed in-plant walk downs to choose four risk-significant fire areas for detailed inspection and review. The four fire areas selected were:

- Fire Area 4, Auxiliary Building (AB) Common Area; AB +716 feet elevation
- Fire Area 13, Battery Rooms; AB +733 feet elevation common area
- Fire Area 16/18, Unit 2 Train A Electrical Penetration Room/2ETA 4160 volt Switchgear Room; AB +750 feet elevation
- Fire Area 24, Main Control Room (MCR); AB +767 feet elevation

For each of the selected fire areas, the team focused the inspection on the fire protection features, and on the systems and equipment necessary for the licensee to achieve and maintain safe shutdown conditions in the event of a fire in those fire areas.

The team evaluated the licensee's FPP against applicable requirements, including Operating License Conditions 2.C.4 and 2.C.7, Fire Protection Program, for Units 1 and 2, respectively; Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), Appendix R, Sections III. G, J, L, and O; 10 CFR 50.48; Appendix A to Branch Technical Position Auxiliary and Power Conversion Systems Branch 9.5-1, Guideline for Fire Protection for Nuclear Power Plants; related NRC Safety Evaluation Reports (SERs); MNS Updated Final Safety Analysis Report (UFSAR), Section 9.5.1; UFSAR Section 16.9, Selected Licensee Commitments (SLC); and plant Technical Specifications (TS). The team evaluated all areas of this inspection, as documented below, against these requirements.

.01 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The team reviewed the licensee's FPP described in UFSAR Section 9.5.1; the MNS Fire Protection Review; safe shutdown analysis (SSA); fire hazards analysis (FHA); SSD essential equipment list; and system flow diagrams to identify the components and systems necessary to achieve and maintain SSD conditions. For each of the selected fire areas, the team focused on the fire protection features, and on the systems and equipment necessary for the licensee to achieve and maintain SSD in the event of a fire in those fire areas. The following Unit 2 systems and components were selected for review:

- Standby Shutdown System (SSS)
- Standby makeup pump (SMP) 2NVPU0046
- SMP suction supply valve 2NV842AC
- Auxiliary feedwater (AFW) suction supply valves 2CA007A and 2CA009B
- Reactor Coolant Pump (RCP) seal water return isolation valve 2NV94AC
- Pressurizer power operated relief valve (PORV) 2NC34A
- PORV isolation valve 2NC33A
- Pressurizer heaters No. 28, 55, 56
- Reactor vessel head vent valves 2NC272AC and 2NC273AC
- Heating, ventilation, and air conditioning (HVAC)

Specific licensee documents, calculations, and drawings reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.02 Fire Protection of Safe Shutdown Capability

a. Inspection Scope

The team reviewed the fire detection system protecting Fire Areas 4, 13, 16/18 and 24 to assess the adequacy of the design and installation. This was accomplished by reviewing design drawings, ceiling beam location drawings, and National Fire Protection Association (NFPA) 72E (code of record 1974 edition) for detector location requirements. The team reviewed the McGuire Fire Protection Code Deviation Calculation to determine if there were any outstanding code detector deviations for the selected areas. The team walked down the fire detection and alarm systems in Fire Areas 13 and 16/18 to evaluate the installed detector locations relative to the NFPA 72E location requirements. Additionally, the team reviewed the surveillance test procedures for the detection and alarm systems to determine compliance with UFSAR Sections 9.5.1 and 16.9.

The team reviewed the adequacy of the design and installation of the fire suppression system protecting the nuclear service water (RN) pump area in Fire Area 4. This was accomplished by reviewing the engineering design drawings, suppression system

hydraulic calculations, as-built system configuration and NFPA 13 (code of record 1978 edition) for sprinkler system location requirements. The team also reviewed the McGuire Fire Protection Code Deviation Calculation for the RN pump sprinkler system to determine the adequacy of the system to control a fire in this area utilizing the 2-1/2 inch by-pass lines as the sole means of supplying the sprinkler system.

The team reviewed the fire hose stations in Fire Areas 4, 13, 16/18 and 24 to assess the adequacy of the design and installation. This was accomplished by reviewing the fire plan drawings, engineering mechanical equipment drawings, pre-fire strategies and NFPA 14 (code of record 1976 edition) for hose station location requirements and effective reach capability. Team members also performed a field walkdown of the selected fire areas to ensure that hose stations were not blocked and to compare hose station location drawings with as-built plant locations.

b. Findings

The team identified an unresolved item (URI) involving the adequacy of the suppression system for Fire Area 4. Dedicated shutdown (DSD) using the SSS was designated by the licensee for a fire in this area. 10 CFR 50, Appendix R, Section III.G.3 (alternative or dedicated shutdown) requires that fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration. The fire suppression system for Fire Area 4 was not installed in accordance with 10 CFR 50, Appendix R, Section III.G.3. The system in Fire Area 4 was a partial automatic sprinkler system effectively protecting the RN pumps and 20 feet north of these pumps. The area protected by this sprinkler system was located between column lines 54-58 and EE-GG. The majority of Fire Area 4 was not provided with automatic sprinkler protection as required by 10 CFR 50, Appendix R, Section III.G.3.

This issue was previously identified by the NRC (URI 50-369/84-28-01, 370/84-25-01) in 1984 during an Appendix R inspection. The licensee considered this issue to be a potential backfit per 10 CFR 50.109 (letter dated September 4, 1984, from H.B. Tucker, Duke Power Company, to H.R. Denton, NRC Office of Nuclear Reactor Regulation). The URI was closed in NRC inspection report (IR) 50-369,370/87-34. The team noted that, subsequent to closure of the URI, licensee Fire Protection Functional Audit SA-99-04(MC)(RA)(FPFA) dated April 9, 1999, identified that MNS did not meet separation and detection/suppression criteria for alternative or dedicated shutdown capability required by 10 CFR 50, Appendix R, Section III.G.3. During the current inspection, the team questioned whether the previous reviews of the sprinkler system for this fire area included an evaluation of the risk impact associated with not providing adequate sprinkler coverage for the RN cabling in this fire area. The team informed the licensee that this issue would be reviewed to determine if the lack of sprinkler coverage in this fire area has an impact on risk. The team noted that a similar condition exists in other fire areas where dedicated shutdown capability using the SSS was designated by the licensee. Pending determination of whether a backfit evaluation is warranted, this issue is identified as URI 50-369, 370/03-07-01, Fire Suppression System for Dedicated Shutdown Areas not in Accordance with 10 CFR 50, Appendix R, Section III.G.3.

.03 Post-Fire Safe Shutdown Circuit Analysis

a. Inspection Scope

The team reviewed the adequacy of separation and fire barriers provided for the power and control cabling of equipment relied on for SSD during a fire in the selected fire areas. On a sample basis, the team reviewed the SSA and the electrical schematics for power and control circuits of SSD components, and looked for the potential effects of open circuits, shorts to ground, and hot shorts. This review focused on the cabling of selected components of the charging/makeup system, reactor coolant system (RCS) and AFW system. The team traced the routing of cables by using the cable schedule and conduit and cable tray drawings. The team walked down the selected fire areas to compare the actual plant configuration to the cable layout on the drawings. Circuit and cable routings were reviewed for the following equipment:

- 0RN4AC, Turbine Driven AFW Suction Supply Valve
- 2CA0007A, Turbine Driven AFW Suction Isolation Valve
- 2CA009B, Motor Driven AFW Suction Isolation Valve
- 2CFLT6080, 6090, 6100, 6110, Steam Generator Level Transmitters
- 2NCLT5151, Pressurizer Level Transmitter
- 2NC34A, Pressurizer PORV
- 2NC33A, PORV Isolation Valve
- 2NC272AC, 273AC, Reactor Vessel Head Vent Valves
- 2NVPU0046, Standby Makeup Pump
- 2NV94AC, RCP Seal Water Return Isolation Valve
- 2NV842AC, SMP Suction Isolation Valve
- 2NV1012C, SMP Discharge to Containment Sump Isolation Valve
- Pressurizer heaters No. 28, 55, 56

The team also reviewed licensee studies of overcurrent protection for alternating current and direct current systems to identify whether fire-induced faults could result in defeating the SSD functions.

b. Findings

Findings associated with valves 2CA0007A, 2NC34A, and 2NC33A are discussed in Section .04 of this IR.

1. Reactor Protection System

Introduction: A finding with potentially greater than very low safety significance was identified in that redundant instrumentation (and possibly other equipment) important to SSD could be damaged by a fire in Fire Area 16/18. This finding involved a violation of NRC requirements. This finding is a URI pending completion of the SDP.

Description: Fire Area 16/18 is the Unit 2 Train A electrical penetration room/2ETA 4160 volt (V) switchgear room. Train B equipment controlled from the MCR room was designated as the SSD train for a fire in this area according to the SSA and plant procedures. During a walkdown of Fire Area 16/18, the team identified that room 805A lacked fire detection and fire suppression. Room 805A is the HVAC equipment room which supplies ventilation to the Unit 2 Train A 4160V switchgear room 2ETA. The team also observed that Train B cables were routed through room 805A. Many of the identified cables were in cable trays near the ceiling and were going from/to the cable spread room, which was on the same elevation; and to/from the control room, which was above room 805A. The licensee had not been aware these Train B cables passed through room 805A, and initiated Problem Investigation Process (PIP) M-03-02106 and M-03-02588. [The team identified that a similar condition also existed in room 803A (Fire Area 17), which is the HVAC equipment room providing ventilation for the Unit 1 Train A 4160V switchgear room 1ETA]. On June 10, 2003, the licensee reported that these cables did not meet the separation criteria of Appendix R and represented an unanalyzed condition (Event No. 39915), and initiated a fire watch as a compensatory measure.

Preliminary investigation by the licensee revealed that cables for primary and backup power supplies for all four reactor protection system (RPS) channels were routed in close proximity and could be damaged during a severe fire. As many as 74 Train B RPS cables may be involved. One consequence of this finding is that fire-induced cable damage may cause many RPS protective functions to spuriously go to the trip condition. Consequently, a safety injection signal could be generated due to spurious high containment pressure. The safety injection signal could in turn trigger a reactor trip and Phase A isolation. [At the same time, many main control panel instruments necessary to achieve and maintain hot shutdown would be lost, including pressurizer level and all four steam generator (SG) level instruments.] The licensee also stated that similar effects could occur for a fire in the Unit 1 Train A switchgear room 1ETA (Fire Area 17).

Analysis: The team determined that this finding was associated with the equipment performance attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events, and is therefore greater than minor. The finding did not present an immediate safety concern, however, the licensee initiated a fire watch on June 10, 2003, as a compensatory measure. The licensee is analyzing the manner in which plant systems would be affected by fire damage to the Train B cables and is reviewing plant abnormal procedures (APs) in light of the degraded instrumentation and any automatic actions that would be initiated. Once the equipment degradations and relevant procedures are understood, the significance determination process (SDP) will be used to determine the level of significance. When assessed in combination with the finding related to inadequate protection of AFW cables and equipment required for SSD in Fire Area 16/18 (Section .03.b.2), this finding may have potential safety significance greater than very low significance.

Enforcement: The licensee's FPP commits to 10 CFR 50, Appendix R, Section III.G. Section III.G.~~1.a.~~ II states, in part, that one train of systems necessary to achieve and maintain hot shutdown shall be free of fire damage.

Contrary to the above, redundant trains of instrumentation necessary to achieve and maintain hot shutdown could be damaged during a fire in room 805A (Fire Area 16/18). Pending determination of the safety significance, the finding is identified as URI 50-369, 370/03-07-02, Failure to Protect Redundant Trains of Reactor Protection System Cables From the Effects of Fire.

2. Inadequate Protection of AFW Cables and Equipment Required for Safe Shutdown

Introduction: A finding was identified in that physical protection of the associated electrical cables for valve 2CA0042B (2B motor driven AFW pump discharge supply to SG 2D) did not meet the requirements of 10 CFR 50, Appendix R, Section III.G.2. Instead, the licensee used a local manual operator action, which had not received prior NRC approval, to achieve and maintain SSD. This is a URI pending completion of the SDP.

Description: The licensee identified (April 2003) that MNS relied on local, manual operator actions outside the MCR for SSD in non-dedicated shutdown fire areas (i.e., areas designated as complying with 10 CFR 50, Appendix R, Section III.G.2). These local, manual operator actions did not have prior NRC approval. The licensee documented this issue in PIP M-03-02311. The team reviewed the local, manual operator action for the Appendix R, Section III.G.2 fire area selected for this inspection (Fire Area 16/18).

The team found that the associated electrical cables for Train B valve 2CA0042B were located in the Unit 2 Train A electrical penetration room (Fire Area 16/18) without adequate spatial separation or fire barriers. Rather than providing adequate physical protection for redundant trains of equipment/systems necessary to achieve and maintain SSD (as specified for Appendix R, Section III.G.2 areas), the licensee substituted the use of a manual operator action outside the MCR. The licensee's SSA stated that de-energizing this valve, after verifying that it was open, was a time critical action because spurious closure of this valve would limit the secondary heat sink to only one SG (rather than the two required to achieve and maintain SSD). The use of local manual operator actions, in fire areas designated as complying with the provisions of Appendix R, Section III.G.2, requires prior NRC review and approval. This local, manual operator action had not received NRC approval.

Analysis: The team determined that this finding was associated with the equipment performance attribute of the mitigating systems cornerstone. It affected this cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events, and is therefore greater than minor. When assessed in combination with the inadequate RPS cable separation finding (Section .03.b.1), this finding may have potential safety significance greater than very low significance.

Enforcement: The licensee's FPP commits to 10 CFR 50, Appendix R, Section III.G. Section III.G.2 states in part, that,

> "...where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided: (1) separation of cables and equipment of redundant trains by a fire barrier having a 3-hour rating; (2) separation of cables and equipment of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; (3) enclosure of cables and equipment of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area."

Contrary to the above, on May 23, 2003, the licensee failed to protect cables of redundant equipment located within the Unit 2 Train A electrical penetration room/4160V switchgear room 2ETA (Fire Area 16/18) with an adequate barrier or to provide 20 feet of separation. Pending determination of the finding's safety significance, this finding is identified as URI 50-370/03-07-05, Failure to Provide Adequate Protection for Cables of Redundant Safe Shutdown Equipment in Fire Area 16/18.

.04 Alternative Post-Fire Safe Shutdown Capability

a. Inspection Scope

The team reviewed the licensee's procedures for fire response, APs for DSD, and the licensee's Appendix R fire area failure analysis and compliance strategy for a fire in Fire Areas 4, 13, and 24. The team also walked down selected portions of the procedures in the plant. The reviews focused on ensuring that the required functions for post-fire safe shutdown and the corresponding equipment necessary to perform those functions were included in the procedures. The review also included assessing whether hot and cold shutdown from outside the MCR could be implemented, and that transfer of control from the MCR to the standby shutdown facility (SSF) could be accomplished within the performance goals stated in 10 CFR 50, Appendix R, Section III.L. The components listed in Section .03.a. of this IR were also reviewed in relation to DSD capability. The team reviewed the most recently completed surveillances for selected instruments required during SSS operation to verify that these surveillances were being completed in accordance with MNS SLC 16.9.7, Standby Shutdown System. The team walked down DSD procedures to determine if they could be performed within the required times given the minimum required staffing level of operators, with or without offsite power available.

The team also reviewed the electrical isolation of selected motor operated valves from the control room to verify that operation of the SSS from the SSF, and other remote plant locations, would not be prevented by a fire-induced circuit fault.

b. Findings

1. Requirements Relative to the Number of Spurious Operations that Must be Postulated

Introduction: The team identified an issue involving the number of concurrent spurious operations associated with a particular component or set of components that must be postulated during SSD analysis of a fire area. This issue is a URI pending review by NRC staff.

Description: The licensee's SSA included the concept that only one spurious operation due to fire damage need be postulated. This concept became evident during review of the pressurizer PORVs. There are three sets of PORVs and PORV isolation valves on the pressurizer of each unit. Should operators in the control room become aware of a fire in any plant area (from a fire alarm or the plant communications system), they would respond by implementing procedure AP/0/A/5500/045, Plant Fire. Depending on the fire location, procedure AP/0/A/5500/045 directed the operator to close the PORV isolation valves within ten minutes. The basis for this time critical action is that spurious opening of the PORV, or damage to the isolation valve circuit would not occur in the first ten minutes of a fire being detected. With the isolation valve closed, it would then take two spurious operations to breach the RCS pressure boundary (i.e., the isolation valve opening and its associated PORV also opening). This concept of postulating only one spurious operation meant that closing the isolation valve was sufficient to ensure RCS pressure boundary integrity. The licensee considered that there was no need to take any other action such as de-energizing the isolation valve after it was closed. Application of this concept is not consistent with NRC's cable protection requirements of Appendix R, Section III.G.

The team reviewed the control circuits and cable routing information for pressurizer PORV 2NC34A, and its associated isolation valve 2NC33A. They observed that cables for both the PORV and isolation valve were routed through Fire Areas 13, 16/18 and 24. The team determined that, for these three fire areas, spurious opening of the PORV could only occur for a MCR fire (Fire Area 24). If more than one spurious operation were to occur, the dedicated shutdown capability (SSS) would not be independent from the MCR in that a fire in the control room could result in conditions outside those specified in Appendix R, Section III.L.

Analysis: The team determined that this finding was associated with the equipment performance attribute of the mitigating systems cornerstone. Because it affected this cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events, this finding is greater than minor. If more than one spurious operation were to occur, the dedicated shutdown capability (SSS) would not be

independent from the MCR in that a fire in the MCR could result in conditions outside of those specified in Appendix R, Section III.L.

Enforcement: In the case of the PORV and PORV isolation valve circuits, operation of the SSS may not be independent of the fire area as required by Appendix R, Section III.G.3. Review of this matter by the NRC will determine whether a violation has occurred. Pending the issuance of new NRC inspection guidance regarding associated circuits, the issue is identified as URI 50-369, 370/03-07-03, Requirements Relative to the Number of Spurious Operations That Must be Postulated.

2. Auxiliary Feedwater Valve 2CA0007A Not Included in Safe Shutdown Analysis

Introduction: A finding with potentially greater than very low safety significance was identified in that AFW suction supply valve 2CA0007A, which could spuriously operate during a MCR fire, was not included in the SSA. Spurious closure of this valve could damage the turbine driven auxiliary feedwater (TDAFW) pump, thus seriously degrading the secondary decay heat removal function of the SSS. This is a URI pending completion of the SDP.

Description: Valve 2CA0007A is a motor operated valve in the suction flow path from the 300,000 gallon AFW storage tank to the TDAFW pump. The valve is open during normal plant operation. 2CA0007A is important to safe shutdown for fire areas where the SSS will be used. The importance is derived from the fact that the SSS relies on the TDAFW pump for secondary decay heat removal. Spurious closure of the valve would immediately reduce suction pressure and quickly shut off all flow through the pump causing severe damage. For a severe fire in the MCR requiring evacuation and transfer of plant shutdown to the SSS, the ability to remove decay heat would be seriously degraded if the TDAFW pump were damaged. The team found that the SSA did not include valve 2CA0007A. The valve was not listed in Appendix E, Unit 1 and Unit 2 Safe Shutdown Equipment; nor Appendix F, Fire Area Failure Analysis and Compliance Strategy, of the SSA (Specification MCS-1465.00-00-0022, Design Basis Specification for Appendix R).

The licensee initiated PIPs M-03-02084, M-03-02118, and M-03-02311 for this issue and took prompt action to prevent spurious operation of this valve. Procedure AP/0/A/5500/045 was revised to specify that the operator ensure, within the first ten minutes of an active fire, that valve 2CA0007A was open and then remove power from 2CA0007A.

The team noted that system design provided for automatic transfer to alternate suction sources initiated by pressure switches in the TDAFW pump suction line. There were three separate alternate suction flow paths. Path 1 was through valves 2CA161C, 2CA162C and 0RN4AC; Path 2 was through valves 2CA086A and 2RN069A; and Path 3 was through valves 2CA116B and 2RN162B. However, key information related to these automatic transfers was not available to the team during the inspection.

Information was subsequently provided to the team, however, this information has not yet been fully reviewed.

Analysis: The team determined that this finding was associated with the equipment performance attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events, and is therefore greater than minor. For a severe fire in the MCR, the MCR would be evacuated and the SSF would be used to achieve and maintain hot shutdown. Because the SSF relies on the TDAFW pump for the decay heat removal, the decay heat removal function would be seriously degraded if the TDAFW pump were damaged due to closure of valve 2CA0007A.

Enforcement: 10 CFR 50.48 states, in part, that each operating nuclear power plant must have a fire protection program that satisfies Criterion 3 of 10 CFR 50, Appendix A. MNS Unit 2 Operating License NPF-17, Condition 2.C.(7) states, in part, that the licensee shall implement and maintain in effect all provisions of the approved FPP as described in the UFSAR for the facility, and as approved in the SER dated March 1978 and SER Supplements 2, 5, and 6 dated March 1979, April 1981, and February 1983, respectively, and the safety evaluation dated May 15, 1989.

The UFSAR states that the overall concept and details of the FPP are presented in the MNS Fire Protection Review (MCS-1465.00-00-0008). The FPP, which includes the SSA (MCS-1465.00-00-0022) for MNS, states in part, that the FPP implemented the philosophy of defense-in-depth protection against fire hazards and effects of fire on SSD equipment. It further states that the SSA performed for MNS considered potential fire hazards and their possible effects on SSD capability. The licensee's SSA designated the MCR (Fire Area 24) and Fire Area 4 as dedicated shutdown areas. Appendix R, Section III.G.3 requires that the alternative/dedicated shutdown capability, and its associated circuits, be independent of cables, systems or components in the area under consideration.

Contrary to these requirements, valve 2CA0007A was not included in the SSA resulting in the dedicated shutdown system (SSS) not being independent from Fire Area 24, in that, a fire in these areas could result in spurious closure of this valve and damage to the TDAFW pump. Pending determination of the safety significance, this finding is identified as URI 50-370/03-07-06, Spurious Closure of Valve 2CA0007A Could Lead to Damage of the TDAFW Pump.

.05 Operational Implementation of Post-Fire Safe Shutdown Capability

a. Inspection Scope

The team reviewed the operational implementation of the SSD capability for a fire in Fire Areas 4, 13, 16/18, or 24 to verify that: (1) the training program for licensed personnel included dedicated safe shutdown capability; (2) personnel required to achieve and maintain the plant in hot standby following a fire using the SSS could be provided from

normal onsite staff, exclusive of the fire brigade; (3) the licensee had incorporated the operability of dedicated shutdown transfer and control functions into plant TS and/or SLCs; and (4) the licensee periodically performed operability testing of the dedicated shutdown instrumentation, and transfer and control functions. The team reviewed procedures AP/1/A/5500/24 and AP/2/A/5500/024, Loss of Plant Control Due to Fire or Sabotage, and AP/0/A/5500/045, Plant Fire. The reviews focused on ensuring that all required functions for post-fire safe shutdown, and the corresponding equipment necessary to perform those functions, were included in the procedures.

b. Findings

The licensee identified that local, manual operator actions outside the MCR were used in lieu of physical protection of equipment and cables relied upon for SSD during a fire without obtaining prior NRC approval. Findings related to this issue for Fire Area 16/18 are discussed in Section 03.b.2 of this IR.

The team identified a URI regarding the adequacy of the licensee's method for controlling RCS pressure during operation from the SSF in the event of a fire. During review of procedures AP/1/A/5500/024 and AP/2/A/5500/024, the team questioned the adequacy of the 70 kilowatts (kW) pressurizer heater capacity (per unit) powered from the SSF to maintain and control RCS pressure in hot standby during a fire in plant areas which require use of the SSS. A procedural note in both AP/1/A/5500/024 and AP/2/A/5500/024 provided guidance to the operators which stated that it was acceptable to allow the pressurizer to go water solid in order to maintain subcooling, and with the pressurizer water solid, the reactor vessel head vents would be used to control pressure. Allowing the pressurizer to go water solid for controlling RCS pressure during hot standby conditions while operating from the SSF was not consistent with Appendix R, Section III.L, for dedicated shutdown capability, nor the design basis description for the SSF as stated in the licensee's letter to the NRC dated March 31, 1980. Also, solid plant operation from the SSF for controlling RCS pressure was neither reviewed nor discussed in any NRC SER/SER Supplements relative to acceptability of the SSF design for dedicated shutdown capability. The team requested information from the licensee (e.g., analyses, calculations, etc.) which demonstrated the following:

- Adequacy of the 70 kW pressurizer heater capacity powered from the SSF for maintaining and controlling RCS pressure in hot standby.

- Validity of the assumptions for pressurizer heat loss stated in the October 21, 1980, letter (based on insulation degradation and/or degraded capacity of the heaters powered from SSF) for current pressurizer heat loss and for determining when the heaters will be needed.

- SMP capacity to achieve and control solid plant operation from the SSF within the required time to maintain subcooling.

- Operator training (job performance measures, simulator, etc.) on solid plant operation from the SSF.

The licensee indicated that there were no specific calculations documented which provided the basis for the number of heaters to be powered from the SSF. The licensee further stated that there was no calculation which demonstrated the performance capability of the SMP during solid plant operation from the SSF. The licensee also indicated that training provided to operators on solid plant operation from the SSF consisted primarily of classroom discussions and tabletop discussions of procedures AP/1/A/5500/024 and AP/2/A/5500/024. The team concluded that sufficient information was not provided to resolve the questions raised above nor to determine the licensee's ability to safely operate the SSF with the pressurizer in a water solid condition during fire events in areas where the SSF is used to achieve SSD. Pending further NRC review of additional licensee information, this issue is identified as URI 50-369,370/03-07-04, Reactor Coolant System Pressure Control During SSF Operation.

.06 Communications

a. Inspection Scope

The team reviewed plant communication capabilities to verify that they were adequate to support unit shutdown and fire brigade duties. This included verifying that site paging portable radios, and sound-powered phone systems were consistent with the licensing basis and would be available during fire response activities. The team reviewed the licensee's communications features to assess whether they were properly evaluated in the licensee's SSA (protected from exposure fire damage) and properly integrated into the post-fire SSD procedures. The team also walked down sections of the post-fire SSD procedures to verify that adequate communications equipment would be available to support the SSD process.

b. Findings

No findings of significance were identified.

.07 Emergency Lighting

a. Inspection Scope

The team compared the installation of the licensee's emergency lighting systems to the requirements of 10 CFR 50, Appendix R, Section III.J, to verify that 8-hour emergency lighting coverage was provided in areas where manual local operator actions were required during post-fire SSD operations, including the access and egress routes. The team's review also included verifying that emergency lighting requirements were evaluated in the licensee's SSA and properly integrated into the post-fire SSD procedures. During team walk downs of the selected areas where local, manual

operator actions would be performed, area emergency lighting units were inspected for operability and the aiming of lamp heads was checked to determine if adequate illumination would be available to correctly and safely perform the actions directed by the procedures.

b. Findings

No findings of significance were identified.

.08 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed the licensee's SSA and existing plant procedures to determine if any repairs were necessary to achieve cold shutdown, and if needed, the equipment and procedures required to implement those repairs were available onsite.

b. Findings

No findings of significance were identified.

.09 Fire Barriers and Fire Area/Zone/Room Penetration Seals

a. Inspection Scope

The team reviewed the selected fire areas to evaluate the adequacy of the fire resistance of fire area barrier enclosure walls, ceilings, floors, fire barrier mechanical and electrical penetration seals, fire doors, and fire dampers. This was accomplished by observing the material condition and configuration of the installed fire barrier features, as well as construction details and supporting fire endurance tests for the installed fire barrier features, to verify the as-built configurations were qualified by appropriate fire endurance tests. The team also reviewed the fire hazards analysis to verify the fire loading used by the licensee to determine the fire resistive rating of the fire barrier enclosures. The team also reviewed the design specification for mechanical and electrical penetrations, fire flood and pressure seals, penetration seal database and Generic Letter (GL) 86-10 evaluations and the calculation for the technical basis of fire barrier penetration seals to verify that the fire barrier installations met licensing basis commitments.

The team reviewed fire barriers shown on the fire plan drawings for the selected fire areas. The team noted that MNS has eliminated selected fire barriers from the approved fire protection program and designated these fire barriers as "Sealed Firewall - Non Committed". These barriers are no longer included in any surveillance and testing program. Therefore, doors, dampers, fire proofing, etc. that exist in these declassified barriers are no longer included in any station surveillance procedures and effectively cannot be relied upon for the fire protection program. Two walls associated with Fire

Area 16/18 have been declassified. The wall between the Unit 2 switchgear room 2ETA (Fire Area 18) and the Unit 2 electrical penetration room (Fire Area 16) was declassified in Revision 9 (2000). The wall between the Unit 2 switchgear room 2ETA (Fire Area 18) and the Unit 2 HVAC equipment room 805A (Fire Area 18) was declassified in Rev. 3 (1982). For the purposes of the inspection of Fire Area 18, the electrical penetration room (Fire Area 16) was included in the inspection plan because the fire wall separating these areas has been declassified and is no longer a "Fire Sealed - NRC Committed" fire barrier. The similar wall at Unit 1 Room 803A was also declassified from a "Sealed Firewall - NRC Committed" to a "Sealed Firewall - Non Committed."

The team walked down the selected fire zones/areas to evaluate the adequacy of the fire resistance of barrier enclosure walls, ceilings, floors, and cable protection. The team selected several fire barrier features for detailed evaluation and inspection to verify proper installation and qualification. These features included fire barrier penetration fire stop seals, fire doors, fire dampers, and fire barrier partitions.

The team observed the material condition and configuration of the selected fire barrier features and also reviewed construction details and supporting fire endurance tests for the installed fire barrier features. This review was performed to verify that the observed fire barrier penetration seal configurations conformed with the design drawings and tested configurations. The team also compared the penetration seal ratings with the ratings of the barriers in which they were installed.

The team reviewed licensing documentation, engineering evaluations of GL 86-10 fire barrier features, and NFPA code deviations to verify that the fire barrier installations met design requirements and license commitments. In addition, the team reviewed surveillance and maintenance procedures for selected fire barrier features to verify the fire barriers were being adequately maintained.

b. Findings

No findings of significance were identified.

.10 Fire Protection Systems, Features, and Equipment

a. Inspection Scope

The team reviewed UFSAR Section 9.5.1, the fire protection design basis specification, fire protection code deviations, and administrative procedures used to prevent fires and control combustible hazards and ignition sources. This review was performed to verify that the objectives established by the NRC-approved FPP were satisfied. The team also toured the selected plant fire areas to observe the licensee's implementation of these procedures.

The team reviewed the adequacy of the design and installation of the automatic wet pipe sprinkler system protecting the RN pumps in Fire Area 4. Team members

performed a walk down of the system to ensure proper placement and spacing of the sprinkler heads and the extent of the sprinkler head obstructions. Selected engineering evaluations for NFPA code deviations were reviewed and compared with the physical configuration of the system. The team reviewed the sprinkler system hydraulic calculations for this system to ensure that the system could be supplied sufficient pressure and volume utilizing the two by-pass lines without opening the deluge valves. The team also inspected one of the by-pass lines located in an outside pit to determine the piping and fitting equivalent length to confirm the accuracy of the design input to the RN pump calculation. The team reviewed the fire protection code deviations calculation for automatic suppression systems relative to the selected fire areas.

The team reviewed the adequacy of the design and installation of the automatic detection and alarm system for the selected fire areas. This was accomplished by reviewing the ceiling reinforcing plans and beam schedule drawings to determine the location of ceiling bays. After the ceiling bay locations were identified, the team conducted a plant tour to confirm that each bay was protected by a fire detector in accordance with the Code of Record requirements - NFPA 72E, 1974. Field tours were conducted in fire areas 13, 16/18 to confirm detector locations. Minor modification package MM-12907 was reviewed where 10 new detectors were added to Fire Area 13 to conform the detection system to NFPA 72E location requirements.

The team reviewed the fire protection code deviations calculation for automatic detection systems relative to the selected areas to determine if there were any code deviations cited for the selected fire areas. The team reviewed the fire protection pre-plans and fire strategies to ensure that hose locations could sufficiently reach the selected fire areas for manual fire fighting efforts. Hose stations in the selected area were inspected to ensure that hose lengths depicted on the engineering documents were also the hose lengths located in the field. This was done to ensure that manual fire fighting efforts could be accomplished in the selected fire areas.

b. Findings

No findings of significance were identified.

4. Other Activities

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of licensee audits, self-assessments, and PIPs to verify that items related to fire protection and to SSD were appropriately entered into the licensee's corrective action program in accordance with the MNS quality assurance program and procedural requirements. The items selected were reviewed for classification, appropriateness, and timeliness of the corrective actions taken, or initiated, to resolve the issues. Included in this review were PIPs G-99-00110, M-99-

01884, M-99-01886, M-03-01675, and minor modification MM-12907 related to the McGuire Fire Protection Functional Audit SA-99-04(MC)(RA)(FPFA). In addition, the team reviewed the licensee's applicability evaluations and corrective actions for selected industry experience issues related to fire protection. The operating experience reports were reviewed to verify that the licensee's review and actions were appropriate.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.01 (Closed) URI 50-369,370/00-09-04: Adequacy of the Fire Rating of Mineral Insulated Cables in Lieu of Thermo-Lag Electrical Raceway Fire Barrier Systems

The NRC had opened this URI for further NRC review of the adequacy of the fire resistance rating of certain mineral insulated cables that the licensee had installed. The licensee had replaced an inadequate 3-hour Thermo-Lag fire barrier with mineral insulated cables for charging pump 1A in the Unit 1 Train B switchgear room. However, the adequacy of the testing of the mineral insulated cables, to assure their 3-hour fire resistance ability, had not been reviewed by the NRC.

The inspectors reviewed the NRC SER of January 13, 2003, on the licensee's use of mineral insulated cables and also reviewed the licensee's 10 CFR 50.59 safety evaluation for the modification. The NRC SER evaluated the licensee's installation and fire testing of the mineral insulated cables and concluded that the licensee had adequately demonstrated that the protection provided by the mineral insulated cables in the specific application was equivalent to the protection provided by a 3-hour rated fire barrier. The NRC SER further concluded that this change to the approved fire protection program did not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire and, therefore, did not require prior approval of the NRC. The inspectors concluded that the licensee's 50.59 safety evaluation for the change had adequately considered that the change did not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Consequently, the licensee's installation of mineral insulated cables was not a violation of NRC requirements. This URI is closed.

4OA6 Meetings

On May 23, 2003, the team presented the inspection results to you and other members of your staff, who acknowledged the findings. The team confirmed that proprietary information is not included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bailey, Mechanical and Civil Engineering (MCE) - Civil
J. Boyle, Training Manager
S. Bradshaw, Superintendent of Operations
H. Brandes, Consulting Engineer, General Office Fire Protection Program
J. Bryant, Regulatory Compliance Engineer
B. Dolan, Safety Assurance Manager
J. Hackney, Operations
T. Harrell, McGuire Station Manager
D. Henneke, Engineer, General Office Probabilistic and Risk Assessment Group
D. Herrick, Civil Engineering Supervisor
D. Jamil, Site Vice President, McGuire Nuclear Station
R. Johansen, Standby Shutdown Facility System Engineer
J. Lukowski, Reactor Electrical Systems (RES) - Power
E. Merritt, RES - Instrumentation and Controls
J. Oldham, Fire Protection Engineer, MCE - Civil
B. Peele, Station Engineering Manager
G. Peterson, Site Vice President, Catawba Nuclear Station
C. Thomas, Regulatory Compliance Manager

NRC Personnel

J. Brady, Senior Resident Inspector, Shearon Harris
E. DiPaolo, Resident Inspector
R. Fanner, Nuclear Safety Intern (Trainee)
C. Ogle, Chief, Engineering Branch 1, Division of Reactor Safety, Region II
R. Rodriguez, Nuclear Safety Intern (Trainee)
S. Shaeffer, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-369,370/03-07-01	URI	Fire Suppression System for Dedicated Shutdown Areas Not in Accordance with 10 CFR 50, Appendix R, Section III.G.3 (Section 1R05.02.b)
50-369,370/03-07-02	URI	Failure to Protect Redundant Trains of Reactor Protection System Cables From the Effects of Fire (Section 1R05.03.b.1)

50-369,370/03-07-03	URI	Requirements Relative to the Number of Spurious Operations that must be Postulated (Section 1R05.04.b.1)
50-369,370/03-07-04	URI	Methods for Reactor Coolant System Pressure Control During SSF Operation (Section 1R05.05.b)
50-370/03-07-05	URI	Failure to Provide Adequate Protection for Cables of Redundant Safe Shutdown Equipment in Fire Area 16/18 (Section 1R05.03.b.2)
50-370/03-07-06	URI	Spurious Closure of Valve 2CA0007A Could Lead to Damage of the TDAFW Pump (Section 1R05.04.b.2)

Closed

50-369,370/00-09-04	URI	Adequacy of the Fire Rating of Mineral Insulated Cables in Lieu of Thermo-Lag Electrical Raceway Fire Barrier Systems (Section 4OA5.01)

Discussed

None

APPENDIX

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection

Procedures

AP/0/A/5500/045, Plant Fire, Rev. 0 and Rev. 2
AP/1/A/5500/024, Loss of Plant Control Due to Fire or Sabotage, Rev. 21
AP/2/A/5500/024, Loss of Plant Control Due to Fire or Sabotage, Rev. 20
NSD 112, Fire Brigade Organization, Training, and Responsibilities, Rev. 5
NSD 313, Control of Combustible and Flammable Material, Rev. 4
NSD 314, Hot Work Authorization, Rev. 2
NSD 316, Fire Protection Impairment and Surveillance, Rev. 6
MP/0/A/7650/122, Inspection of Fire Hose and Hydrant Houses, Rev. 5
OP/0/A/6100/020, Operational Guidelines Following a Fire In Aux Bldg or Vital Area, Rev. 16
PT/0/A/4250/004, Fire Barrier Inspection, Rev. 19
PT/0/A/4250/011, Fire Door Inspections, Rev. 14
PT/0/A/4250/020, Roll-Up Fire Door Semi-Annual Inspection/Test, Rev. 2
PT/0/A/4400/001A, Fire Protection System Periodic Test, Rev. 24
PT/0/A/4400/001C, Fire Protection System Monthly Test, Rev. 54
PT/0/A/4400/001K, Fire Protection Annual Valve Test, Rev. 35
PT/0/A/4400/001M, Fire Protection System Flow Test, Rev. 14
PT/0/A/4400/008, Fire Hose Hydrostatic Test SLC-Committed Hose Stations, Rev. 11
PT/0/A/4400/010A, Main Fire Pump A, Rev. 15
PT/0/A/4400/010B, Main Fire Pump B, Rev. 10
PT/0/A/4400/010C, Main Fire Pump C, Rev. 11
PT/0/A/4400/017, Fire Pump A and B Operability Test, Rev. 13
PT/0/A/4400/018, Fire Pump C Operability Test, Rev. 11
PT/1/A/4400/001L, Fire Protection Containment Header Test, Rev. 9
PT/1/A/4400/001N, Halon 1301 System Periodic Test, Rev. 29
PT/2/A/4400/001L, Fire Protection Containment Header Test, Rev. 7
PT/0/A/4600/016A, Fire Detection System Operational Tests, Rev. 18
PT/0/B/4600/015, Fire Detection System Monthly Test, Rev. 14
PT/0/A/4700/049, SLC Fire Hose Inspection, Rev. 1
PT/1/A/4700/042, SLC Fire Hose Station Valve Operability Test, Rev. 3
PT/2/A/4700/043, SLC Fire Hose Station Valve Operability Test, Rev. 3
PT/1/A/4150/001B, Reactor Coolant Leakage Calculation, Rev. 47

Drawings

MC-1042-4, General Arrangement, Auxiliary Building, Elevation 750+0, Rev. 6
MC-1201-2-A, General Arrangement, Auxiliary Building, Elevation 716+0, Rev. 67
MC-1201-3-A, General Arrangement, Auxiliary Building, Elevation 716+0, Rev. 67

MC-1201-4, General Arrangement, Auxiliary Building, Elevation 733+0, Rev. 27
MC-1223-38, Auxiliary Building, Unit 1 & Unit 2, Beam Schedule at Elevation 733+0, Concrete and Reinforcing, Sheet 1, Rev. 4
MC-1223-39, Auxiliary Building, Unit 1 & Unit 2, Beam Schedule at Elevation 733+0, Concrete and Reinforcing Sheet 2, Rev. 6
MC-1223-6, Auxiliary Building, Unit 1, Plan at Elevation 733+0, Reinforcing Sheet 1, Rev. 8
MC-1223-7, Auxiliary Building, Unit 2, Plan at Elevation 733+0, Reinforcing Sheet 2, Rev. 5
MC-1223-8, Auxiliary Building, Unit 1, Plan at Elevation 733+0, Reinforcing Sheet 3, Rev. 6
MC-1223-9, Auxiliary Building, Unit 2, Plan at Elevation 733+0, Reinforcing Sheet 4, Rev. 6
MC-1223-27, Auxiliary Building, Units 1 & 2, Sections at Elevation 733+0, Concrete Sheet 3-1, Rev. 27
MC-1224-9, Auxiliary Building Unit 1, Plan at Elevation 750+0, Reinforcing Sheet 3, Rev. 9
MC-1224-10, Auxiliary Building Unit 1, Plan at Elevation 750+0, Reinforcing Sheet 4, Rev. 10
MC-1224-39, Auxiliary Building, Beam Schedule at Elevation 750+0, Concrete & Reinforcing Sheet 1, Rev. 6
MC-1225-10, Auxiliary Building Unit 2, Plan at Elevation 767+0, Reinforcing Sheet 4, Rev. 5
MC-1225-11, Auxiliary Building, Plan at Elevation 767+0, Reinforcing Sheet 5, Rev. 4
MC-1225-39, Auxiliary Building, Beam Schedule at Elevation 767+0, Concrete & Reinforcing, Rev. 6
MC-1225-40, Auxiliary Building, Beam Schedule at Elevation 767+0, Concrete & Reinforcing, Sheet 2, Rev. 5
MC-1226-8, Auxiliary Building, Plan at Elevation 784+0, Reinforcing Sheet 3, Rev. 1
MC-1226-9, Auxiliary Building, Plan at Elevation 784+0, Reinforcing Sheet 4, Rev. 2
MC-1226-19, Auxiliary Building, Beam Schedule at Elevation 784+0, Concrete and Reinforcing, Rev. 1
MC-1315-01.02-105, General Arrangement, Fire, Flood & HVAC Boundaries, Elevation 716+0, Rev. 0
MC-1384-06.02, Fire Protection Layout, Plan at Elevation 716+0, Rev. 7
MC-1384-06.03, Fire Protection Layout, Plan at Elevation 733+0, Rev. 7
MC-1384-06.04, Fire Protection Layout, Plan at Elevation 750+0, Rev. 7
MC-1384-06.05, Fire Protection Layout, Plan at Elevation 767+0, Rev. 7
MC-1384-07.12-00, Fire Plan, Auxiliary Building, Elevation 695+0, Rev. 3
MC-1384-07.01-00, Fire Plan, Unit 1 Turbine Building, Elevation 739+0, Rev. 11
MC-1384-07.13-00, Fire Plan, Auxiliary Building, Elevation 716+0, Rev. 12
MC-1384-07.13-01, Fire Plan, Auxiliary Building, Elevation 716+0, Rev. 9
MC-1384-07.14-00, Fire Plan, Auxiliary Building, Elevation 733+0, Rev. 12
MC-1384-07.14-01, Fire Plan, Auxiliary Building, Elevation 733+0, Rev. 9
MC-1384-07.14-02, Fire Plan, Auxiliary Building, Elevation 733+0 & 736+6, Rev. 9
MC-1384-07.14-03, Fire Plan, Auxiliary Building, Elevation 733+0 & 736+6, Rev. 9
MC-1384-07.15-00, Fire Plan, Auxiliary Building, Elevation 750+0, Rev. 10
MC-1384-07.15-01, Fire Plan, Auxiliary Building, Elevation 750+0, Rev. 2
MC-1384-07.15-01, Fire Plan, Auxiliary Building, Elevation 750+0, Rev. 3
MC-1384-07.15-01, Fire Plan, Auxiliary Building, Elevation 750+0, Rev. 9
MC-1384-07.15-02, Fire Plan, Auxiliary Building, Elevation 750+0, Rev. 10
MC-1384-07.16-00, Fire Plan, Auxiliary Building, Elevation 760+6, Rev. 7

MC-1384-07.17-00, Fire Plan, Auxiliary Building, Elevation 767+0, Rev. 10
MC-1384-07.17-01, Fire Plan, Auxiliary Building, Elevation 767+0, Rev. 9
MC-1384-07.18-01, Fire Plan, Auxiliary Building, Elevation 778+10, Rev. 8
MC-1518-06.43-00, Piping Layout, Interior Fire Protection, Nuclear Service Water Pumps, Sprinkler Addition, Rev. 1
MC-1518-06.43-01, Piping Layout, Interior Fire Protection, Component Cooling Pumps, Sprinkler Addition, Rev. 1
MC-1518-25.85-01, Piping Layout, Service Water Piping, Outside Pumphouse, Rev. 29
MC-1710-01.00, Plan, Control Room Computer Room, Elevation 767+0, Rev. 49
MC-1710-04.08, Battery Room Junction Points Elevation 747, Rev. 15
MC-1710-04.09, Battery Room Junction Points Elevation 746, Rev. 23
MC-1710-04.10, Battery Room Junction Points Elevation 745, Rev. 20
MC-1710-04.11, Battery Room Junction Points Elevation 744, Rev. 24
MC-1710-04.12, Battery Room Junction Points Elevation 743, Rev. 22
MC-1710-04.13, Battery Room Junction Points Elevation 742, Rev. 24
MC-1710-04.14, Battery Room Junction Points Elevation 741, Rev. 23
MC-1710-04.15, Battery Room Junction Points Elevation 740, Rev. 23
MC-1762-01.00-02, Location Diagram, Fire Detectors Located on Elevation 716+0, Rev. 7
MC-1762-01.00-03, Location Diagram, Fire Detectors Located on Elevations 733+0 & 739+0, Rev. 10
MC-1762-01.00-04, Location Diagram, Fire Detectors Located on Elevation 750+0, Rev. 10
MC-1762-01.00-06, Location Diagram, Fire Detectors Located on Elevations 760+6 & 767+0, Rev. 13
MC-2901-01.01, Auxiliary Building Plan Below Elevation 733'+0, Rev. 44
MC-2907-01.01, Penetration and Switchgear Rooms Plan Below Elevation 776'+0, Rev. 25
MCEE-138-00.02, Turbine Driven AFW Suction Supply Valve, Rev. 5
MCEE-138-00.04, Turbine-driven AFW Suction Supply Valve, Rev. 11
MCEE-138-00-01, Turbine Driven AFW Suction Supply Valve, Rev. 5
MCEE-211-00.52, Pressurizer Heaters, Rev. 2
MCEE-211-00.52-01, Pressurizer Heaters, Rev 9
MCEE-211-00.52-02, Pressurizer Heaters, Rev. 8
MCEE-211-00.52-03, Pressurizer Heaters, Rev. 9
MCEE-211-00.52-04, Pressurizer Heaters, Rev. 4
MCEE-211-00.52-05, Pressurizer Heaters, Rev. 3
MCEE-244-02.01, Steam Generator Level and Pressurizer Level, Rev. 4
MCEE-247-10.00, Motor Driven AFW Isolation Valve, Rev. 0
MCEE-247-20.00, Turbine Driven AFW Isolation Valve, Rev. 0
MCEE-247-20.01, Turbine Driven AFW Isolation Valve, Rev. 0
MCEE-247-32.00, Turbine-driven AFW Isolation Valve, Rev. 1
MCEE-247-33.00, Turbine Driven AFW Isolation Valve, Rev. 0A
MCEE-250-00.03, Pressurizer Power-operated Relief Valve
MCEE-250-00.03-01, Pressurizer Power-operated Relief Valve
MCEE-250-00.06, Pressurizer Power-operated Relief Valve Isolation Valve
MCEE-250-00.24, Unit 2 Chemical and Volume Control Isolation Valve, Rev. 01
MCEE-250-00.28, Reactor Vessel Head Vent Valves, Rev. 6

MCEE-250-00.29, Reactor Vessel Head Vent Valves, Rev. 5
MCEE-250-00.33, Reactor Vessel Head Vent Valves, Rev. 5
MCEE-257.00.54, Chemical and Volume Control Containment Isolation Valve, Rev. 3
MCEE-257-00.24, Chemical and Volume Control Containment Isolation Valve, Rev. 5
MCEE-257-00.50, Unit 2 Chemical and Volume Control Isolation Valve, Rev. 6
MCEE-257-00.52, Chemical and Volume Control Isolation Valve, Rev. 1
MCEE-257-00.55, Standby Makeup Pump, Rev. 1
MCFD-1574-01.00, Nuclear Service Water, Rev. 6
MCFD-1574-01.01, Nuclear Service Water, Rev. 10
MCFD-1599-01.00, P&ID, Flow Diagram of Fire Protection, Rev. 13
MCFD-1599-01.01, P&ID, Flow Diagram of Fire Protection, Rev. 14
MCFD-1599-02.00, P&ID, Flow Diagram of Fire Protection, Rev. 15
MCFD-1599-02.01, P&ID, Flow Diagram of Fire Protection, Rev. 15
MCFD-1599-02.02, P&ID, Flow Diagram of Fire Protection, Rev. 5
MCFD-1599-02.03, P&ID, Flow Diagram of Fire Protection, Rev. 6
MCFD-1599-03.00, P&ID, Flow Diagram of Fire Protection, Rev. 7
MCFD-1599-03.01, P&ID, Flow Diagram of Fire Protection, Rev. 3
MCFD-2574-02.00, Nuclear Service Water, Rev. 12
MCFD-2574-02.01, Nuclear Service Water, Rev. 2
MCFD-2592-01.01, Auxiliary Feedwater System, Rev. 13
MCFD-2592-02.00, Auxiliary Feedwater System, Rev. 2
MCM.1206.07-0074.001, McNeary Insurance Consulting Services, FP-12
MCM.1206.07-0087.001, McNeary Insurance Consulting Services, FP-18

Completed Maintenance And Surveillance Test Procedures/Records

Work Order 98410020, PT 2NCLP5151, SSF Pressurizer Level, dated 3/13/02
Work Order 98410021, PT 2NCLP5121 NC Loop D Hot Leg W/R Pressure, dated 3/13/02
Work Order 98410083, PM 2CFLP6110, S/G D W/R Level, dated 2/28/02
Work Order 98410084, PM 2CFLP6100, S/G C W/R Level, dated 3/5/02
Work Order 98410085, PM 2CFLP6090, S/G B W/R Level, dated 3/1/02
Work Order 98410086, PM 2CFLP6080, S/G A W/R Level, dated 2/28/02

Cable Installation Data for the Following Components

2CA0007A
2CA009B
2CFLT6080, 6090, 6100, 6110
2NC272AC, 273AC
2NC33A, 35B
2NCLT5151
2NV1012C
2NV842AC
2NV94AC
2NVPU0046

ORN4AC

Calculations and Evaluations

MCC-1223.04-00-0010, Determine the Reactor Coolant Pump Sealwater Flow Requirements for the SSF Auxiliary Makeup Pump, Type II
MCC-1223.42-00-0030, Documentation of the Adequacy of the Assured Suction Sources to the CA Pumps, Rev. 8

MCC-1223.49-00-0030, Sprinkler System for Nuclear Service Water Pumps @ Elevation 716-0, Rev. 0
MCC-1435.00-00-0006, Calculation for the Technical Basis of Fire Barrier Penetration Seals, Rev. 1
MCC-1435.03-00-0002, Fire Exposure to Unprotected Steel Hangers for HVAC Ducts, Rev. 2
MCC-1435.03-00-0004, Supports for Cable Tray Penetrating Fire Barriers, Rev. 0
MCC-1435.03-00-0012, MNS Penetration Seal Database and GL 86-10 Evaluations, Rev. 0
MCC-1435.03-00-0013, Fire Protection Code Deviations, Rev. 0
MCS-1435.00-00-0001, Fire Protection Acceptance Specification, Rev. 17
MCS-1435.00.00-0003, Design Specification for Mechanical and Electrical Penetrations; Fire Flood and Pressure Seals
National Fire Codes - Volume 1, Codes & Standards: NFPA 13 - Standard for the Installation of Sprinkler Systems, 1978 Edition

Design Basis Document

MCS-1223.SS-00-0001, Design Basis Specification for the Standby Shutdown System, Rev. 12
MCS-1465.00-00-0008, Design Basis Specification for Fire Protection, Rev. 4.
MCS-1465.00-00-0022, Design Basis Specification for Appendix R, Rev. 2

Problem Investigation Process Reports Reviewed

G-99-00110, McGuire Fire Protection Functional Audit (SITA) SA-99-04(MC)(RA)(FPFA).
M-97-03311, All three CA pumps may have been dead headed during the U1 Rx trip recovery.
M-99-01884, GL 86-10 guidance for circuit failure modes, hot short duration, and design basis transients for dedicated shutdown not evaluated for applicability to MNS methodology.
M-99-01886, NFPA code deviations not documented in UFSAR or FHA as per GL 86-10.
M-99-03926, Effect of warmer seal injection water on RCP seals during SSF event not adequately taken into consideration on SMP capacity. Evaluate applicability to McGuire.
M-00-01900, Unit 1 CA pumps normal suction sources inadvertently isolated following a reactor trip and automatically aligned to RN.
M-00-04466, Evaluate UFSAR Section 9.5-1 Clarifications for Fire Suppression Systems.
M-00-04469, Evaluate Fire Pump Loss Due to Fire in Fire Area 19 and Main Control Room.
M-00-04483, The fire protection RY by-pass lines around 1RY 113 and 1RY 114 do not Permit the Maximum Flow for the Largest Sprinkler Demand.
M-00-04487, Fire Brigade Drills Had Not Been Performed Within 10 Years in Areas Considered

Safety Significant.
M-00-04491, NRC Appendix R inspection in certain fire areas determined the potential for NC PORV and block valve actuation. We need to evaluate this cabling as to "if" this will occur.
M-00-04516, Adequacy of Pzr heater capacity at SSF due to increase safety valve leakage.
M-02-01708, It has been discovered that pressurizer ambient heat losses are greater than calculated in OSC-3144 impacting SSF ASW system operability (TS 3.10.1 and TS 3.4.9).
M-02-03214, SSS and NC DBDs identified errors related to pressurizer heater requirements.
M-02-05031, RO closed 1CA-0002, resulted in temp low suction flow to running 1B CA pump.
M-02-05096, Information on system problem [PIP M-02-05031] not documented for resolution.
M-03-01675, Fire Detection System Not Installed to NFPA Codes.
M-03-01748, Smoldering fire on roof of Unit 1 Diesel Generator building.

Prblem Investigation Process Reports Generated During This Inspection

M-03-02084, Fire scenarios that could cause suction loss to U2 TDCA pump for SSF areas.
M-03-02086, Discrepancy between Appendix R DBD and Procedure AP/2/A/5500/24.
M-03-02091, Unit 1 and Unit 2 HVAC areas do not have fire detectors.
M-03-02092, Discrepancy between drawings and fire pre-plans for fire hose lengths.
M-03-02093, Drawing discrepancy for as-built configuration of HVAC Equipment Room 805A.
M-03-02106, B train cables in A SWGR room Fire Area which are not previously identified.
M-03-02115, Appendix R logic diagrams not updated to show function of valve 2CA002.
M-03-02118, Appendix R logics for AFW do not show valve 2CA0007A.
M-03-02249, Detector zones 203 and 204 not in SLC 16.9.6, Table 16.9.6-1.
M-03-02275, Calculation (MCC 1223.48-00-0030) in support of sprinkler system design over the
nuclear service water pumps needs revising.
M-03-02294, SLC Table 16.9.7-1 appears to be missing some information.
M-03-02311, Evaluate May 2003 NRC Fire Protection Inspection items.
M-03-02327, Calc MCC-1435.03-00-0002 contains deleted pages not marked as being deleted.
M-03-02588, Apparent Appendix R violation in the 1ETA and 2ETA switchgear HVAC rooms.

Miscellaneous

MNS Units 1 and 2 Safety Evaluation Report (SER), March 1978
SER Supplement 2 (SSER 2), Appendix D, Fire Protection Review, Units 1 & 2, March 1979
SSER 5, Appendix B, McGuire SER, Fire Protection Review, Unit 1 & 2 (Revised), April 1981
SSER6, Appendix C, McGuire SER - Standby Shutdown System, February 1983
MNS Updated Final Safety Analysis Report (UFSAR) Section 9.5.1, Fire Protection System
UFSAR Section 16.9.7, Selected Licensee Commitments (SLC), Standby Shutdown System
Letter from W.O. Parker, Duke Power Co., to H.R. Denton, NRC, McGuire Nuclear Station Fire Protection, dated January 9, 1981
Letter from D.S. Hood, NRC, to H. B. Tucker, Duke Power Co., Fire Protection Deviations, McGuire Nuclear Station, Units 1 and 2, dated May 15, 1989
Fire Area Ventilation Rates, Fire Areas 4, 13, 18 & 24
Fire Area Oil Quantities, Fire Area 4, 13, 18 & 24

Fire Area 4 Correlation List between Rooms Number vs. Detection Zones
Fire Qualification Test on Silicone Foam Floor Pen Seals, Slab No. 5, Project No. 03-5656-001

Applicable Codes and Standards

NFPA 13, Standard for the Installation of Sprinkler Systems, 1978 Edition
NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 1976 Edition
NFPA 72E, Standard on Automatic Fire Detectors, 1974 Edition

Modifications

Minor Modification MM-12907A thru F

LIST OF ACRONYMS

AB	Auxiliary Building
AFW	Auxiliary Feedwater
AP	Abnormal Procedure
DSD	Dedicated Shutdown
FHA	Fire Hazards Analysis
FPP	Fire Protection Review
GL	Generic Letter
HVAC	Heating Ventilation and Air Conditioning
IPEEE	Individual Plant Examination for External Events
IR	Inspection Report
kW	Kilowatt
MCR	Main Control Room
MNS	McGuire Nuclear Station
NC	Reactor Coolant
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NSD	Nuclear System Directive
NV	Chemical and Volume Control
PIP	Problem Investigation Process
PORV	Power Operated Relief Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RN	Nuclear Service Water
RPS	Reactor Protection System
SDP	Significance Determination Process
SER	Safety Evaluation Report
SG	Steam Generator
SLC	Selected Licensee Commitment
SMP	Standby Makeup Pump
SSA	Safe Shutdown Analysis
SSD	Safe Shutdown
SSF	Standby Shutdown Facility
SSS	Standby Shutdown System
TDAFW	Turbine-Driven Auxiliary Feedwater
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volt