From: Charles R. Ogle
To: Fillion, Paul; Thomas, McKenzie
Date: 7/8/03 11:06AM
Subject: MY COMMENTS ON MCG IR 03-07

Overall, I thought this was a good report. Attached are some of my comments. I've also attached a comparison version so that you can see what changes were made between what you gave Charlie and what I signed. (Needs to be opened in WP to see the changes.)

WHile the attacment identifies areas for improvement on the report, it does not diminish the fact that I think that the inspection and issue raised were outstanding.

CC: Payne, Charlie

H/32

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

1

July ~~XX~~3, 2003

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

SUBJECT: MCGUIRE NUCLEAR STATION - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 50-369/03-07 AND 50-370/03-07

Dear Mr. ~~Jamil~~Peterson:

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~~An interim exit was held with Mr. D. Jamil and other members of your staff on May 22, 2003, to discuss the results of that effort. Following completion of additional review in the Region II office, a final exit was held with you and other members of your staff on July 2, 2003. The enclosed report documents our findings from this inspection.

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

This report documents ~~two~~three findings that~~, combined,~~ have potential safety significance greater than very low significance, however, a safety significance determination has not been completed. ~~One~~These findings did not present an immediate safety concern ~~and~~at the time of the interim exit. However, your subsequent analyses of one of the findings associated with Fire Area 16/18 resulted in identification of additional cables associated with reactor protection system instrumentation (and possibly other equipment) required for safe shutdown located in the same fire area that could be susceptible to fire damage. Upon discovery of this condition on June 10, 2003, a fire watch was ~~put in place on June 10, 2003,~~established as a compensatory measure.

~~In addition, the report documents one NRC-identified finding which was determined to involve a violation of NRC requirements. However, the significance of this finding has not been determined. Also, one licensee-identified violation is listed in this report.~~ 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-

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: Inspection Report 50-369, 370/03-07
w/Attachment: Supplemental Information

cc w/encl:
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DEC 4

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OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:Consultant	RII:DRS	RII:DRP
SIGNATURE	RA	RA	RA	RA	RA	RA	RA

DEC 5

NAME	MThomas	PFillion	RMaxey	RSchin	BMelly	CPayne	RHaag
DATE	~~7/~~	~~7/~~	~~7/~~	~~7/~~	~~7/~~	~~7/~~	7/3/2003
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
PUBLIC DOCUMENT	YES NO						

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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.:	50-369/03-07 and 50-370/03-07
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station~~, Units 1 and 2~~
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; ~~5/9 – 23/2003~~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~~ NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- TBD~~.~~ The team identified a violation ~~involving~~in that Train A and Train B cables associated with ~~the~~redundant reactor protection system instrumentation (and possibly other equipment) important to safe shutdown were located in the same fire area (Fire Area 16/18) 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. Theis 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 ~~would~~could be lost.~~ The finding represented an operability concern, which the licensee resolved by posting a fire watch in the area.~~ 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 ~~Fire Protection Program (i.e.,~~ 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 ~~close~~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 ~~spurious closure of the valve could damage~~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. This finding may have potential safety

significance greater than very low significance because the standby shutdown system relies on the TDAFW pump for decay heat removal, and ~~seriously degrade~~ the decay heat removal function would be seriously degraded if the TDAFW pump were damaged due to closure of valve 2CA0007A. (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 ~~Switchgear Room/Electrical~~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 ~~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 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~~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 ~~and because it affects the mitigating systems cornerstone objective~~. 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~~
~~.01~~ ~~Systems Required to Achieve and Maintain Post-Fire Safe Shutdown~~
~~a.~~ ~~Inspection Scope~~
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 ~~fire protection program (~~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 ~~(BTP)~~ Auxiliary and Power Conversion Systems Branch ~~(APCSB)~~ 9.5-1, Guideline for Fire Protection for Nuclear Power Plants; related NRC Safety Evaluation Reports (SERs); ~~McGuire Nuclear Station (~~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.
~~The team reviewed 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 Common Area – a fire in this area would involve alternative shutdown from the standby shutdown facility (SSF) using the standby shutdown system (SSS)~~

- ~~**Fire Area 13:** Battery Rooms Common Area – a fire in this area would involve alternative shutdown from the SSF using the SSS~~

- ~~**Fire Area 16/18:** Unit 2 Train A 4160 Volt Switchgear Room/Electrical Penetration Room – a fire in this area would involve shutdown from the main control room using Train B equipment~~

- ~~**Fire Area 24:** Main Control Room (MCR) – a fire in this area would involve alternative shutdown from the SSF using the SSS~~

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

a. Inspection Scope

The team reviewed the licensee's FPP ~~documented~~described in UFSAR Section 9.5.1; the MNS Fire Protection Review; safe shutdown analysis (SSA); fire hazards analysis (FHA); ~~safe shutdown (SSD)~~ essential equipment list; and system flow diagrams to identify the components and systems necessary to achieve and maintain ~~safe shutdown~~SSD conditions. ~~Specific licensee documents, calculations, and drawings reviewed during this inspection are listed in Attachment 1. The objective of this evaluation was to assure the SSD equipment and post-fire SSD analytical approach were consistent with and satisfied the Appendix R reactor performance criteria for SSD.~~ 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. ~~S~~The following Unit 2 systems ~~and/or~~and components were selected for review ~~included~~: ~~s~~

- Standby shutdown system (SSS)~~; Unit 2 standby~~
- Standby makeup pump (SMP) 2NVPU0046 ~~and~~
- SMP suction supply ~~motor operated valve (MOV)~~ 2NV842AC~~; auxiliary~~
- Auxiliary feedwater (AFW) suction supply ~~MOVs~~valves 2CA007A~~,~~ and 2CA009B~~, 2CA161C, and 2CA162C; reactor~~
- Reactor coolant pump (RCP) seal water return isolation valve 2NV94AC~~; pressurizer~~
- Pressurizer power operated relief valve (PORV) 2NC34A ~~and~~
- PORV isolation valves 2NC33A~~; Unit 2 pressurizer~~
- Pressurizer heaters Nos. 28, 55, ~~and 56; reactor~~
- Reactor vessel head vent valves 2NC272AC and 2NC273AC~~; and heating~~
- 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~~, 16,~~ and ~~18~~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 ~~the~~ 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. ~~Alternative~~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. ~~T~~However, 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~~designed to protect the RN pumps and the area 20 feet north of these pumps. The area protected by this sprinkler system was located between ~~C~~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~~ for alternative and dedicated shutdown~~.

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 (URI 50-369/84-28-01, 370/84-25-01). 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 ~~reviewed and~~ 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 as required by 10 CFR 50, Appendix R, Section III.G.3. During ~~this~~the current inspection, the team questioned whether the previous NRC 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 ~~will~~would be reviewed~~ further~~ to determine if the lack of sprinkler coverage in this fire area has an impact on risk.~~ This issue is identified as URI 50-369,370/03-07-01, Fire Suppression System for Alternative Shutdown Areas not in Accordance with 10 CFR 50, Appendix R, Section III.G.3.~~ The team noted that a similar ~~conditions, regarding the fixed fire suppression system complying with 10 CFR 50, Appendix R, Section III.G.3, was applicable to other MNS~~condition exists in other fire areas where ~~alternative~~dedicated shutdown capability using the SSS was designated by the licensee~~ (examples include Fire Areas 14 and 21). This issue is unresolved pending further NRC review using risk insights to determine if a 10 CFR 50.109 (backfit)~~. 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/zones~~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 ~~for~~of the ~~charging/safety injection~~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.~~ Walkdowns were performed to compare cables indicated on the drawings with actual plant installation.~~ Circuit and cable routings were reviewed for the following equipment:-

- 0RN4AC, ~~Turbine-driven~~Turbine Driven AFW Suction Supply Valve
- 2CA0007A, ~~Turbine-driven~~Turbine Driven AFW Suction Isolation Valve
- 2CA009B, ~~Motor-driven~~Motor Driven AFW Suction Isolation Valve
- 2CFLT6080, 6090, 6100, 6110, Steam Generator Level Transmitters
- 2NCLT5151, Pressurizer Level Transmitter

- 2NC34A, ~~33A,~~ Pressurizer PORV ~~and~~
- 2NC33A, PORV Isolation Valve
- 2NC272AC, 273AC, Reactor Vessel Head Vent Valves
- 2NVPU0046, Standby Makeup Pump ~~(SMP)~~
- 2NV94AC, RCP Seal Water Return Isolation Valve
- 2NV842AC, SMP Suction Isolation Valve
- 2NV1012C, SMP Discharge to Containment Sump Isolation Valve
- Pressurizer ~~H~~heaters No~~s~~. 28, 55, 56

The team also reviewed licensee studies of overcurrent protection ~~on both~~for alternating current ~~(AC)~~ and direct current ~~(DC)~~ systems to identify whether fire-induced faults could result in defeating the ~~safe shutdown~~SSD functions.

b. Findings

Findings associated with valves 2CA0007A, 2NC34A, and 2NC33A are discussed in Section ~~1R05~~.04 of this ~~inspection report.~~
~~1.~~ ~~Reactor Protection System~~IR.

1. Inadequate Separation of Cables Associated With Safe Shutdown Instrumentation

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

Description: Fire Area 16/18 is the Unit 2 Train A electrical penetration room/2ETA 4160 volt (V) switchgear room~~,~~ and the associated HVAC equipment room 805A. Train B equipment controlled from the ~~main control~~MCR room was ~~intended~~ designated as t~~o be used~~he SSD train for a fire in this area according to the ~~analysis~~SSA and plant procedures (i.e., this fire area complies with 10 CFR 50, Appendix R, Section III.G.2). During a walkdown of Fire Area 16/18, the team identified that room 805A ~~in Fire Area 16/18~~ lacked fire detection and fire suppression. Room 805A is the HVAC equipment room ~~providing~~which supplies ventilation to the Unit 2 Train A 4160V ~~Switchgear Room 2ETA. This area has a moderate to high fire loading consisting principally of cables. The team identified that a similar condition also existed for room 803A, which is the HVAC equipment room providing ventilation for the Unit 1 Train A 4160V Switchgear Room 1ETA in Fire Area 17~~switchgear room 2ETA. The team also observed that Train B cables were routed ~~in this~~through room 805A. Many of the identified cables were in ~~a~~ cable trays near the ceiling and were going from/to the cable spread room, which ~~is~~was on the same elevation~~,~~; and to/from the control room, which was above ~~the~~room 805A. The licensee was not aware that 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 supplying ventilation for the Unit 1 Train A 4160V switchgear room~~. The licensee had not been aware of all of these "opposite train" cables, and they initiated PIP M-03-02106~~ 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).

~~As many as 74 "opposite train" cables are involved related to the reactor protection system.~~ The licensee subsequently 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 in room 805A 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 ~~would~~to spuriously go to the trip condition. ~~Subsequently~~Consequently, a safety injection signal ~~would~~could be generated~~ due to spurious "high containment pressure."~~ AThe safety injection signal ~~would~~could in turn trigger a reactor trip and Phase A isolation. [At the same time, many ~~important~~ main control panel instruments ~~would~~necessary to achieve and maintain hot shutdown could be lost~~. For example,~~, including pressurizer level and all four steam generator ~~level, which are instruments necessary to achieve and maintain hot shutdown.~~(SG) level instruments.] The licensee also stated that ~~a~~ similar ~~situation exists~~effects could occur for a fire in the Unit 1 Train A switchgear room 1ETA (Fire Area 17).

<u>Analysis</u>: The ~~fact that~~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 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 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.
<u>Enforcement</u>: The licensee's FPP commits to 10 CFR 50, Appendix R, Section III.G. Section III.G.2 requires in part, that cables or equipment for one of the redundant trains of a system necessary to achieve and maintain hot shutdown (located in the same fire area) shall be ensured to be free of fire damage by one of the following: (1) separated by a 3-hour rated fire barrier; (2) separated by 20-feet or more horizontal distance with no intervening combustibles or fire hazards, and having suppression and detection; or (3) enclosure of the cables in a 1-hour rated fire barrier and having suppression and detection.

Contrary to the above, electrical cables associated with redundant trains of RPS instrumentation necessary to achieve and maintain hot shutdown could be ~~lost due to a credible fire in one area as described above constitutes a violation of 10 CFR 50, Appendix R, Section III.G.2. This section requires that one train of systems necessary to achieve and maintain hot shutdown shall be free of fire damage. The fact that the area presented an exposure fire hazard to safe shutdown equipment and did not have automatic fire detection systems represents a violation of 10 CFR 50, Appendix R,~~

~~Section III.F. 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 present an operability concern, which the licensee resolved by posting a fire watch in the area of concern. Once the licensee has fully analyzed the manner in which plant systems would have been affected by damage to the "opposite train" cables and reviewed the abnormal operating procedures in light of the degraded instrumentation and any automatic actions that would be initiated, the NRC will review this analysis. Once the equipment degradations and relevant procedures are understood, a significance determination process (SDP) will be performed to determine the level of significance. When assessed in combination with the finding related to inadequate protection of 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.~~

~~Enforcement: As described above, the finding is a violation of Appendix R requirements of greater than minor significance~~damaged during a fire in room 805A (Fire Area 16/18). Pending determination of the systems affected and the safety significance, the finding is identified as URI 50-369, 370/03-07-02, ~~Failure to Protect Reactor Protection System Cables Results in Loss of Required Shutdown~~Inadequate Separation and Protection of Cables Associated With Redundant Trains of Instrumentation Located in the Same Fire Area.

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

Introduction: A finding was identified in that physical protection of the ~~associated~~ electrical cables ~~for~~associated with valve 2CA0042B (2B motor driven AFW pump discharge supply to ~~steam generator~~SG 2D) did not meet the requirements of 10 CFR 50, Appendix R, Section III.G.2. Instead, the licensee ~~substituted the use of~~d 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: ~~On April 2, 2003, t~~ The licensee identified (April 2003) that MNS relied on local, manual operator actions outside the MCR for SSD in ~~non-alternative~~non-dedicated shutdown fire areas (i.e., areas designated as complying with 10 CFR 50, Appendix R, Section III.G.2)~~ and the~~. These local, manual operator actions did not have prior NRC approval. The licensee documented this issue in PIP ~~M-03-02311~~M-03-02311. The team reviewed the local, manual operator actions 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 ~~2ETA/E~~electrical ~~P~~penetration ~~R~~room (Fire Area 16/18) without adequate spatial separation or fire barriers. ~~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 steam generator instead of the two required for SSD. However, r~~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~~s~~ 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. ~~These~~This local, manual operator action~~s~~ had not received NRC approval.

Analysis: The team determined that this finding was associated with the ~~"equipment~~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 ~~reactor protection system~~RPS cable separation finding (~~also discussed in this inspection report~~Section .03.b.1), this finding may have potential safety significance greater than very low significance.

Enforcement: The licensee's ~~Fire Protection Program~~FPP commits to 10 CFR 50, Appendix R, Section III.G. Section III.G.2 ~~states~~requires 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~~ for one of the redundant trains of a system~~s~~ necessary to achieve and maintain hot shutdown ~~conditions are~~ (located ~~within~~in the same fire area ~~outside~~) shall be ensured to be free of ~~primary containment,~~fire damage by 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~~: (1) separated by a 3-hour rated fire barrier; (2) separated by 20-feet or more 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;~~, and having suppression and detection; or (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."~~ the cables in a 1-hour rated fire barrier and having suppression and detection.

Contrary to the above, on May 23, 2003, the ~~team found that the~~ licensee failed to protect electrical cables ~~of~~associated with redundant equipment located within the Unit 2 Train A ~~Switchgear Room/Electrical P~~electrical penetration ~~R~~room (Fire Area 16/18) with an adequate barrier or to provide 20 feet of separation. Instead, the licensee used a local manual operator action, which had not received prior NRC approval, to achieve and maintain SSD. Pending determination of the finding's safety significance, this finding is identified as URI 50-370/03-07-053, ~~Failure to Provide Adequate~~Use of a Local Manual Operator Action in Lieu of Providing Physical 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, ~~abnormal procedures~~APs for ~~alternative shutdown (ASD)~~DSD, and the licensee's Appendix R ~~manual action requirements analyses~~fire area failure analysis and compliance strategy for a fire in~~-the selected~~ 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 ~~1R05~~.03.a. of this ~~inspection report~~IR were also reviewed in relation to ~~alternative post-fire safe shutdown~~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 downs ~~focused on ensuring that the~~ DSD procedures to determine if they could ~~reasonably~~ be performed within the required times~~,~~ given the minimum required staffing level of operators ~~and~~, 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. ~~The objective of these reviews was to assure that the post-fire safe shutdown analytical approach, safe shutdown equipment, and procedures were consistent and complied with the Appendix R reactor performance criteria for safe shutdown.~~

b. Findings

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

Introduction: ~~An unresolved item was~~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~~. Resolution~~ during SSD analysis of ~~the unresolved item is~~a fire area. This issue is a URI pending review ~~by~~of NRC ~~staff~~guidance in this area.

Description: The licensee's ~~fire protection analysis~~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 ~~PORV/PORV~~PORVs and PORV isolation valves on the pressurizer of each unit. Should operators in the control room become aware of a fire in any ~~area of the~~ plant ~~through~~area (from a fire alarm or the plant communications system), they would respond by ~~following the instructions in abnormal p~~implementing Procedure AP/0/A/5500/045, Plant Fire. Depending on the fire location, ~~p~~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 the licensee's assumption 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. ~~Then w~~With the ~~block~~isolation valve closed, it would then take two spurious operations to breach the RCS pressure boundary (i.e., ~~namely one block~~the isolation valve opening and its associated PORV also opening).- Theis concept of

postulating only one spurious operation ~~need be postulated~~ meant that closing the ~~block~~isolation valve was sufficient ~~in itself~~ to ensure ~~the desired result~~RCS pressure boundary integrity. The licensee considered that there was no need to take any other action such as de-energizing the isolation valves after ~~they were~~it was closed. ~~T~~Application of this concept ~~was~~is not ~~necessarily~~ consistent with ~~NRC~~NRC's cable protection requirements ~~for protection of cables~~of Appendix R, Section III.G.

The team reviewed the control circuits and cable routing information for ~~valve 2NC34A,~~ pressurizer PORV 2NC34A, and ~~2NC33A~~ its associated isolation valve 2NC33A. They observed that cables for both the PORV and isolation valve ~~are~~were routed ~~in~~through Fire Areas 13, 16/18 and 24. ~~When the control circuit for the PORV is analyzed and considering that the cables are armored type cables (except in the control room) one can conclude~~The team determined that, for these three fire areas, spurious opening of the PORV could only occur for ~~the fire in Fire Area 24, the control room. Considering this information, the team postulated the following scenario. A fire starts in the control room. Operators close the isolation valves per procedure AP/0/A/5500/045 within ten minutes. Later, isolation valve 2NC33A spuriously opens due to a fire induced short-circuit. Operators take no action to counter the spurious opening of the isolation valve because they have no information that it occurred. Subsequently PORV 2NC34A spuriously opens due to a fire induced short-circuit. At this point, it would be possible to close the PORV by opening the appropriate circuit breaker at the 125 VDC distribution panel. This would take time, and it is not covered by the fire response procedure. Before the PORV can be re-closed, the fire has progressed and the decision is made to abandon the control room and shutdown using the SSS. The PORV would now be closed by operating the control room/SSS transfer switch as directed by abnormal procedure AP/2/A/5500/024, Loss of Plant Control Due to Fire or Sabotage. The situation now is that the PORV/isolation valves were opened for a period of time and the RCS is may not be at normal level and pressure. The standby makeup pump has relatively low capacity and may not have the capacity to maintain hot shutdown in this scenario, and RCS variable parameters may be outside the requirements of Appendix R, i.e. outside the range predicted for a loss of offsite power. For example, an open PORV following a reactor trip could result in pressurizer level lower than that predicted for a trip caused by a loss of offsite power.~~

~~Analysis: The team was not certain whether the licensee's analysis of circuits for spurious operation was consistent with the requirements for independence of cables, systems or components in the area under consideration as stipulated by Appendix R, III.G.3 and III.L. In the example of the PORVs described above, if~~a MCR fire (Fire Area 24). If more than one spurious operation ~~would~~were to occur, the dedicated shutdown capability (i.e., the SSS) would not be independent from the ~~control room~~MCR in that, during a fire in the ~~control room~~MCR, pressurizer level may not remain within the indicating range which could result in conditions outside ~~of~~ those specified in ~~III.L~~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 ~~must be considered then there would be a violation of Appendix R~~

~~requirements having more than minor significance. The equipment reliability objective of the cornerstones of mitigating systems and barrier integrity could be affected.~~
were to occur, the dedicated shutdown capability (i.e., the SSS) would not be independent from the MCR in that a fire in the MCR could result in pressurizer level not remaining within the indicating range.

Enforcement: In the case of the ~~PORV/isolation~~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~~ depending on whether more than one spurious operation must be postulated~~. Review of this matter by the NRC will determine whether a violation has occurred. ~~If a violation has occurred, the significance will be determined. T~~ Pending review of NRC guidance in this area, the issue is identified as URI ~~50-369~~50-369/03-07-03, 370/03-07-0~~3~~4, Requirements Relative to the Number of Spurious Operations ~~t~~That ~~m~~Must be Postulated.

~~2.~~ ~~Valve 2CA0007A~~ 2. Auxiliary Feedwater Valve 2CA0007A not Included in Safe Shutdown Analysis

Introduction: A finding ~~of~~with potentially greater than very low safety significance was identified in that ~~a valve in the auxiliary feedwater system~~AFW suction supply valve 2CA0007A, which could spuriously operate during a MCR fire, was not included in the ~~safe shutdown analysis and it could spuriously close due to a fire in the main control room.~~SSA. Spurious closure of this valve could damage the turbine driven auxiliary feedwater (TDAFW) pump, thus seriously degrading the ~~core residual~~secondary decay heat removal function of the ~~safe shutdown system~~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 ~~auxiliary feedwater~~AFW storage tank to the ~~turbine driven auxiliary feedwater~~TDAFW pump. The valve is open during normal plant operation. Valve 2CA0007A is important to safe shutdown for fire areas where the ~~safe shutdown system (SSS)~~ will be used~~. The importance is derived from fact that~~ because the SSS ~~uses~~relies on the TDAFW pump for secondary decay heat removal~~ and potential for spurious closure of the valve. The team found that the safe shutdown analysis for Unit 2 did not recognize valve 2CA0007A. It was not listed in Appendix E, list of important equipment, nor Appendix F, list of potential problem cables.~~

~~One scenario could be a fire starts in the control room which leads to a plant trip and loss of offsite power. In this case, the TDAFW pump would receive an automatic start from the "LOOP on safety-related bus" logic or possibly "low steam generator level" due to loss of the feedwater pump. Even though the safe shutdown analysis for a fire in the control room ultimately relies on the SSS, operators may remain in the control room if they believe the plant is still under control. The TDAFW pump could be running and taking suction from the auxiliary feedwater storage tank with flow through 2CA0007A. Since control wires to the open/close control switch for this valve run in the control room (in single-conductor plug cable, bundled in groups of approximately 30 wires), the valve could spuriously close due to fire-induced short-circuit between two of the wires.~~
Spurious closure of the valve would immediately reduce suction pressure and quickly shut off all normal AFW flow through the pump. ~~Assuming that the TDAFW pump is~~

~~damaged by spurious closure of 2CA0007A and if plant conditions deteriorated due to progressing fire in the control room forcing~~Closure of this valve could cause severe damage to the pump if automatic transfer to the alternate suction sources does not initiate within sufficient time. 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~~.~~

~~Besides the control room, there are open/close switches for this valve at auxiliary feedwater panel 2A and the auxiliary feedwater turbine control panel (2AFPT). Cable 2*CA517 runs between area terminal cabinet 2ATC2 and the auxiliary feedwater panel 2A, and it runs through fire area FA-4. Cable 2*CA519 runs between area terminal cabinet 2ATC2 and panel 2AFPT, and it runs through fire area FA-4. Cable 2*CA557 contains power and control for the valve, and represents a potential for spurious operation of the valve. Therefore a fire in FA-4 could also result in spurious closure of~~ if the TDAFW pump were damaged. The team found that the SSA did not include valve 2CA0007A. ~~This could lead to problems similar to that described above for the control room fire. It is not expected that a fire in FA-4 would lead to a loss of offsite power. However, a problem scenario could be as follows: If the fire becomes severe and the decision is made to use the SSS, procedures direct the operator to trip the normal feedwater pump. This could cause low steam generator level which in turn will auto-start the TDAFW pump. If 2CA0007A has already spuriously closed, the pump has no through flow upon starting~~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 (MCS-1465.00-00-0022, Design Basis Specification for Appendix R).

The licensee initiated ~~a corrective action document~~PIPs —03-02084, M-03-02118, and M-03-02311 for this issue~~, PIP M-03-02084,~~ and ~~they~~ took prompt action to ~~restore operability. They revised AP-24~~prevent spurious operation of this valve. Procedure AP/0/A/5500/045 was revised to specify that the operator ~~check that valve 2CA007A is open and remove power from 2CA0007A~~ensure, within the first ten minutes of ~~a fire~~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~~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 ~~control room~~MCR, the ~~control room~~MCR would be ~~abandoned~~evacuated and the ~~safe shutdown facility~~SSF would be used to achieve and maintain hot shutdown. The ~~safe shutdown facility relies on the turbine driven auxiliary feedwater pump for the decay heat~~

~~removal function. With the decay heat removal function seriously degraded and other mitigating systems potentially affected by a severe control room fire or Fire Area 4, the~~ finding was also determined to had~~ave~~ potential safety significance greater than very low.~~ The team was aware that system design provided for automatic transfers to alternate suction sources initiated by pressure switches in the 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 at the time of this inspection report issuance. One question was whether the automatic transfer on low suction pressure would occur fast enough to protect the pump for the case~~ significance because the SSF relies on the TDAFW pump for decay heat removal, and the decay heat removal function would be seriously degraded if the TDAFW pump were damaged due to closure of valve 2CA0007A~~ closing since this valve was close to the pump. In answering this question, the licensee stated, and presented some information, that a few events had occurred over the years where suction valves were inadvertently closed while motor driven AFW pumps were running, and the pump was not damaged. Details of these events and similarity of the motor driven and turbine driven pumps have not been reviewed by the team. Secondly, the licensee provided information to the team, subsequent to the inspection, on the routing of all the valves involved in the automatic transfers. However, this information has not yet been fully reviewed by team to determine whether or not the transfers could be affected by the same fire which caused the 2CA0007A valve to spuriously close. This information would be needed to complete the significance determination process.~~

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 ~~R, Section II.B. requires that a~~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 McGuire FPP, which includes the SSA (MCS-1465.00-00-0022), states in part, that the FPP implemented the philosophy of defense-in-depth protection against fire hazards ~~analysis shall be performed by qualified fire protection and reactor systems engineers to determine the consequences of fire in any location of the plant on the ability to safely shutdown the reactor~~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 ~~analysis~~SSA designated the MCR (Fire Area 24) and Fire Area 4 as ~~dedicated/alternative~~dedicated shutdown areas. Appendix R, Section III.G.3 requires that the ~~dedicated/alternative~~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 ~~fire hazards analysis~~SSA resulting in the ~~alternative/dedicated~~dedicated shutdown system (SSS) not being independent from Fire ~~Areas 4 and~~Area 24, in that, a fire in these areas could

result in spurious closure of ~~the~~is valve~~. This in turn could lead to damage to the turbine-driven auxiliary feedwater pump which was required for alternative shutdown using the SSS~~ and damage to the TDAFW pump. Pending determination of the safety significance, this finding is identified as URI 50-370/03-07-0~~6~~5, 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 ~~alternative shutdown~~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 ~~alternative or~~ 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 ~~alternative~~dedicated shutdown transfer and control functions into plant TS and/or SLCs; and (4) the licensee periodically performed operability testing of the ~~alternative~~dedicated shutdown instrumentation, and transfer and control functions. The team reviewed ~~abnormal p~~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.

~~The objective of this review was to assure that the safe shutdown equipment, shutdown procedures, and the post-fire safe shutdown analytical approach were consistent and satisfied the Appendix R reactor performance criteria for safe shutdown.~~

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 ~~on~~upon for SSD during a fire~~,~~ without obtaining prior NRC approval. ~~F~~A specific finding~~s~~ related to this issue ~~are~~for Fire Area 16/18 is discussed in Section ~~1R05.~~03.b.2 of this ~~inspection report for Fire Area 16/18~~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 (k~~w~~W) 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. ~~The question was raised when the team observed that a~~ 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 ~~RCS~~pressurizer to go water solid in order to maintain subcooling, and~~,~~ with the ~~RCS~~pressurizer water solid, the reactor vessel head vents would be used to control pressure. ~~The team questioned why this guidance was in these procedures.~~ 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 ~~alternative~~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 ~~alternative~~dedicated shutdown capability. The team requested information from the licensee (e.g., analyses, calculations, etc.) which demonstrated the following:

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

- ~~Are~~Validity of the assumptions for pressurizer heat loss stated in the October 21, 1980, letter ~~still valid~~ (based on insulation degradation and/or degraded capacity of the heaters -powered from SSF) for ~~assuming~~ 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 (~~JPMs~~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 ~~walk-throughs~~discussions of ~~p~~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. ~~T~~Pending further NRC review of additional licensee information, this issue is identified as URI ~~50-369~~50-369/03-07-04, 370/03-07-0~~4~~6, Methods for Reactor Coolant System Pressure Control During SSF Operation~~, pending further NRC review of additional licensee information~~.

.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 ~~(PA),~~ 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 ~~plant~~team walk downs of the selected areas where local, manual operator actions would be performed, ~~the team inspected~~ area emergency lighting units ~~(ELUs)~~were inspected for operability and ~~checked~~ the aiming of lamp heads was checked to determine if adequate illumination ~~was~~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 ~~station~~team noted that MNS has eliminated selected fire barriers from the approved fire protection program and designateesd these fire barriers as "Sealed Firewall - Non ~~Committed".~~ 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 ~~18~~16/18 have been declassified. The wall between the ~~S~~Unit 2 switchgear ~~R~~room 2ETA (Fire Area 18) and the ~~E~~Unit 2 electrical ~~P~~penetration ~~Area~~room (Fire Area 16) was declassified in Revision 9 (2000)~~ and 1~~. The wall between the ~~S~~Unit 2 switchgear ~~R~~room 2ETA (Fire Area 18) and the Unit 2 HVAC ~~E~~equipment ~~Area~~room 805A (Fire Area 18) was declassified in ~~Revision~~Rev. 3 (1982).~~ The team requested the Licensee to provide the engineering analyses that supports the declassification of these barriers.~~ For the purposes of the inspection of Fire Area 18, the ~~E~~electrical ~~P~~penetration ~~Area~~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~~, and Thermo-Lag electrical raceway fire barrier system (ERFBS) enclosures~~.

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 ~~and ERFBS~~ configurations conformed with the design drawings and tested configurations. The team also compared the penetration seal~~ and ERFBS~~ ratings with the ratings of the barriers in which they were installed.

The team reviewed licensing documentation, engineering evaluations of ~~Generic Letter~~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, ~~Design Basis Specification for Fire Protection, Fire Protection Code Deviations, and Administrative~~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 ~~against~~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 ~~accurateness~~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~~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 ~~CAP~~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~~ (OE)~~ reports were reviewed to verify that the licensee's review and actions were appropriate.

b. Findings

~~One licensee-identified finding (related to the use of manual operator actions in Fire Area 16/18 without prior NRC approval) involved a violation of NRC requirements. The enforcement considerations for this violation are discussed in Section 1R05.03.b.2 of this inspection report.~~

~~The team observed that the adequacy and timeliness of corrective actions to address the findings from the Fire Protection Functional Audit SA-99-04(MC)(RA)(FPFA) regarding fire detection in the Battery Rooms (Fire Area 13) were not commensurate with the risk significance associated with a fire in this area. The licensee's IPEEE identified that a fire in the Battery Rooms ranked as the top contributor to CDF. The fire detection findings~~No findings of significance were identified~~ in a 1999 licensee self-initiated technical audit (SITA) SA-99-04. However, the initial minor modification (MM-12907) scope was inadequate in that only two additional detectors were to be installed in the battery rooms (instead of nine required to comply with the NFPA Code). Additionally, the modification implementation date was postponed at least twice. Also, the licensee had initiated PIP M-03-01675 (dated April 10, 2003) regarding detectors not being installed in accordance with NFPA codes. When the battery rooms fire area were selected by the team during the pre-inspection information gathering visit, the team noted that the modification was revised to install the required number of detectors and received high priority status for implementation. The Battery Room detectors were installed prior to the first week of the onsite inspection (May 5-9, 2003)~~.

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 ~~t~~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 ~~Safety Evaluation Report (~~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, t~~The team presented the interim inspection results to ~~you~~Mr. D. Jamil and other members of ~~your staff, who~~the licensee's staff on May 22, 2003. A final exit meeting was held via telephone with Mr. G. Peterson, and other members of the licensee's staff on July 2, 2003, to present the final results of the inspection. The licensee acknowledged the findings presented. ~~The team confirmed that p~~Proprietary information is not included in ~~this~~the inspection 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
M. Dicks, Engineer, Reactor and Electrical Systems (RES)
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, MCE
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~~McGuire Nuclear Station
C. Thomas, Regulatory Compliance Manager
K. Thomas, Manager, RES

NRC Personnel

J. Brady, Senior Resident Inspector, ~~Shearon Harris~~
E. DiPaolo, Resident Inspector
R. Fanner, Nuclear Safety Intern (Trainee)
C. Ogle, Chief, Engineering Branch ~~Chief~~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 ~~Alternative~~Dedicated Shutdown Areas ~~n~~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 Reactor Protection System Cables Results in Loss of Required~~Inadequate Separation and

Protection of Cables Associated With Redundant Trains of Instrumentation Located in the Same Fire Area (Section 1R05.03.b.1)

~~50-369,370/03-07-03~~50-370/03-07-03 ~~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~~ URI Use of a Local Manual Operator Action in Lieu of Providing Physical Protection for Cables of Redundant Safe Shutdown Equipment in Fire Area 16/18 (Section 1R05.03.b.2)

~~50-370/03-07-06~~50-369/03-07-03, 370/03-07-04 URI Requirements Relative to the Number of Spurious Operations That Must be Postulated (Section 1R05.04.b.1)

50-370/03-07-05 URI Spurious Closure of Valve 2CA0007A Could Lead to Damage of the TDAFW Pump (Section 1R05.04.b.2)

50-369/03-07-04, 370/03-07-06 URI Methods for Reactor Coolant System Pressure Control During SSF Operation (Section 1R05.05.b)

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

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Attachment

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-2554-01.00, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 5
MCFD-2554-01.01, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 5
MCFD-2554-01.02, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 6
MCFD-2554-01.03, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 3
MCFD-2554-02.01, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 6
MCFD-2554-05.00, Unit 2 Flow Diagram of Chemical and Volume Control System, Rev. 4
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
MCSF-1560.SS-01, Summary Flow Diagram Standby Shutdown System (SSS), Rev. 2

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

Attachment

LIST OF ACRONYMS

~~AHU~~ - ~~Air Handling Unit ALARA~~ - ~~As Low As Reasonably Achievable ANS -American Nuclear Standard ANSI-American National Standards Institute AP-Abnormal Procedure ARM-Area Radiation Monitor ASME-American Society of Mechanical Engineers ASTM-American Society for Testing Materials CA-~~ABAuxiliary Building

AFW Auxiliary Feedwater~~CAP~~ - ~~Corrective Action Program CCF -Central Calibration Facility CF-Feedwater CFR-Code of Federal Regulations Co-Cobalt CP-Chemistry Procedure DPC-Duke Power Company DRP-Discrete Radioactive Particle ECCS-Emergency Core Cooling System ED-Electronic Dosimeter EDG-Emergency Diesel Generator EMF-Effluent Monitoring EnRad-Environmental Radiation EOC-End-Of-Cycle EP-Emergency Procedure ESF-Engineered Safeguards Feature ESFAS-Engineered Safety Feature Actuation System EVCC-Vital Battery C FWST-Refueling Water Storage Tank GPM-Gallons Per Minute GV-Governor Valve GWR-Gaseous Waste Release HP-Health Physics HRA-High Radiation Area HEPA-High Efficiency Particulate Air INPO-Institute of Nuclear Power Operations IR-~~

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

~~ISFSI~~ - ~~Independent Spent Fuel Storage Installation LCO~~ - ~~Limiting Condition for Operation LER-Licensee Event Report LHRA-Locked High Radiation Area LLD-Lower Limit of Detection LOCA-Loss of Coolant Accident LWR-Liquid Waste Release MGTM-Temporary Modifications MNS-~~kWKilowatt

MCR Main Control Room
MNS McGuire Nuclear Station

~~KCNC~~ - ~~Cooling water NCV~~ - ~~Non-Cited Violation ND -Residual Heat Removal NEI-Nuclear Energy Institute NI-Safety Injection NOED-Notice of Enforcement~~

Attachment

~~Discretion NSD-Nuclear Site Directive~~
~~NV~~-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
~~ODCM - Offsite Dose Calculation Manual OS - Occupational Radiation Safety PAGSS -Post Accident Gas Sampling System PI-Performance Indicator~~-PIP-Problem Investigation Process
~~report PMT - Post Maintenance Testing PS - Public Radiation Safety PT -Performance Test PWR-Pressurized Water Reactor QC-Quality Control RAB-Reactor Auxiliary Building RAP-Regulated Air Pump RCA-Radiologically Controlled Area RCZ-Radiation Control Zone RD-Radiation Dosimetry and Records Procedure REMP-Radiological Environmental Monitoring Program RF-Fire System RG-Regulatory Guide RN~~-PORVPower Operated Relief Valve
RCP Reactor Coolant Pump
RCS Reactor Coolant System
RN Nuclear Service Water
~~ROATC - Reactor Operator at the Controls RP - Radiation Protection RTP-Rated Thermal Power RWP -Radiation Work Permit SAM-Small Article Monitor SCBA-Self-contained Breathing Apparatus SDP~~-RPSReactor Protection System
SDP Significance Determination Process
~~SEIT - Significant Event Investigation Team SFP - Spent Fuel Pool SH -Shared Health Physics Procedure SLC~~-SERSafety Evaluation Report
SG Steam Generator
SLC Selected Licensee Commitment
~~SSC~~ -SMP ~~Structures, Systems, Components~~Standby Makeup Pump
SSA Safe Shutdown Analysis
SSD Safe Shutdown
SSF - Standby Shutdown Facility
SSPS - ~~Solid State Protection~~Standby Shutdown System
~~TDCA~~ -TDAFW Turbine-Driven Auxiliary Feedwater
~~TEDE - Total Effective Dose Equivalent TH - Temporary Health Physics Procedure TI -Temporary Instruction TLD-Thermoluminescent~~

Attachment

~~Dosimeter~~ TS-Technical Specifications
~~U2~~ - ~~Unit 2~~ UFSAR - Updated Final Safety Analysis Report
~~VCT~~ - ~~Volume Control Tank WBC~~ - ~~Whole-body Count WGDT -Waste Gas Decay Tank WO Work Order YC Chilled Water (control room)~~ URIUnresolved Item
V Volt

Attachment

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