From: Charlie Payne
To: Charles R. Ogle
Date: 7/7/03 4:01PM
Subject: Fwd: McGuire 2003-07 TFPI Report Draft

As requested.

H/31

From: McKenzie Thomas
To: Charlie Payne
Date: 6/26/03 9:15AM
Subject: McGuire 2003-07 TFPI Report Draft

See Attached

July XX, 2003

Duke Energy Corporation
ATTN: Mr. D. Jamil
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:

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

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

This report documents two findings that, combined, have potential safety significance greater than very low significance, however, a safety significance determination has not been completed. One finding did present an immediate safety concern and a fire watch was put in place on June 10, 2003, 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,

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

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

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

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Enclosure

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) 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; McGuire Nuclear Station, Units 1 and 2; Triennial Fire Protection

The report covered a two-week period of inspection by regional inspectors and a consultant. Three unresolved items with potential safety significance greater than Green were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- TBD. The team identified a violation involving Train A and Train B cables associated with the reactor protection were not protected from fire damage.

  This finding is unresolved pending determination of the systems affected and completion of a significance determination. The finding is greater than minor because instrumentation important for post-fire safe shutdown would 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 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 due to fire damage.

  The finding is unresolved pending completion of a significance determination. The finding is greater than minor because spurious closure of the valve could damage the TDAFW pump and seriously degrade the decay heat removal function. (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 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 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 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

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 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

The team reviewed the licensee's FPP documented 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 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. Systems and/or components selected for review included: standby shutdown system (SSS); Unit 2 standby makeup pump (SMP) 2NVPU0046 and SMP suction supply motor operated valve (MOV) 2NV842AC; auxiliary feedwater (AFW) suction supply MOVs 2CA007A, 2CA009B, 2CA161C, and 2CA162C; reactor coolant pump (RCP) seal water return isolation valve 2NV94AC; pressurizer power operated relief valve (PORV) 2NC34A and PORV isolation valves 2NC33A; Unit 2 pressurizer heater Nos. 28,55, and 56; reactor vessel head vent valves 2NC272AC and 2NC273AC; and heating, ventilation, and air conditioning (HVAC).

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 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 shutdown using the SSS was designated by the licensee for a fire in this area. 10 CFR 50, Appendix R, Section III.G.3 (alternative or dedicated shutdown) requires that fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration. The fire suppression system for Fire Area 4 was not installed in accordance with 10 CFR 50, Appendix R, Section III.G.3. The system in Fire Area 4 was a partial automatic sprinkler system effectively protecting the RN pumps and 20 feet north of these pumps. The area protected by this sprinkler system was located between Column lines 54-58 and EE-GG. The majority of Fire Area 4 was not provided with automatic sprinkler protection as required by 10 CFR 50, Appendix R, Section III.G.3 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. 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 50-369, 370/87-34. The team noted that, subsequent to closure of the URI, licensee Fire Protection Functional Audit SA-99-04(MC)(RA)(FPFA) dated April 9, 1999, identified that MNS did not meet separation and detection/suppression criteria for alternative or dedicated shutdown capability required by 10 CFR 50, Appendix R, Section III.G.3. During this inspection, the team questioned whether the previous reviews of the sprinkler system for this fire area included an evaluation of the risk impact associated with not providing adequate sprinkler coverage for the RN cabling in this fire area. The team informed the licensee that this issue will 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 similar conditions, regarding the fixed fire suppression system complying with 10 CFR 50, Appendix R, Section III.G.3, was applicable to other MNS fire areas where alternative 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) evaluation is warranted.

.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. 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 the charging/safety injection system, RCS and AFW system. The team traced the routing of cables by using the cable schedule and conduit and 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 AFW Suction Supply Valve

- 2CA0007A, Turbine-driven AFW Suction Isolation Valve
- 2CA009B, Motor-driven AFW Suction Isolation Valve
- 2CFLT6080, 6090, 6100, 6110, Steam Generator Level Transmitters
- 2NCLT5151, Pressurizer Level Transmitter
- 2NC34A, 33A, Pressurizer PORV and 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 Heaters Nos. 28, 55, 56

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

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

Description: Fire Area 16/18 is the Unit 2 Train A switchgear room, and Train B equipment controlled from the main control room was intended to be used for a fire in this area according to the analysis and procedures. 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 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. The team also observed Train B cables routed in this room. Many of the identified cables were in a cable tray near the ceiling and were going from/to the cable spread room, which is on the same elevation, to/from the control room, which was above the switchgear room. The licensee had not been aware of all of these "opposite train" cables, and they initiated PIP M-03-02106 . On June 10, 2003, the licensee reported these cables represented an unanalyzed condition (Event No. 39915).

As many as 74 "opposite train" cables are involved related to the reactor protection system. Preliminary investigation by the licensee revealed that cables for primary and backup power supplies for all four RPS channels were routed in close proximity and could be damaged. One consequence of this is that many RPS protective functions would spuriously go to the trip condition. Subsequently, a safety injection signal would be generated due to spurious "high containment pressure." A safety injection signal

would in turn trigger a reactor trip and Phase A isolation. At the same time, many important main control panel instruments would be lost. For example, pressurizer level and all four steam generator level, which are instruments necessary to achieve and maintain hot shutdown. The licensee also stated that a similar situation exists for the Unit 1 Train A switchgear room (Fire Area 17).

Analysis: The fact that 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. Pending determination of 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 Instrumentation.

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

Introduction: A finding was identified in that physical protection of the associated electrical cables for valve 2CA0042B (2B motor driven AFW pump discharge supply to steam generator 2D) did not meet the requirements of 10 CFR 50, Appendix R, Section III.G.2. Instead, the licensee substituted the use of 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, the licensee identified that MNS relied on manual operator actions outside the MCR for SSD in non-alternative shutdown fire areas (i.e., areas designated as complying with 10 CFR 50, Appendix R, Section III.G.2) and the manual actions did not have prior NRC approval. The licensee documented this issue in PIP M-03-02311. The team reviewed the local, manual operator actions for the Section III.G.2 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 Train A 2ETA/Electrical Penetration 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, 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 manual operator actions outside the MCR. 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 local manual actions had not received NRC approval.

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

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

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

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

.04 Alternative Post-Fire Safe Shutdown Capability

a. Inspection Scope

The team reviewed the licensee's procedures for fire response, abnormal procedures for alternative shutdown (ASD), and the licensee's Appendix R manual action requirements analyses for a fire in the selected Fire Areas 4, 13, and 24. The team also walked down selected portions of the procedures. 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 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 were also reviewed in relation to alternative post-fire safe shutdown capability. The team reviewed the most recently completed surveillances for selected instruments required during SSS operation to verify that surveillances were being completed in accordance with MNS SLC 16.9.7, Standby Shutdown System. The walk downs focused on ensuring that the procedures 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 remote 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 identified involving the number of concurrent spurious operations associated with a particular component or set of components that must be postulated. Resolution of the unresolved item is pending review by NRC staff.

Description: The licensee's fire protection analysis 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 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 a fire alarm or the plant communications system, they would respond by following the instructions in abnormal procedure AP/0/A/5500/045, Plant Fire. Depending on the fire location, procedure AP/0/A/5500/045 directed the operator to close the PORV isolation valves within ten minutes. The basis for this time critical action is that spurious opening of the PORV or damage to the isolation valve circuit would not occur in the first ten minutes of a fire being detected. Then with the block valve closed it would take two spurious operations to breach the RCS pressure boundary, namely one block valve opening and its associated PORV opening. The concept of only one spurious operation need be postulated meant that closing the block valve was sufficient in itself to ensure the desired result. The licensee considered that there was no need to take any other action such as de-energizing the isolation valves after they were closed. This concept was not necessarily consistent with NRC requirements for protection of cables.

The team reviewed the control circuits and cable routing information for valve 2NC34A, pressurizer PORV, and 2NC33A its associated isolation valve. They observed that cables for both the PORV and isolation valve are routed in 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 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 more than one spurious operation would occur, the dedicated shutdown capability (SSS) would not be independent from the control room in that a fire in the control room could result in conditions outside of those specified in III.L. 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.

Enforcement: In the case of the PORV/isolation valve circuits, operation of the SSS may not be independent of the fire area as required by 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. The issue is identified as URI 50-369,370/03-07-03, Requirements Relative to the Number of Spurious Operations that must be Postulated.

2. Valve 2CA0007A

Introduction: A finding of potentially greater than very low safety significance was identified in that a valve in the auxiliary feedwater system was not included in the safe shutdown analysis and it could spuriously close due to a fire in the main control room. Spurious closure of this valve could damage the turbine driven auxiliary feedwater pump, thus seriously degrading the core residual heat removal function of the safe shutdown system. This is a URI pending completion of the SDP.

Description: Valve 2CA0007A is a motor operated valve in the flow path from the 300,000 gallon auxiliary feedwater storage tank to the turbine driven auxiliary feedwater pump. The valve is open during normal plant operation. 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 the SSS uses the TDAFW pump for 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 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 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 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 licensee initiated a corrective action document for this issue, PIP M-03-02084, and they took prompt action to restore operability. They revised AP-24 to specify that the operator check that valve 2CA007A is open and remove power from 2CA0007A within the first ten minutes of a fire.

Analysis: The team determined that this finding was associated with the "equipment performance" attribute and affected the objective of the mitigating systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events, and is therefore greater than minor. For a severe fire in the control room, the control room would be abandoned and the safe shutdown facility would be used to 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 had a 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 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, Appendix R, Section II.B. requires that a 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. The licensee's analysis designated the MCR and Fire Area 4 as dedicated/alternative shutdown areas. Appendix R, Section III.G.3 requires that the dedicated/alternative 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 resulting in the alternative/dedicated shutdown system (SSS) not being independent from Fire Areas 4 and 24 in that a fire in these areas could result in spurious closure of the valve. This in turn could lead to damage to the turbine driven auxiliary feedwater pump which was required for alternative shutdown using the SSS. Pending determination of the safety significance, this finding is identified as URI 50-370/03-07-06, Spurious Closure of Valve 2CA0007A Could Lead to Damage of the TDAFW Pump.

.05 Operational Implementation of Post-Fire Safe Shutdown Capability

a. Inspection Scope

The team reviewed the operational implementation of the alternative shutdown capability for a fire in Fire Areas 4, 13, 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 shutdown transfer and control functions into plant TS and/or SLCs; and (4) the licensee periodically performed operability testing of the alternative shutdown instrumentation and transfer and control functions. The team reviewed abnormal 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 manual operator actions outside the MCR were used in lieu of physical protection of equipment and cables relied on for SSD during a fire, without obtaining prior NRC approval. Findings related to this issue are discussed in Section 1R05.03.b.2 of this inspection report for Fire Area 16/18.

The team identified a URI regarding the adequacy of the licensee's method for controlling RCS pressure during operation from the SSF in the event of a fire.

During review of procedures AP/1/A/5500/024 and AP/2/A/5500/024, the team questioned the adequacy of the 70 kilowatts (kw) pressurizer heater capacity per unit powered from the SSF to maintain and control RCS pressure in hot standby during a fire in plant areas which require use of the SSS. The question was raised when the team observed that 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 to go solid in order to maintain subcooling and, with the RCS 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 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 shutdown capability. The team requested information from the licensee (e.g., analyses, calculations, etc.) which demonstrated the following:

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

  - Are 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, 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 of procedures AP/1/A/5500/024 and AP/2/A/5500/024. The team concluded that sufficient information was not provided to resolve the questions raised above nor to determine the licensee's

ability to safely operate the SSF with the pressurizer in a water solid condition during fire events in areas where the SSF is used to achieve SSD. This issue is identified as URI 50-369,370/03-07-04, 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 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 walk downs of selected areas where local manual operator actions would be performed, the team inspected area emergency lighting units (ELUs) for operability and checked the aiming of lamp heads to determine if adequate illumination was 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. The station has eliminated fire barriers from the approved fire protection program and designates these fire barriers as "Sealed Firewall - Non Committed". These barriers are no longer included in any surveillance and testing program. Therefore, doors, dampers, fire proofing, etc. that exist in these declassified barriers are no longer included in any station surveillance procedures and effectively cannot be relied upon for the fire protection program. Two walls associated with Fire Area 18 have been declassified. The wall between the Switchgear Room (Fire Area 18) and the Electrical Penetration Area (Fire Area 16) was declassified in Revision 9 (2000) and the wall between the Switchgear Room (Fire Area 18) and the HVAC Equipment Area (Fire Area 18) was declassified in Revision 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 Electrical Penetration Area (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, 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 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 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 evaluation for NFPA code deviations were reviewed and compared against 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 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 areas.

The team reviewed the adequacy of the design and installation of the automatic detection and alarm system for the selected 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 areas.

The team reviewed the fire protection pre-plans and fire strategies to ensure that hose locations could sufficiently reach the selected 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 areas.

b. Findings
No findings of significance were identified.

4. Other Activities

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of licensee audits, self-assessments, and PIPs to verify that items related to fire protection and to SSD were appropriately entered into the licensee's CAP 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 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 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, the team presented the inspection results to you and other members of your staff, who acknowledged the findings. The team confirmed that proprietary information is not included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

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

NRC Personnel

J. Brady, Senior Resident Inspector, Shearon Harris
E. DiPaolo, Resident Inspector
R. Fanner, Nuclear Safety Intern (Trainee)
C. Ogle, Engineering Branch Chief, 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 Shutdown Areas not in Accordance with 10 CFR 50, Appendix R, Section III.G.3 (Section 1R05.02.b)

50-369,370/03-07-02 URI Failure to Protect Reactor Protection System Cables Results in Loss of Required Instrumentation (Section 1R05.03.b.1)

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

50-369,370/03-07-04 URI Methods for Reactor Coolant System Pressure Control

During SSF Operation (Section 1R05.05.b)

50-370/03-07-05 URI Failure to Provide Adequate Protection for Cables of Redundant Safe Shutdown Equipment in Fire Area 16/18 (Section 1R05.03.b.2)

50-370/03-07-06 URI Spurious Closure of Valve 2CA0007A Could Lead to Damage of the TDAFW Pump (Section 1R05.04.b.2)

Closed

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

Discussed

None

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	-	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	-	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	-	McGuire Nuclear Station
KC	-	Cooling water

Attachment

NCV	-	Non-Cited Violation
ND	-	Residual Heat Removal
NEI	-	Nuclear Energy Institute
NI	-	Safety Injection
NOED	-	Notice of Enforcement Discretion
NSD	-	Nuclear Site 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	-	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	-	Significance Determination Process
SEIT	-	Significant Event Investigation Team
SFP	-	Spent Fuel Pool
SH	-	Shared Health Physics Procedure
SLC	-	Selected Licensee Commitment
SSC	-	Structures, Systems, Components
SSF	-	Standby Shutdown Facility
SSPS	-	Solid State Protection System
TDCA	-	Turbine-Driven Auxiliary Feedwater
TEDE	-	Total Effective Dose Equivalent
TH	-	Temporary Health Physics Procedure
TI	-	Temporary Instruction
TLD	-	Thermoluminescent Dosimeter
TS	-	Technical Specifications
U2	-	Unit 2
UFSAR	-	Updated Final Safety Analysis Report
VCT	-	Volume Control Tank

Attachment

WBC	-	Whole-body Count
WGDT	-	Waste Gas Decay Tank
WO	-	Work Order
YC	-	Chilled Water (control room)

Attachment

Mail Envelope Properties (3F09D1A6.A4B : 4 : 51305)

Subject:	Fwd: McGuire 2003-07 TFPI Report Draft
Creation Date:	7/7/03 4:01PM
From:	Charlie Payne
Created By:	DCP@nrc.gov

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CRO (Charles R. Ogle)

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Expiration Date:	None
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Reply Requested:	No
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Concealed Subject:	No
Security:	Standard