UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-369/84-28 and 50-370/84-25

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242

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

Facility Name: McGuire

Inspection Conducted: September 24 - 28, 1984

Inspectors:	T E Conlon for M. D. Hunt	11-28-84 Date Signed
	W. H. Miller, Jr. P. M. Madden	11-28-84 Date Signed
	W. H. Miller Jr. W. H. Miller, Jr.	11-20-84 Date Signed
	P. A. Taylor P. A. Taylor	11-27-84 Date Signed
	G. R. Wiseman G. R. Wiseman	11-28-84 Date Signed

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Approved by: T E Conlon 11-28-84
T. E. Conlon, Section Chief Date Signed
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope: This special, announced inspection entailed 350 (70 contractor and 280 NRC) inspector-hours on site in the areas of fire protection, standby shutdown system (SSS) and related features required to meet 10 CFR 50 Appendix R, Sections III.G, III.J, III.L and III.O.

Results: Of the area inspected, six apparent violations were identified: inadequate or failure to provide fixed suppression system in accordance with 10 CFR 50, Appendix R, Section III.G.3 for rooms, area, and zones under consideration - paragraph 5.a; failure to provide adequate breaker/fuse protection for equipment required for hot standby - paragraph 5.b; failure to comply with the requirements of 10 CFR 50, Appendix R, Section III.J. - paragraph 7.a.; failure to provide automatic fire detection for, and fire barriers to separate, safety-related pumps - paragraph 8.a; structural steel fire barrier supports not provided with fire resistant rating equivalent to the fire barrier - paragraph 9.a; inadequate Appendix R, Section III.G, fire protection features and separation provided for redundant trains of normal shutdown systems and the standby shutdown system - paragraph 9.b. Two deviations were found: failure to provide battery powered hand lanterns in the control room - paragraph 7.a.; Failure to provide adequate radio communications between local control stations and the standby shutdown facility - paragraph 8.d.

REPORT DETAILS

1. Licensee Employees Contacted

*J. V. Almond, Safety Supervisor
*T. A. Belk, Engineer Associate
*H. D. Brandes, Design Engineer
*J. M. Bugs, Design Engineer
*K. S. Canady, Manager, Nuclear Engineering Service
*R. Gill, Licensing Engineer
*A. D. Harrington, Training and Safety Coordinator
*J. R. Hendricks, Principle Design Engineer
*D. B. Hyde, Associate Engineer
*J. A. Keane, Associate Engineer
*D. P. Kimball, Associate Engineer
*T. A. Ledford, Superintendent, Design Engineering
*T. V. Lyerly, IAE Staff Coordinator
*W. N. Matthews, Design Engineer
*S. H. McInnis, Compliance
M. D. McIntosh, Station Manager
*D. Mendezoff, Licensing Engineer
*J. A. Oldham, Design Engineer
*D. J. Rains, Superintendent of Maintenance
*W. O. Reeside, Associate Engineer
*R. W. Revels, Design Engineer
*N. Rutherford, Licensing Engineer
*B. Travis, Operations Engineer
*G. Vaughn, General Manager, Nuclear Stations
*L. E. Weaker, Superintendent Station Services
*C. H. Whitmore, Senior Designer

NRC Resident Inspectors

*W. T. Orders
*R. C. Pierson

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on September 28, 1984, with those persons indicated in paragraph 1 above. The following inspection findings were identified to the licensee:

a. Violation Item (369/84-28-01 and 370/84-25-01), Inadequate or Failure to Provide Fixed Suppression Systems in Accordance With 10 CFR 50, Appendix R, Section III.G.3 for Rooms, Areas, or Zones Under Consideration - paragraph 5.a.

b. Unresolved Item (369/84-28-02 and 370/84-25-02), Inadequate Fixed Fire Suppression System Provided for the Cable Spreading Room and Battery Room - paragraph 5.a.(2)(b).

c. Violation Item (369/84-28-03 and 370/84-25-03), Failure to Provide Adequate Breaker/Fuse Protection for Equipment Required for Hot Standby - paragraph 5.b.(1).

d. Violation Item (369/84-28-04 and 370/84-25-04), Failure to Comply With the Requirements of 10 CFR 50, Appendix R, Section III.J - paragraph 7.a.

e. Deviation Item (369/84-28-05 and 370/84-25-05), Failure to Provide Battery Powered Hand Lanterns in the Control Room - paragraph 7.a.

f. Inspector Followup Item (369/84-28-06 and 370/84-25-06), Inadequate Surveillance Testing Procedures for Emergency Lighting - paragraph 7.b.

g. Violation Item (369/84-28-07 and 370/84-25-07), Failure to Provide Automatic Fire Detection for, and Fire Barriers to Separate Safety-Related Pumps - paragraph 8.a.

h. Unresolved Item (369/84-28-08 and 370/84-25-08), Adequacy of Power Supplies for Fire Pumps A, B, and C - paragraph 8.c.

i. Deviation Item (369/84-28-09 and 370/84-25-09), Failure to Provide Adequate Radio Communications Between Local Control Stations and Standby Shutdown Facility - paragraph 8.d.

j. Violation Item (369/84-28-10 and 370/84-25-10), Structural Steel Fire Barrier Supports Not Provided With Fire Resistant Rating Equivalent to the Fire Barrier - paragraph 9.a.

k. Violation Item (369/84-28-11), Inadequate Appendix R, Section III.G, Fire Protection Features and Separation Provided for Redundant Trains of Normal Shutdown Systems and the Standby Shutdown System - paragraph 9.b.

l. Unresolved Item (369/84-28-12 and 370/84-25-12), NRR Evaluation of Appendix R, Deviation Request - paragraph 9.c.

m. Inspector Followup Item (369/84-28-13 and 370/84-25-13), Amplify and Clarify Certain Steps of OP/0/A/6100/17 - paragraph 5.c.(2)(a).

n. Unresolved Item (369/84-28-14 and 370/84-25-14) Analysis Effects on SSF/SSS Operation, Potential Excessive Feedwater to Steam Generators - paragraph 5.c.(2)(a)(7).

o. Unresolved Item (369/84-28-15 and 370/84-25-15), Correct Procedure Deficiencies used to Accomplish 10 CFR 50, Appendix R, Section III.L - paragraph 5.c.(2)(b).

3. Licensee Action on Previous Enforcement Matters

   This subject was not addressed in the inspection.

4. Unresolved Items

   Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. New unresolved items identified during this inspection are discussed in paragraphs 5.a(2)(b), 5.c.(2)(a)(7), 5.c.(2)(b), 8.c. and 9.c.

5. Compliance With 10 CFR 50, Appendix R Sections III.G. and III.L

   An inspection was conducted to determine if the fire protection features provided for structures, systems, and components important to safe shutdown at McGuire were in compliance with 10 CFR 50, Appendix R, Sections III.G. and III.L. Since the McGuire Nuclear Station utilizes the dedicated shutdown system approach, the scope of this inspection determined if the fire protection features provided were capable of maintaining either the safe shutdown system or one train of normal plant hot standby systems free from fire damage, and were capable of limiting potential fire damage to both trains of redundant normal plant safe shutdown systems in those plant areas where alternate or dedicated shutdown capabilities are provided.

   a. Safe Shutdown Capabilities

      In order to ensure safe shutdown capabilities, where cables or equipment of redundant trains of systems necessary to achieve and maintain hot stand-by conditions are located within the same fire area outside the primary containment, 10 CFR 50, Appendix R, Section III.G.2 requires that one train of hot standby systems be maintained free of fire damage by one of the following means:

      Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating;

      Separation of cables and equipment and associated non-safety circuits 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; or,

      Enclosure of cable and equipment and associated non-safety circuits 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.

Where the protection of systems whose function is required for hot standby does not satisfy the above requirements or Section III.G.2, alternative or dedicated shutdown capabilities independent of cables, systems or components in the area, room or zone under consideration shall be provided in accordance with 10 CFR 50, Appendix R, Section III.G.3 and III.L. In addition, Section III.G.3 requires that fire detection and fixed suppression be installed in the area, room or zone under consideration.

On the basis of the above Appendix R criteria, the inspectors made an audit of cabling and components associated with the dedicated standby shutdown system (SSS) to determine the adequacy of the separation afforded to the SSS with respect to plant areas containing both redundant trains of normal essential hot standby systems (i.e, auxiliary feedwater system, component cooling water system, nuclear service water system, chemical volume control system and reactor coolant system). In addition, the inspectors made an audit of the standby shutdown system's ability to achieve and maintain hot standby and determined the adequacy of the fire protection features afforded for those plant areas which contain both redundant trains or normal essential plant systems required to achieve and maintain safe shutdown conditions.

(1) Separation of Standby Shutdown System (SSS) from Normal Plant Shutdown Systems.

A walk down inspection was made of the following SSS cables routes within the Unit 1 auxiliary building to verify that the SSS cables were separated from the redundant or compliment device of the normal essential plant shutdown system in accordance with the requirements of 10 CFR 50, Appendix R, Section III.G.2.

Function	SSS Device	SSS Cable(s)	Compliment Device	Compliment Cable(S)
Pressurizer Level Inst.	1NCP5151	1CF726 1NC992 1NC990	1NCLT5170	1NC665
Steam Generator B Level Inst.	1CFP6090	1CF726 1CF773 1CF771	1CFP5530	1CF588
Reactor Coolant Pressure Inst.	1NCP5121	1CF726 1NC995 1NC993	1NCPT5170	1NC713
Changing Flow Inst.	1NVP6420	1CF726 1NV826 1NV827	Control Gauge	1NV657

Reactor Coolant System Vent	1NC1, Valve NC272AC	1NC984	Control Room Controls (Note*)	1NC971
	1NC2, Valve NC273AC		Valves NC272AC & NC273AC	
Reactor Coolant System Isolation	1NC3, Valve NC27	1NC977	Control Room Controls (Note*) Valve NC27	1NC707
	1NC4, Valve NC29	1NC977	Valve NC29	1NC706
	1ND4, Valve ND2AC	1WZ540	Valve ND2AC	1EPE590
	1NV1, Valve NV94AC	1WZ540	Valve NV94AC	1EPE590

NOTE: *Normal shutdown and SSS cables terminate in cabinet SSSFARC. Normal shutdown cables extend from this cabinet to normal system isolation valves. Cables are routed by separate routes to assure that the cables will not ground or fault in such a manner to cause spurious valve operations.

In general, the SSS cables are routed with Train A cables and are separated from Train B cables by three hour fire rated barriers. A review was conducted of drawing Nos. MC-1919-01.01, MC1920-01.01, and MC1921-02.01 and "computer cable routing program data sheets" to determine the above cable routes within the auxiliary and reactor buildings. Portions of the SSS cabling and the normal shutdown cabling within the annulus of the reactor building are in close proximity to each other. This situation is not in violation to Appendix R, Section III.G.2, since the annulus is considered part of the containment and is provided with fire detection and automatic sprinkler protection.

For a fire within the remainder of the containment, the SSS is not required to bring the plant to hot standby condition. The normal shutdown systems would be used and these systems have been evaluated by the licensee and NRR and found to be adequate to bring the plant to a safe shutdown condition following a containment fire.

A review of components and cable route drawing for the SSS within Unit 2 indicated that the installation within Unit 2 was similar to Unit 1 and appears to be provided with sufficient separation from the normal shutdown components to meet the requirements of 10 CFR 50 Appendix R, Section III.G.

(2) Fire Protection of Safe Shutdown Capabilities

(a) An inspection was made to determine if the fire protection features provided for various auxiliary building areas meet the fire protection requirements of 10 CFR 50 Appendix R, Section III.G.3.

1) Fire Area 4 - Auxiliary Building Common Area 649

On elevation 716'-0" of the Auxiliary Building a partial fixed automatic sprinkler system is provided to Common Area 649 to protect the Nuclear Service Water (RN) pumps. The area of sprinkler application in Common Area 649 is provided between column line EE-GG-54 and column line EE-GG 58. Power cables 1*RN571 and 2*RN559 for RN pumps 1A and 2A are partially routed outside the area protected by sprinklers in Common Area 649 from column line GG 56 to HH 54, where they enter the unprotected electrical duct shaft adjacent to Charging Pump Room 627 and are routed up through elevation 733'-0" to elevation 750'-0." Power cables 1*RN572 and 2*RN560 for RN pumps 1B and 2B are routed in Common Area 649 from the pumps to the unprotected electrical duct shaft near column FF57 within the sprinklered area, where they enter the shaft and are routed up to elevation 733'-0."

Even though train "B" RN pump power cables are located within the sprinklered area and the train "A" RN pump power cables are partially routed through this area, the water discharge pattern for the sprinkler heads installed near the ceiling level over both redundant trains of RN pumps and their associated cabling appears to be obstructed by cable trays and piping. In addition, the sprinkler protection does not fully protect the train "A" RN pump cabling in Common Area 649. Therefore, based on the sprinkler obstructions and the lack of adequate fixed suppression coverage for both redundant trains of normal shutdown systems located in Auxiliary Building Common Area 649, it cannot be assured that fire damage to both trains of the nuclear service water system can be minimized if an exposure fire were to occur within this plant area.

2) Fire Area 14 - Auxiliary Building Common Area 723

Component Cooling Water (KC) pump suction isolation valves 1*KC-1A, 1*KC-2B, 1*KC-3A and 1*KC-18B are located in Common Area 723 on Auxiliary Building elevation 733'-0". Common Area 723 is partially

protected by an automatic sprinkler system. The area of sprinkler application in Common Area 723 is provided between column line GG-JJ 55 and column line GG-JJ 57. Valves 1*KC-1A, 1*KC-2B and 1*KC-18B are located within the Component Cooling Water (KC) pump area which is protected by the automatic sprinkler system. The sprinkler water discharge patterns for those sprinkler heads which provide partial protection for valves 1*KC-1A, 1*KC-2B and 1*KC-18B and their control cables 1*KC501, 1*KC527, 1*KC515, 1*KC541, and 1*KC516 appear to be obstructed by cable trays and piping. Component Cooling Water (KC) suction valve 1*KC-3A and its control cabling 1*KC502 and 1*KC527 and control cables 1*KC501 are located outside the sprinklered area near column line HH54 and JJ54. Therefore, based on the sprinkler obstructions and the lack of adequate fixed suppression coverage for both redundant trains of normal shutdown systems located in Common Area 723, it cannot be assured that fire damage to both trains of the Component Cooling Water System can be minimized if an exposure fire were to occur within this plant area.

Unit 1 Centrifugal Charging (NV) pump 1A and 1B power cables 1*NV501 and 1*NV502 are located in Common Area 723 on auxiliary building elevation 733'-0". Common Area 723 is partially protected by an automatic sprinkler system. The area of sprinkler application in Common Area 723 is provided between column line GG-JJ55 and column line GG-JJ57. Cable 1*NV502 is routed up to elevation 733'-0" from elevation 716'-0" through the unprotected cable shaft located outside the sprinklered area near column line HH-54. Once on elevation 733'-0", cable 1*NV502 stays in the cable shaft and is routed up to auxiliary building elevation 750'-0" where it terminates. Cable 1*NV501 is routed up to elevation 733'-0" from elevation 716'-0" through an electrical floor penetration near column line JJ-55. Cable 1*NV501, once on elevation 733'-0", is routed across Common Area 723 outside the sprinklered area to column line JJ-56. Once cable 1*NV501 has reached this plant location, it makes a 90° turn south and is routed through the sprinklered area. The water discharge pattern for the sprinklers installed near the ceiling level over both redundant trains of Component Cooling Water pumps and cable 1*NV501 appears to be obstructed by piping and cable trays.

Therefore, based on the sprinkler obstructions and the lack of adequate fixed suppression coverage for both redundant trains of normal shutdown systems located in Common Area 723, it cannot be assured that fire damage

to both trains of the Unit 1 Chemical Volume Control System can be minimized if an exposure fire were to occur within this plant area.

Unit 2 Centrifugal Charging (NV) pump 2A and 2B power cables 2*NV538 and 2*NV537 are located in auxiliary building Common Area 723 on elevation 733'-0". Common Area 723 is partially protected by an automatic sprinkler system. The area of sprinkler application in Common Area 723 is provided between column line GG-JJ55 and column line GG-JJ57. Cable 2*NV537 is routed up to elevation 733'-0" from elevation 716'0" through the unprotected cable shaft located outside the sprinklered area near column line JJ-58. Once on elevation 733'-0", cable 2*NV537 stays in the cable shaft and is routed up to auxiliary building elevation 750'-0" where it terminates. Cable 2*NV538 is routed up to elevation 733'-0", from elevation 716'-0" through an electrical floor penetration near column line HH57. Cable 2*NV538 once on elevation 733'-0", is routed through the sprinklered portion of common area 723. However, the water discharge pattern for the sprinkler heads installed near the ceiling level over cable 2*NV538 appears to be obstructed by piping and cable trays. Therefore, based on the sprinkler obstructions and the lack of adequate fixed suppression coverage for both redundant trains of normal shutdown systems located in Common Area 723, it cannot be assured that fire damage to both redundant trains of the Unit 2 Chemical Volume Control System can be minimized if an exposure fire were to occur within this plant area.

Cabling for both trains of redundant Charging Flow Isolation Valves is located in Auxiliary Building Common Area 723 on elevation 733'-0". Common Area 723 is partially protected by an automatic sprinkler system. The area of sprinkler application within this area is provided between column line GG-JJ57 and GG-JJ55. Cable 1*NV555, which is associated with Charging Flow Isolation Valve 1*NV244A, is routed up to elevation 733'-0" from elevation 716'-0' through the unprotected cable shaft located outside the sprinklered area near column line HH56. Cables 1*NV575 and 1*NV576 are routed through the portion of Common Area 723 which is protected by sprinklers. However, the water discharge pattern of the sprinkler heads installed near the ceiling over cables 1*NV575 and 1*NV576 appears to be obstructed by cable trays and piping. Therefore, based on the sprinkler obstructions and the lack of fixed suppression coverage for both redundant trains of normal shutdown systems located in Common Area 723, it cannot

be assured that fire damage to both redundant trains of Unit 1 Chemical Volume Control System can be minimized if an exposure fire were to occur within this plant area.

3) Fire Area 4 - Mechanical Penetration Room 603

Normal Charging Flow Isolation Valves 1*NV244A and 1*NV245B to the Reactor Coolant System are located in Mechanical Penetration Room 603 on elevation 716'-0" of the auxiliary building. This room is not provided with a fixed suppression system. The cabling associated with these valves is also located within this room. Cables 1*NV55B and 1*NV579 are routed out of Mechanical Penetration Room 603 in two separate directions. These cables and cable 1*NV576 associated with valve 1*NV245B and cable 1*NV555 associated with valve 1*NV244A reconverge in auxiliary building Common Area 649. Common Area 649 is partially protected by an automatic sprinkler system. The area of sprinkler application within this area is provided between column lines EE-GG 54 and EE-GG58. Cable 1*NV579 and 1*NV576 for Charging Flow Isolation Valve 1*NV245B terminate within cabinet 1ATC3. Termination cabinet 1ATC3 and cable 1*NV576 are located within the sprinklered area. However, cable 1*NV579 is only partially located in the sprinklered area. Cables 1*NV558 and 1*NV555 for Charging Flow Isolation Valve 1*NV244A terminate within cabinet 1ATC6. Termination cabinet 1ATC6 and cables 1*NV558 and 1*NV555 are located outside the sprinklered area. The water discharge pattern for the sprinkler heads installed near the ceiling over termination cabinet 1ATC3 and in the area of the cable routes for cables 1*NV579 and 1*NV576 appears to be obstructed by cable trays and piping. Therefore, based on the sprinkler obstructions and the lack of fixed suppression coverage for both redundant trains of normal shutdown systems located in Common Area 649 and mechanical penetration room 603, it cannot assured that fire damage to both redundant trains of the Unit 1 Chemical Volume Control System can be minimized if an exposure fire were to occur within either of these plant areas.

4) Fire Area 14 - Corridor 731

Volume Control Tank Isolation Valves 1*NV141A and 1*NV142B are located in corridor 731 on auxiliary building elevation 733'-0". Cable 1*NV560 associated with valve 1*NV141A and cable 1*NV582 associated with valve 1*NV142B are routed from the valves down to the end of Corridor 731, then they take a 90° turn and run

down the corridor adjacent to the Boric Acid Tank Rooms and into Common Area 753 where they separate. If an exposure fire were to affect these valves or their associated cables, causing these valves to spuriously operate and go closed, the normal suction flow path to the charging pumps could be precluded. There are no fixed suppression capabilities provided in either Corridor 731 or the corridor adjacent to the Boric Acid Addition Tank Rooms to protect the Volume Control Tank Isolation Valves and their associated cabling. Therefore, based on the lack of fixed suppression in these plant areas, it cannot be assured that fire damage to both redundant trains of Unit 1 Chemical Volume Control System can be minimized.

5) Fire Area 21 - Auxiliary Building Common Area 806

Nuclear Service Water Supply Isolation Valve 1*RN86A to Component Cooling Water Heat Exchanger 1A and Nuclear Service Water Supply Isolation Valve 1*RN187B to Component Cooling Water Heat Exchanger 1B and their associated cabling are located in Common Area 806 (fire area 21) on auxiliary building elevation 750'-0." If an exposure fire were to affect these valves or their associated cables, causing these valves to spuriously operate and go closed, the normal nuclear service water supply to the component cooling water heat exchangers could be precluded. There are no fixed suppression capabilities provided in Common Area 806 near the area the heat exchangers to protect the nuclear service water supply isolation valves and associated cabling. Therefore, based on the lack of fixed suppression in the area of the nuclear service water supply isolation valves to the component cooling water heat exchangers and their associated cabling, it cannot be assured that fire damage to both redundant trains of Unit 1 Component Cooling Water System can be minimized.

The plant areas identified in items 5.a.(2)(a)(1) through 5.a.(2)(a)(5), are areas, rooms, or zones under consideration and if affected by a fire condition would require the utilization of the dedicated standby shutdown system to achieve and maintain hot standby conditions. 10 CFR 50, Appendix R, Section III.G.3, requires a fixed suppression to be installed in the room, zone or area under consideration. The plant areas, zones, or rooms previously identified, are either not provided with fixed suppression capabilities or the fixed suppression system provided does not provide adequate coverage to protect both redundant trains of

normal safe shutdown systems. Therefore, if an exposure fire were to occur within any of these identified areas, it could not be assured that the fire damage sustained by both redundant trains of normal shutdown systems would be minimal. The failure to meet the fire protection requirements of 10 CFR 50 Appendix R, Section III.G.3, as required by the operating license, is identified as Violation Item (369/84-28-01 and 370/84-25-01) Inadequate or the Failure to Provide Fixed Suppression Systems in Accordance with 10 CFR 50 Appendix R, Section III.G.3, for Rooms, Areas or Zones Under Consideration.

(b) An inspection was made to determine if the fixed manual water spray systems in the cable spreading rooms and the automatic sprinkler system provided to protect the cable tray stacks at the east and west ends of the battery room were designed and installed in accordance with Unit 1 license Condition 2.C.(4) and Unit 2 License Condition 2.C.(7). These license conditions require the licensee to maintain in effect and fully implement all the provisions of the approved fire protection plan and the NRC staff's Safety Evaluation Report, Supplement No. 2.

Safety Evaluation Report, Supplement No. 2, indicates that the water suppression systems are designed in accordance with NFPA 13 and 15. In addition, this safety evaluation report required the cable spreading rooms to be protected by a manual fixed water spray system with a level of open spray heads at the ceiling and an additional level of heads below the lowest cable trays throughout the room. However, the present systems do not appear to be designed in accordance with NFPA 13 and 15 as the spray nozzles are not distributed uniformly throughout the cable spreading rooms at the ceiling level and at the level below the lowest cable trays. Therefore, it cannot be assured, that if an exposure fire were to occur in the cable spreading room, that the present manual fixed water spray system would assist in controlling a potential exposure fire and minimize fire damage to redundant trains of cabling.

The Safety Evaluation Report also indicated that the sprinklers installed in the battery room provide protection for the cable tray stacks at the east and west ends of the room from an exposure fire. The present battery room sprinkler system design does not comply with the guidance provided by NFPA-13. The present placement of the sprinkler heads under the cable trays is approximately 5 ft. above the floor. Without sprinklers placed at or near the ceiling, it

cannot be assured that the present sprinkler design would react in a timely manner to protect the cable tray stacks against the potential effects of an exposure fire.

These fixed suppression systems inadequacies have been identified as Unresolved Item (369/84-28-02 and 370/84-75-02), Inadequate Fixed Fire Suppression System Provided for the Cable Spreading Room and Battery Room, pending disposition by NRR.

b. Protection of Associated Circuits

The inspection was conducted to verify compliance with the associated circuit provisions of 10 CFR 50 Appendix R, Sections III.G and III.L. The emphasis was on the following areas of concern:

Common Bus Concern
Spurious Signal Concern
Common Enclosure Concern

(1) Common Bus Concern

The common bus concern is found in circuits, either non-safety or safety related, where there is a common power source with shutdown equipment and the power source is not electrically protected from the circuit of concern.

A number of circuits were identified which did not have adequate circuit breaker or fuse coordination. The licensee indicated that an ongoing breaker coordination program is in effect. The inspectors identified the following circuits which did not meet the requirements of Appendix R, Section III.G.2:

125 V. D.C. control power for charging pumps CCPA or CCPB from panels EVDA or EVDD respectively.

600 VAC power supply for auxiliary feedwater supply MOVs CA46B, CA50B, CA54AC and CA58A.

600 VAC Power supply for PORV block valves MOV1NC31B and MOV1NC35B

600 VAC power supply for RHR isolation valve MOV1NO1B

600 VAC power supply for turbine driven auxiliary feedwater pump suction valve CA7A

600 VAC power for nuclear service valve RN16B

600 VAC power for VCT outlet valves NV141A and NV142B

600 VAC power for component cooling pump suction valves form RWST NV221A and NV222B

The licensee should perform an analysis to ensure that power to hot standby equipment is protected from faults in commonly powered equipment. Presently, these conditions do not meet the requirements of Appendix R, Section III.G.2 and are identified as Violation Item (369/84-28-03 370/84-25-03), Failure to Provide Adequate Breaker/Fuse Protection for Equipment Required for Hot Standby.

(2) Spurious Signal Concern

A review of the licensee's spurious signal analysis was conducted to determine if the following conditions had been considered:

The false motor, control and instrument readings such as what occurred at the 1975 Browns Ferry Fire. These could be caused by fire initiated grounds, shorts or open circuits.

Spurious operation of safety-related or non-safetyrelated components that would adversely affect shutdown capability (e.g., RHR/RCS Isolation Valves).

The licensee intends to remove power and control voltages from valves that could affect safe shutdown of the unit should they operate due to a fire induced spurious signal. These are:

<u>Pressurizer Power Operated Relief Valves</u>
Power will be removed from the pressurizer power operated relief valves by removable disconnect cables in the 125 VDC control circuit for valves INC32B and INC36B and by an interlock relay for INC36A when the shutdown facility is to be activated.

<u>Reactor Head Vent Valves</u>
The head vent valves NC247B and NC275B will be inhibited from operation by the removal of cable disconnects in the 125 VDC control circuitry.

The head vent valves NC272,C and NC273A,C will have control capability transferred to the standby shutdown facility which is electrically isolated from the control room.

The RCS/RHR boundary valves were the only high low pressure interface which were identified and analyzed for spurious operation. There are installed, a number of interlocks in series in the control circuit for the RHR suction valve and none of them are

located in the same fire area. The valve opening circuit contains in series a control switch contact, a pressure interlock and valve limit switch interlock contacts. The limit switch 1SW27 is locked opened and consequently inhibits the ability to apply control power to the control circuits for the 1ND1E and 1ND2A RCS isolation valves.

These additional circuit analyses were reviewed by the inspectors:

RCS Boundary Valves and Centrifugal Pump Charging

The licensee conducted an analysis to determine the availability of a charging path to maintain reactor inventory. It was determined, at the time of the inspection, that the necessary modifications, such as one-hour fire retardant blanket wraps on circuits for the RHR/RCS valves ND2AC, were installed in the turbine driven auxiliary feedwater pump room.

Main Steam Isolation Valves (MSIV) and the Main Steam Power Operated Relief Valve (PORV)

The control circuits for main steam isolation valves 1SM1AB 1SM3AB, 1SM3AB, 1SM7AE were analysed. Two solenoid valves are installed in series in the pneumatic control line, the closure of any one will cause the MSIV to close. The control circuits for main steam power operated relief valves 15VIAB, 1SV7ABC, 1SV13AB, 1SV19AB were analyzed. Three solenoid valves are connected in series in the pneumatic control lines for the main steam power operated relief valves, the closure of any one will cause the respective main steam PORV to close. The McGuire Nuclear Station does not depend on the main steam PORVs for safe shutdown.

Dedicated instrumentation, electrically independent of control room, has been provided at the standby shutdown panel to monitor the following parameters:

Pressurizer Pressure
Pressurizer Level
Standby Makeup Pump Discharge Pressure
Steam Generator Level
Incore Temperature (T Hot)

The licensee has committed to install RTDs to monitor T Cold during the next refueling.

Source range instrumentation was not installed and an exemption request had been granted.

The spurious signal concern was not satisfactorily addressed since some buses feeding safe shutdown equipment did not have adequate coordination of circuit breakers and fuses. Some of the safe shutdown components which were affected by the lack of coordination were as follows:

Centrifugal Charging Pump CCPA
Centrifugal Charging Pump CCPB
Turbine Driven Auxiliary Feedwater Pump Suction Valve CA7A
PORV Block Valves - MOV INC31B and MOVING35B
RHR Isolation Valve MOV INDIE

This is another example of Violation Item (369/84-28-03 and 370/84-25-03), Failure to Provide Adequate Breaker/Fuse Protection for Equipment Required for Hot Shutdown.

(3) The Common Enclosure Concern

The common enclosure concern is found when redundant trains are routed together with a non-safety circuit which crosses from one raceway or enclosure to another, and the non-safety circuit is not electrically protected or fire can destroy both redundant trains due to inadequate fire protection means.

The common enclosure concern at McGuire was not a concern since the standby shutdown system cables were not run in the same trays with either the redundant trains or their associated non-safety-related cables.

For fires where redundant Trains A and B are to be used instead of the standby shutdown system, the licensee wrapped one redundant train. This was done in the Unit 1 Train B switchgear room where some Train A cabling was wrapped with 3-hour rated fire retardant blanket.

c. Dedicated Shutdown and Fire Damage Control Capabilities

(1) System Description and Operation

SSER No. 6, dated January 25, 1983, documents NRR review of the licensee dedicated shutdown system and its conformance to 10 CFR 50, Appendix R, Section III.L. The McGuire dedicated shutdown system is identified as the Standby Shutdown System (SSS). This system provides an alternate and independent means to achieve and maintain the reactor coolant system in hot standby condition for one or both units. The (SSS) is placed into operation only if a postulated fire results in the installed normal and emergency plant systems becoming inoperable. A masonry structure located adjacent to and outside the plant, houses the

major equipment and controls for the (SSS). This facility is designated the Standby Shutdown Facility (SSF) and consist of a diesel generator, starting batteries and supporting auxiliary systems. Normal electrical power is supplied via 6.9 kv site transformer, dedicated emergency power via the diesel generator, battery bank for 125 VDC, 600 VAC and 125 VAC power distribution systems, and a control panel for monitoring and controlling primary and secondary volumes. Reactor coolant system pressure control is provided by manual control of a bank of pressurizer heaters and pressurizer level via manual operation of reactor vessel head vent valves. Makeup water to the reactor coolant system and sealing water to the reactor coolant pumps (RCP) are provided by a 26 gpm makeup pump connected to the RCP sealwater injection line. Each standby makeup pump is located in the unit's containment building annulus and takes suction from the spent fuel pool transfer area. Steam generator volume control and decay heat removal are accomplished by utilizing the normal auxiliary feedwater system to maintain steam generator water level requirements. The main steam safety valves are used to control secondary side pressure and to dump steam to provide decay heat removal from the reactor coolant system.

The licensee gave a presentation on the general operator actions and the procedures to be used when the SSF/SSS is placed into operation for those cases where the fire renders the control room and the auxiliary shutdown panels inoperable. The procedures and sequence were described as follows:

- AP 1/A/5500/01, Reactor Trip

  This procedure provides the instructions to stabilize and control the plant following the trip of the reactor which would take place upon detection of a disabling fire.

- OP/O/A/6100/17, Operation of the Standby Shutdown Facility

  This procedure describes the use of the SSF/SSS systems, operational controls, and stations to be manned in order to maintain the unit or units in a hot standby mode.

- OP/O/A/6100/20, Operational Guidelines Following Fire in Auxiliary Building or Vital Area

  This procedure describes the steps to be taken and plant systems required to bring the plant to hot shutdown followed by cold shutdown within 72 hours. It is intended by the licensee to use normal operating and plant shutdown procedures to accomplish these evolutions.

- IP/O/A/3090/23, Fire Damage Control Procedure

    This procedure establishes methods for the restoration of instrumentation, electric motors, system valves, electric breakers, etc., following a fire so that plant equipment and systems necessary to bring the plant to cold shutdown conditions are made operable. This procedure is performed in conjunction with the aforementioned procedures in order to expedite efforts to take the plant to cold shutdown.

(2) Review of Operating and Surveillance Procedure

The inspectors reviewed the completed data in PT/A/4209/01C, Standby Makeup Pump Flow Periodic Test for Units 1 and 2 to verify that the output capacity for each pump met the 26 gpm minimum specified in the SSER No. 6. The test results for Unit 1 (completed 9/5/84), indicated a flowrate of 29.0 gpm and for Unit 2 (completed 8/31/84) a flowrate of 30 gpm which satisfies those values specified in SSER No. 6.

The inspectors reviewed those plant procedures identified for use in the case where a fire causes the control room to be abandoned and the auxiliary shutdown panels are also rendered inoperable. This review was conducted to verify that information in design and engineering documents and the information provided in SSER No. 6 have been factored into the appropriate plant operating procedures for the SSF/SSS systems and procedure for subsequent cold shutdown.

In addition to the procedure reviews, walkdown of selected procedures were conducted to ensure that the instruction provided was complete and usable; that steps identified components and equipment correctly and equipment is accessible to plant operators for operation.

The following plant procedures were reviewed:

- AP/1/A/5500/01 (Change 1), Reactor Trip
- OP/1/A/6100/07 (Change 8), Operation of the Standby Shutdown Facility
- OP/0/A/6100/20 (Change 0), Operational Guidelines Following Fire in Auxiliary Building or Vital Area
- OP/0/A/6100/13 (Change 0), Operational Guidelines Following Fire in Containment or Doghouse
- AP/1/A/5500/17 (Change 0), Loss of Control Room

The review and walkdown of the procedures resulted in the following findings and concerns, which apply equally to both units:

(a) OP/0/A/6100/17, Operation of the Standby Shutdown Facility

1) Provide appropriate steps in the procedure to ensure that pressurizer spray isolation valves are shut and the reactor coolant pumps are shutdown. SSER No. 6, Section C.3.5, describes these components as the means of terminating pressure decreases in the event, the spray valves become open due to electrical shorts caused by a fire

2) Enclosure 4.1, Section 1, Step 2.2, requires the normal power source for SSS system equipment to be switched to its alternate power source, MCC-SMXG. Appropriate steps need to be added to the procedure that ensure that either the diesel generator is supplying power to the alternate bus or the offsite power supply is available.

3) Enclosure 4.1, Section 1, Step 2.4, Note: Requires obtaining a master key to gain access through fire doors. This step needs to be moved to the front of the procedure as preoperations and made readily available so as not to delay the detailed procedure steps.

4) Enclosure 4.1, Section 2, Step 2.0, second Note: Specifies maintaining the spent fuel water level per OP/1/A/6200/05.

   Appropriate precautionary measures need to be added to the procedure to ensure that the makeup water added to the spent fuel pool is from the refueling water storage tank or other suitable source. In addition, the boron concentration is required to be equal to or greater than 2000 ppm. The standby makeup pump suction is from the spent fuel pool.

5) Enclosure 4.1, Section 4, Step 2.2. Provide instructions for converting the incore thermocouple digital readout to core temperature.

6) Enclosure 4.1, Section 2, Step 2.2 Note: This note needs to be expanded and clarified so that the operator knows what to look for when monitoring the standby makeup pump D/P filter gauge.

   The above items collectively are identified as Inspector Followup Item (369/84-28-13 and 370/84-25-13), Amplify and Clarify Certain Steps of OP/0/A/6100/17.

7) Enclosure 4.1, Section 3, Step 2.1, states to verify that the auxiliary feedwater pumps start when Lo-Lo level is reached on 2/4 steam generators. Steam generator levels are maintained by dispatching an operator to the auxiliary feedwater pump area and position manual valves to control levels. The inspector expressed a concern as to whether having three auxiliary feedwater pumps feeding the steam generators, with the unit in hot standby and natural circulation in progress would be excessive. The inspector requested that an analysis be performed to determine if SSF/SSS operations would be jeopardized regarding maintaining hot standby conditions, primary, and secondary volume control. The parameters of specific interest that need to be addressed are: effects on cooldown rate, amount of reactor coolant system temperature decrease, the effects on pressurizer level (amount of shrinkage), overfill of steam generators, and effects on reactivity shutdown margin. The time for an operator to be dispatched to the auxiliary feedwater pump area, and conduct operations to gain level control need to be considered in the analysis. These concerns are identified as an Unresolved Item (369/84-28-14 and 370/84-25-14), Analysis Effects on SSF/SSS Operation, Potential Excessive Feeding Steam Generators.

(b) AP/1/A/5500/01, Reactor Trip;
AP/1/A/5500/17, Loss of Control Room; and
OP/0/A/6100/20, Operational Guidelines Following Fire in Auxiliary Building or Vital Area

As a result of reviewing and conducting walkdowns of the above procedures, and holding discussions with licensee personnel, it appears that the overall coordinated effort for a smooth departure from the control room to the SSF/SSS to maintain plant hot standby conditions is fragmented, not well defined, and may lead to confusion, delays, and possible errors.

The procedures as presently written has AP/1/A/5500/17, Loss of Control Room, as the controlling document for establishing systems and plant conditions for leaving the control room and then going to the auxiliary shutdown panels for continued plant control to eventually cold shutdown. The licensee has stated, however, that when a fire disables the control room, it will also make the auxiliary shutdown stations not available for use. Based on these conditions, the inspectors have the following findings and concerns:

1) Review and correct the Initial Conditions specified in OP/0/A/6100/17. Some of these conditions need to be made procedure steps in order to establish the initial conditions since AP/1/A/5500/17 will not be used.

2) Establish the necessary procedure steps to place the plant and equipment in a stable hot standby which will permit an orderly departure from the Control Room and the use of the Standby Shutdown System.

3) Establish the necessary procedure steps to take the plant from hot standby to cold shutdown. These procedures will address shutdown from the control room and shutdown from outside the control room.

4) Conduct final walkdowns of procedures giving particular attention to procedural adequacy, access to system components, and verify communications are satisfactory.

These concerns and findings were discussed with the licensee on October 1, 2, and 3, 1984. Subsequent to these discussions, a Confirmation of Action Letter dated October 9, 1984, was issued which identified October 5, and October 19, 1984, as dates for completing corrective actions on procedures. The inspector identified this area as Unresolved Item (369/84-28-15 and 370/84-25-15), Correct Procedure Deficiencies Used to Accomplish 10 CFR 50, Appendix R, Section III.L.

(c) Personnel Training

The inspectors held discussions with training department instructors to determine that training is being given concerning the operation and use of the SSF/SSS. It was determined that personnel who are receiving training included senior reactor operator (SRO), reactor operator (RO), and nuclear equipment operators (NEO). The licensee has scheduled training for the required personnel. The inspectors reviewed the licensee's lesson plans, training schedules, examination questions, and found these documents to be well organized, detailed, and comprehensive.

(d) Fire Damage Control Procedures

The inspectors reviewed McGuire Nuclear Station Fire Damage Control Procedure No. IP/0/A/3090/23. The purpose of this procedure is to establish a method of making the diesel generators operable, controlling 4.16kv breakers, installing power and control cables to certain 4.16kv motors, installing instrumentation and restoring valve operability after a fire in order to place the plant in cold shutdown status. The procedure identified the motors, valves and instrumentation required to bring the plant

to cold shutdown that would require restoration if damaged by fire.

The procedure identifies the types of power cables required for the identified motors. Included in the procedures are instructions for preparation of stress cones, instructions and routing for the installation of temporary power and control cables and clearing any control cables that may be faulted due to fire damage. This procedure provides for the manual operation of 4.16kv breakers after disconnecting certain remote control conductors. Instructions are included for replacing certain electronic sensing instrumentation (level transmitters, flow transmitters, pressure transmitter, etc.) with direct heading gauges. While drawings are provided as part of the procedure, instructions require that the latest revision of the drawings be used to perform the work.

The inspector examined a large box containing the required number of stress cone kits needed to restore all the 4.16kv motor power cables that were identified in the procedure. Included in this box were the necessary gauges required for instrumentation restoration along with various lengths of sensing line and fittings. The licensee had designated nine reels of cables to be used for the power cable replacement. The box of materials was tagged to indicate the intended use and stored in the warehouse. The cable reels had not yet been located in a designated area.

The inspector examined the designated routes for the replacement power cables for the centrifugal charging pump motors, the residual heat removal pump motors and the component cooling pump motors. The routes were found to be practicable and all areas were accessible for cable pulling. It appears that the cables could be installed with off-site power not available to power electric pulling equipment but would require additional personnel.

6. Compliance With 10 CFR 50 Appendix R, Section III.O

The inspector(s) reviewed the as-built documentation/drawing file of the oil collection system for the reactor coolant pumps.

Potential oil leakage points for each pump have been provided with a Westinghouse designed and furnished RCP oil containment system consisting of a upper bracket oil overflow enclosure, lower bearing oil catcher, oil lift enclosure, and oil cooler enclosure. These enclosures are connected to drain piping which discharges into a separate collection tank provided for each reactor coolant pump. The reactor coolant pump oil collection system was originally designed and installed at this facility prior to the issuance of Appendix R. In a letter dated January 9, 1981, to the NRC, the licensee acknowledged their previous commitments to provide the oil collection system, but noted that the RCP oil containment system and its associated drainage system would require additional analysis to verify compliance with

the requirements of Appendix R, Section III.O. The analysis was completed and the oil containment system and related drain piping were seismically upgraded and modified to function following a design base seismic event. This upgrade was performed under the licensee's seismic quality assurance (QA) program as verified by a review of the following records:

a. Seismic Analysis:

- Westinghouse Electric Corporation, Engineering Report Memorandum 5802, dated September 10, 1982
- Duke Power Memorandum, File MC-1435.03, MC-1223.03, MC-1415.00, MC-1421.00, dated February 17, 1982
- Flow diagrams MC-1553-4.0 and MC-2553-4.0

b. Upgraded Seismic Hanger Inspection:

Hanger Number	Design Isometric Drawing	Date of Final QC Inspection
1-MCR-NC-2342	MCSRD-SPM118/144D	11/16/82
1-MCR-NC-2269	MCSRD-SPM118/144A	02/21/83
1-MCR-NC-2303	MCSRD-SPM118/144C	11/13/82

Since both units were operating at full power during the inspection period, the inspector(s) were unable to review the reactor coolant pump oil collection system's installation for conformance to the design requirements. This review will be made during a subsequent NRC inspection when the units are in a refueling outage. Within the areas reviewed, no items of noncompliance or deviations were identified.

7. Compliance With 10 CFR 50, Appendix R, Section III.J

Section III.J, requires that: "Emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto."

a. Emergency Lighting System Walkdown

The inspector(s) performed a walkdown examination of the design and installation of the 8-hour emergency lighting units for the facility based upon the licensee design drawings, and the post-fire alternate shutdown procedures utilizing the standby shutdown system, OP/O/A/6100/17, Operation of the Standby Shutdown Facility, and AP/1/A/5500/17, Loss of Control Room. Also, the battery powered lighting units for several plant areas were observed while energized to determine the lighting beam direction and relative illumination levels.

As a result of the emergency lighting walkdown, it appears that inadequate emergency lighting conditions existed in the following plant areas:

(1) Several lights in the Unit 1 doghouse were mounted behind concrete columns, piping, etc., which eliminate their effectiveness to illuminate access ladders to safe shutdown related SM and SV valves.

(2) No 8-hour battery powered lighting units were provided for Units 1 and 2, Common Corridor 908 which provide a portion of the access/egress routes between the main control room and the standby shutdown facility.

The failure to meet the emergency lighting requirements of 10 CFR 50, Appendix R, Section III.J is identified as Violation Item (369/84-28-04 and 370/84-25-04), Failure to Comply with the Requirements of 10 CFR 50, Appendix R, Section III.J.

In addition, the licensee requested exemptions from the emergency lighting requirements for several plant areas in letters dated November 18, 1983, and February 20, 1984. These letters requested exemption from the requirement to provide 8-hour battery powered emergency lighting units in the standby shutdown facility and along a portion of the yard area access route between the standby shutdown facility and the turbine building. NRR granted the above exemption based upon the licensee's commitment to place battery-powered hand lanterns in the control room to be used in emergency situations by the plant operators. During the plant walkdown, an inspection was made to determine if the battery powered hand lights were provided. Contrary to the licensee's commitment the hand lights had not been installed in the control room. However, after this item was identified, the licensee took immediate corrective action and provided, for emergency use, two battery powered hand lanterns within the control room. This is identified as Deviation Item (369/84-28-05 and 370/84-25-05), Failure to Provide Battery Powered Hand Lanterns in the Control Room.

b. Evaluation of Emergency Lighting Testing Procedures

The inspector(s) reviewed the licensee's periodic opeational testing procedures (PT/1/B/4350/09, Unit 1 Emergency Lighting Annual Test; PT/2/B/4350/09, Unit 2 Emergency Lighting Annual Test), and completed tests records for 1983 and 1984 on the emergency lighting system. And as a result of this review, the following discrepancies were identified:

(1) The emergency lighting annual test procedures did not address verifying whether the light beam is pointed in the correct direction to illuminate all proper pieces of equipment, electrical cabinets, enclosures, or access/egress path-ways required for emergency operation.

(2) The procedures did not provide for immediate corrective actions or compensatory measures to be taken for deficiencies found during performance testing of the periodic operational test.

(3) The procedures did not provide the testing frequency and scope of performance testing in accordance with the manufacturer's recommendations which:

- Require the test button be depressed for at least one to two minutes to assure verification of AC to battery DC transfer capability and adequate electrical power drain from the battery to verify operation of the battery charging circuit of the battery charging circuit, and

- Require a specific testing frequency for a particular manufacturer's lighting unit. The emergency lighting testing frequency at McGuire is greater than that recommended by the manufacturer. From the review of the test records, it appears that the test period for all emergency battery powered lighting units ranged from two months to almost seven months duration for both units. Specific dates that individual battery powered lighting units were tested are not recorded making testing frequency of the individual lighting units indeterminate.

(4) The procedures did not include or reference capacity of the 8-hour self-contained battery packs nor scheduled battery pack replacement.

Based upon the deficiencies listed above, the 8-hour emergency lighting system design, performance testing, and surveillance and maintenance programs appear to be inadequate for ensuring that such a system is always operational to enable operators to transfer control to, and operate the SSF and SSS functions. These findings are identified as Inspector Followup Item (369/84-28-06 and 370/84-25-06), Inadequate Surveillance Testing Procedures for Emergency Lighting, and will be reviewed during a subsequent NRC inspection.

8. Fire Protection and Prevention Program

a. Fire Protection for Safety-Related Pumps

The Unit 1 license condition 2.C.(4) and Unit 2 license condition 2.C(7) requires the licensee to fully implement and maintain in effect all provisions of the approved fire protection plan. The McGuire Nuclear Station Fire Protection Review, September 1982 Revision, Section F.11 - "Safety-Related Pumps", indicates that redundant safety-related pumps are separated by required fire barriers and automatic detection with alarm and annunciation in the control room. An inspection was made to determine if this license condition was fully

implemented with regard to section F.11 of the approved fire protection plan.

The following safety-related pumps have been identified as not being separated by a fire barrier:

- Recycle Evaporator Feed Pumps, Room 620
- Waste Drain Tank Pumps, Room 639
- Boron Injection Recir. Pumps - Unit 2, Room 788
- Boron Injection Recir. Pumps - Unit 1, Room 730
- Fuel Pool Cooling Pumps - Unit 1, Room 816
- Fuel Pool Cooling Pumps - Unit 2, Room 829

The following safety-related pumps have been identified as not being provided with automatic fire detection capabilities:

- Recycle Evaporator Feed Pumps, Room 620
- Waste Drain Tank Pumps, Room 639
- Fuel Pool Cooling Pumps - Unit 1, Room 816
- Fuel Pool Cooling Pumps - Unit 2, Room 829

The failure to fully implement the provisions of the approved fire protection plan as required by the operating license is identified as Violation Item (369/84-28-07 and 370/84-25-07), Failure to Provide Automatic Fire Detection for and Fire Barriers to Separate Safety-Related Pumps.

b. Hydrogen Piping Systems

A review was made of the hydrogen gas piping systems to the volume control tanks and reactor coolant drain tanks. The hydrogen for the volume control tanks is supplied from the bulk gas storage located in the plant yard. The hydrogen to the reactor coolant drain tanks is supplied from two H2 cylinders also located in the bulk gas storage structure outside in the plant yard. The following documentation was reviewed to verify that the systems were seismically supported.

<u>Flow Diagram No.</u>

- MC-1603-1.0
- MC-1565-1.1
- MC-2565-1.1, and
- Duke Power Memorandum W. H. Taylor, Jr. to P. R. Herran, dated September 27, 1984, McGuire Nuclear Station, W. L. System Upgrade of Piping to Class F, File MC-1206.02.88

The hydrogen gas piping to the volume control tanks inside the auxiliary building is designed and installed as Duke Class F (ANSI B31.1-Seismic loading). Hydrogen piping to the reactor drain tanks extends from the yard through the turbine building, service building,

auxiliary building and reactor building through the drain tank. The piping at the entrance to the auxiliary building to the reactor building containment isolation boundary is Duke Class F, Class C and Class B respectively, designed for seismic loading.

As noted in the above Duke memorandum, the piping (approximately 43 inches in length) from containment isolation valves 1WL39A (Unit 1) and 2WL39A (Unit 2) to the reactor coolant drain tanks has been modified to seismically qualified, Duke Class F. The supports are designed with seismic loads per calculation MCC-1206.12-05-2014 (Unit 1) and MCC1206.16-27-3104 (Unit 2). Based upon the above review, the hydrogen piping within safety-related plant areas appears to meet the general design requirements.

c. Fire Pumps

During the inspection, a question arose regarding the availability of the three fire pumps for fire suppression activities in certain instances. Fire pumps A and B are powered from 6900 VAC non-safety load centers 2TB and 1TD respectively, and fire pump C is powered from a 44kv substation separate from the plant switchyard.

Appendix R requires the plant to recover from the effects of a fire and be in cold shutdown in 72 hours with the loss of offsite power (LOP). An LOP at McGuire could render all three fire pumps inoperative, in that none of the fire pumps are fed from the emergency diesel generators, which would greatly hamper fire suppression efforts. Assuming that LOP is caused by grid instability, pump C could also be lost even though it is fed from a dedicated off-site 44kv line which is separate from the station switchyard.

A review of the Technical Specifications for McGuire Nuclear Station revealed another area of concern. Section 3/4.7.10, Fire Suppression System, requires that at least two fire suppression pumps be operable at all times. There is no requirement as to which pumps must be operable. Therefore, pump C could be inoperative for an unlimited amount of time as long as pumps A and B are inservice/available. Pumps A and B would be lost on the LOP and with pump C out of service the plant would have very limited fire suppression capability. Since Appendix R requires that the LOP to be considered concurrent with the fire event, it is conceivable that no fire pumps would be available to support fire fighting activities.

This item is being reviewed by the NRC staff and is identified as Unresolved Item (369/84-28-08 and 370/84-25-08), Adequacy of Power Supplies for Fire Pumps A, B and C.

d. Emergency Communications

Post Fire Alternate Shutdown Procedure OP/0/A/6100/17, Operation of the Standby Shutdown Facilities, requires that communications must be established between the various areas of the plant where local control actions must be taken. This procedure identifies that portable radios can be utilized as one method of establishing communications. By letter dated December 14, 1982, the licensee responded to NRR's questions concerning the standby shutdown system. The licensee's response to questions O and P states that portable radios will be available for communications between the standby shutdown facility and the auxiliary feedwater local control stations.

An inspection was made which evaluated the adequacy of the radio communications by actually establishing radio communications between the various local control stations (i.e., Units 1 and 2 Train "A" switchgear rooms, Unit 1 doghouse, and Units 1 and 2 Steam Driven Auxiliary Feedwater Pump Rooms) and the Standby Shutdown Facility. However, direct radio communications were not feasible between the standby shutdown facility and the auxiliary feedwater local control stations and other essential control stations required for the operation of the standby shutdown system. This item is identified as Deviation Item (369/84-28-09 and 370/84-25-09), Failure to Provide Adequate Radio Communications Between Local Control Stations and the Standby Shutdown Facility.

9. Licensee Identified Items

a. June 1, 1984 Nonconformance Report, Potential Fire Damage to Unit 2 Turbine Driven Auxiliary Feedwater Pump Suction Valves: On May 18, 1984, the licensee discovered that the location and protection of auxiliary feedwater pump suction valves 2CA-161C and 2CA-1626 were not in accordance with previous commitments made to the NRC and were in nonconformance to 10 CFR 50, Appendix R, Section III.G. These valves are required to open automatically to align a long term source of water supply to the suction of the turbine driven auxiliary feedsater pump (TDAFP) in case of fire in the adjacent motor driven auxiliary feedwater pump (MDAFP) room. A fire within the MDAFP room could have damaged the two MDAFPs and caused damage to the TDAFP suction valve operators and/or associated cables thus eliminating the capacity for automatic alignment to the long term water source. However, a postulated fire in the MDAFP room of sufficient intensity to damage both MDAFPs and the automatic switchover capacity of valves 2CA-161C and 2CA-162C was unlikely due to the light combustible fire loading and the availability of the early warning automatic fire detection system and the automatic wet pipe sprinkler system within the room. Furthermore, the normal suction source to the TDAFP would be available to provide a water source for a minimum of 4½ hours. This would probably have allowed adequate time for the fire brigade to extinguish the fire and operators to manually realign the valves if necessary.

This nonconformance was identified by the licensee and promptly reported to the NRC on May 18, 1984. Also, as noted above the deficiency did not present a significant threat to the health and safety of the public. Therefore, since this discrepancy meets the guidelines of 10 CFR 2, Appendix C, Section IV.A, for licensee identified problems, no violation is being issued.

The licensee's corrective action consisted of the installation of a one-hour fire barrier enclosure for valves 2CA-161C and 2CA-162C and associated cables. This arrangement brought this area up to meet the provisions of Appendix R, Section III.G.2, except the structural steel supports for TDAFP piping and cables to operators for valves 2CA-161C and 2CA-162C are not protected to provide a fire resistance equivalent to that of the one-hour fire barriers. This does not meet the requirements of 10 CFR 50, Appendix R, Section III.G.2, as required by License Section 2, Item C.7.a, and is identified as an example of Violation Item (369/84-28-10 and 370/84-25-10), Structural Steel Fire Barrier Supports Not Provided with Fire Resistant Rating Equivalent to the Fire Barrier.

b. August 2, 1984 Nonconformance Report, Potential Fire Damage to Redundant Safe Shutdown Equipment and Cabling in Various Fire Areas of Units 1 and 2. On July 18, 1984, the licensee discovered a number of areas in Units 1 and 2 which were in nonconformance to commitments made to the NRC and to the requirements of 10 CFR 50, Appendix R, Section III.G. These discrepancies were promptly reported to the NRC on July 18, 1984, and with a followup written report sent to Region II on August 2, 1984. The following items were identified:

(1) Two suction valves which are arranged to open automatically to align a long term source of water supply to the suction of the Unit 1 TDAFP and associated cabling to the valve operators are located within a Unit 1 pipe chase and mechanical penetration room. Redundant components and the cabling for the normal plant shutdown systems of centrifugal charging pumps 1A and 1B are also located within this same fire area. A postulated fire in this area could have incapacitated portions of all safe shutdown trains and plant shutdown would have been difficult to obtain. The room was provided with automatic fire detectors but a fire suppression system was not provided, the area is not readily accessible for manual fire fighting operations, and is in a potentially high radiation area.

The failure to meet the separation requirements of 10 CFR 50, Appendix R, Section III.G, for this area as required by the operating license is identified as Violation Item (369/84-28-11), Inadequate Appendix R, Section III.G, Fire Protection Features and Separation Provided for Redundant Trains of Normal Shutdown Systems and the Standby Shutdown System.

The licensee maintained an hourly fire watch patrol for this room until the TDAFP cabling and valve operators were enclosed in a three-hour fire rated barrier. This barrier was reviewed by the inspectors and appeared satisfactory, except the structural steel supports for valves, piping and cabling to the valve operators were not provided with a three-hour fire resistant rating as required by Appendix R, Section III.G.

(2) Unit 1 Train A associated control circuits and the standby shutdown system (SSS) cables are both located in Train B switchgear room. None of these cables were enclosed in a three-hour fire barrier. A fire detection system is provided for this area but a fire suppression system is not provided. A postulated fire within the Unit 1 Train B switchgear room could have damaged control cables for Train A centrifugal changing and auxiliary feedwater pumps, control and power cables for the standby makeup pump, and control and power cables for Train B centrifugal changing and auxiliary feedwater pumps. This could have prevented safe plant shutdown.

The failure to meet the separation requirements of 10 CFR 50, Appendix R, Section III.G, for this area as required by the operating license is identified as Violation Item (369/84-28-11), Inadequate Appendix R, Section III.G, Fire Protection Features and Separation Provided for Redundant Trains of Normal Shutdown Systems and the Standby Shutdown System.

The licensee maintained an hourly fire watch for this area until the normal Train A shutdown component cabling within Train B switchgear room was enclosed within a three-hour fire barrier. This barrier was reviewed by the inspectors and appeared satisfactory.

(3) Control and cabling for the MDFPs components and the other Train A and B safe shutdown components including the centrifugal changing pumps are routed through the Units 1 and 2 TDAFP rooms. The TDAFP rooms are provided with fire detection systems and automatic halon fire suppression systems. However, the cabling in this area was not enclosed within a one-hour fire barrier as required by Appendix R, Section III.G. A postualted fire within one of the TDAFP rooms of sufficient intensity to damage the shutdown components to prevent plant shutdown was not probable due to the low fire loading within the rooms, early warning fire detection system, and automatic halon fire suppression system.

This discrepancy was promptly reported to the NRC after being identified by the licensee on July 18, 1984, and with a written report sent to Region II on August 2, 1984. An hourly fire watch was initiated and is to be maintained until the controls and cabling associated with the MDAFP suction valves are removed from the Units 1 and 2 TDAFP rooms and/or the Train B shutdown cabling

within the TDAFP rooms are enclosed within a one-hour fire barrier. Based on the above, this discrepancy did not present a significant threat to the health and safety of the public, was identified by the licensee, and appropriate corrective action was initiated. Therefore, this discrepancy meets the guidelines of 10 CFR 2, Appendix C, Section IV.A, for licensee identified problems and no violation is being issued.

c. August 3, 1984 Appendix R Deviation Notice:

During the licensee's ongoing fire protection program review, several deviations from Appendix R were identified. These deviations and technical justifications were forwarded by the licensee to NRC/NRR on August 3, 1984. These deviations as listed below will remain outstanding pending NRR evaluation:

(1) Steel Penetrating Fire Barriers

(a) The 1½ hour fire barriers between redundant nuclear service water pumps and component cooling water pumps are penetrated by cable tray hangers.

(b) The 3-hour fire walls separating the TDAFP and MDAFP room are penetrated by steel pipe supports and restraints.

(2) Reactor Building Wall Penetrations

(a) Process piping penetrations in reactor building are designed for pressure boundary integrity in lieu of fire boundary penetrations.

(b) Spare sleeves and instrument tubing penetrations are sealed by steel plate or pipe cap on the auxiliary building side of sleeve.

(c) HVAC duct penetrations do not have fire dampers.

(d) Access into the reactor building from the auxiliary building is provided by two portals which have not been fire tested.

(3) Fire Boundary Doors With Security Hardware

Fire doors have been modified to meet security requirements and some fire walls have security and other special type doors in lieu of standard fire doors.

(4) Cork Expansion Joints

Cork has been provided in the structural joints between some structures for seismic considerations. However, this configuration has not been tested for three-hour fire resistance rating.

The above items do not technically meet Appendix R requirements and other NRC guidelines. Therefore, pending NRR evaluation, these items are identified as Unresolved Item (369/84-28-12 and 370/84-25-12), NRR Evaluation of Appendix R Deviation Request of August 3, 1984, and will be reviewed during a subsequent NRC inspection.