UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-259/95-37, 50-260/95-37, and 50-296/95-37

Licensee: Tennessee Valley Authority
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1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-259, 50-260, and 50-296

License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry Nuclear Power Station Units 1, 2, and 3

Inspection Conducted: July 17-21, 1995

Inspector: for G. Wiseman, Reactor Inspector — 8/30/95 Date Signed

Accompanying Personnel: P. Fillion, Reactor Inspector
R. Musser, Resident Inspector
A. Fresco, Brookhaven National Laboratory

Approved by: for M. B. Shymlock, Acting Chief — 8/30/95 Date Signed
Test Programs Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This special announced inspection was conducted in the area of safe shutdown to evaluate Browns Ferry Units 2 and 3 for compliance with Sections III.G, III. J, and III. L, of Appendix R to 10 CFR Part 50. Compliance with the fire barrier and protection related features of III.G will be evaluated during future inspection.

The basis for this inspection was the Browns Ferry Nuclear Plant Fire Protection Report (FPR) submitted by the licensee on December 20, 1994. At the time of this inspection, the licensee's FPR had not been reviewed or approved by NRC.

Enclosure

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The inspection primarily focussed on Unit 3 plant equipment and systems identified by the licensee to achieve and maintain safe shutdown conditions.

Results:

In the areas inspected, no violations or deviations were identified.

* The inspectors verified that the plant's Appendix R safe shutdown capability had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions. The inspectors concluded that the identified changes in the Safe Shutdown Analysis provided an overall improvement to the Appendix R safe shutdown capability for Unit 3. Subject to further review by NRR, the safe shutdown capability was found acceptable.

* All cable and circuit separation configurations observed were found to be acceptable. The inspectors determined that the separation of those systems necessary to achieve safe shutdown satisfied the separation requirements of Section III.G and III. L, of Appendix R to 10 CFR Part 50. Compliance with the fire barrier and protection related features of Section III.G of Appendix R will be evaluated during future inspections.

* The inspectors performed a sample review of the electrical distribution system buses for the residual heat removal system in the low pressure coolant injection mode of operation and verified the protective device coordination through review of the design studies and a plant walk down check of the actual settings on the relays and breakers. The inspectors concluded that proper coordination existed. However, it was noted that two (2) modifications related to the circuits remain to be completed.

* The inspectors verified the remote safe shutdown isolation capability of Reactor Core Isolation Cooling system suction valves 3-FCV-71-17 and 18 for Fire Area 16 (Control Building). The inspectors concluded that the circuit design would achieve the desired isolation and annunciation was provided in the control room should the switches be in the emergency position during normal plant operation.

* The inspectors performed a sample review of electrical protection of non-essential circuits routed with required safe shutdown circuits. Calculations were reviewed to verify that cables at Browns Ferry Nuclear Plant were protected from insulation damage such that its insulation will not exceed its cable insulation damage temperature due to the protection criteria for the surrounding cables. The inspectors concluded that the majority of the cables at Browns Ferry fully meet the cable insulation damage criteria. All faults having sufficient energy to be cleared by the magnetic portion of the circuit breaker, will be cleared without exceeding the cable insulation damage temperature.

* The inspectors observed the Appendix R, Section J., emergency lighting units provided in the control room and the operator access pathway to the remote shutdown panel room (Electrical Board Room 3A). The inspectors conducted a "lights out" test to verify the battery unit emergency lighting levels were adequate for reading a procedure and identify panel nameplates. The results were satisfactory.

* The inspectors verified that the content of the alternate safe shutdown procedure for Fire Area 16 (Control Building) was adequate to implement safe shutdown from outside the zone/area of concern.

Two new Inspector Follow-up Items were identified.

Inspector Follow-up Item 50-296/95-37-01, Verification of Emergency Lighting Levels (Section 5.0).

Inspector Follow-up Item 50-296/95-37-02, Performance of Simulated Shutdown for an Appendix R Event (Section 6.0).

REPORT DETAILS

1.0 Persons Contacted

*D. Burrell, Lead Electrical Engineer
*C. Crane, Assistant Plant Manager
*T. Dexter, Training Manager
*R. Gilbert, Operations Procedures
*J. Gomez, Principal Electrical Engineer, Appendix R
*R. Kerstetter, Operations Procedures
*R. Machon, Site Vice President
*P. Salas, Licensing Manager
R. Sampson, Electrical Engineer
*T. Shriver, Nuclear Assurance and Licensing (NA&L) Manager
D. Stewart, Mechanical/Nuclear Engineer
*D. Stinson, Unit 3 Recovery Manager
*S. Wetzel, Compliance Manager
*H. Williams, Engineering and Materials Manager

Other licensee employees contacted during this inspection included technicians, engineers, and administrative personnel.

NRC Employees

*J. Munday, Resident Inspector
*R. Musser, Resident Inspector
*M. Lesser, Acting Branch Chief, Division of Reactor Projects
*Denotes those individuals that attended the exit meeting.

Acronyms and abbreviations used in this report and its attachments are listed in section 8.0.

2.0 COMPLIANCE TO 10 CFR, APPENDIX R, SECTION III.G. (IP 64100)

Safety Evaluation Reports of April 1988, (NUREG-1232, Vol.3), October 1989, (NUREG-1232, Vol.3, Supp. 1) and January 1989, (NUREG-1232, Vol.3, Supp 2) evaluated the Browns Ferry Appendix R fire protection program and the Browns Ferry Nuclear Performance Plan for the Browns Ferry Unit 2 restart. Sections 3.1 and 3.11.4 of Supplement 2 to the SER documented the NRC evaluation of the Browns Ferry cable separation issues and Appendix R safe shutdown capability for Unit 2 restart.

Appendix R safe shutdown was inspected for Unit 2 restart and documented in inspection reports (IR) 50-260/89-13 and 50-260/89-28 to address Appendix R safe shutdown. The results of these inspections were that the implementation of the cable separation program was adequate and Browns Ferry complied with Appendix R to 10 CFR Part 50 for the restart of Unit 2.

In correspondence dated January 9, 1991, the licensee detailed their plans for the return of Unit 3 to service. In this correspondence they identified Fire Protection/10 CFR 50, Appendix R as a program for Unit 3 restart that would be implemented under criteria which was different from the precedent established during the Unit 2 restart.

By cover letter dated December 20, 1994, the Tennessee Valley Authority (TVA) submitted its revised combined Fire Protection Report (FPR) for Units 2 and 3 of the Browns Ferry Nuclear Plant to the NRC for review. The FPR was prepared to document compliance of Units 2 and 3 with the fire protection requirements contained in Sections III.G, Safe Shutdown Capability; III.J, Emergency Lighting; and, III.L, Alternate Shutdown Capability, of Appendix R to 10 CFR 50.

Therefore, this inspection was conducted to determine if the Appendix R separation of the identified "Safe Shutdown Systems" (SSDS) utilized to shutdown the units during and after a fire was sufficient to comply with the requirements of Appendix R to 10 CFR 50.

2.1 REVIEW OF CHANGES TO THE SAFE SHUTDOWN CAPABILITY

The licensee's FPR included a plant Appendix R Safe Shutdown Analysis (SSA) that demonstrated that sufficient system separation existed for systems needed for safe shutdown during and after a fire. The SSA identified the SSDS that included those components, cabling, and support equipment needed to achieve safe shutdown for a fire event. The inspectors reviewed the SSA to verify that the plant's shutdown capability had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions.

The inspectors questioned a number of aspects related to the plant's safe shutdown capability. Specifically, a concern was noted in the SSA summary paragraph for Fire Area 20. The paragraph did not identify the availability of the Reactor Core Isolation Cooling system (RCIC) for safe shutdown. The concern was discussed with the licensee. The licensee indicated that the RCIC system was available for the safe shutdown of Unit 3 in Fire Area 20, and that a change to the FPR had been generated (PUNCHLIST ITEM NO. REC-0211) to reflect the system's availability in the summary paragraph. The inspectors determined that this action appropriately addressed their concern.

Based on their review of the SSA, the inspectors assembled an Appendix R safe shutdown matrix that identifies the availability of high pressure make-up coolant systems and other major SSDS equipment necessary to achieve and maintain safe shutdown conditions. This matrix is included as Attachment 1 to this inspection report.

From this review the inspectors determined that the safe shutdown capability was basically the same since the issuance of the NRC SERs for Unit 2 restart. Where available, a high pressure make-up coolant system will be used to maintain vessel inventory for the first two hours of the event, followed by depressurization of the vessel by use of the main steam relief valves (MSRVs) and use of the residual heat removal (RHR) system for establishing cold shutdown via the low pressure coolant injection (LPCI) shutdown cooling mode and the residual heat removal service water system (RHRSW). However, some changes had been made to the SSA for the operation of Unit 2 and Unit 3.

The specific changes to the safe shutdown capability are:

(1) Use of the RCIC system as an additional high pressure coolant makeup system for the first two hours of a fire event for Unit 3.

(2) Assuring the availability of four (4) MSRVs for Unit 3.

(3) Deletion of drywell temperature instrumentation for Unit 3.

(4) Reactor Water Cleanup (RWCU) system automatic isolation on high temperature for Unit 3.

The inspectors evaluated the specific changes (1), (2), and (4) above.

(1) High Pressure Coolant Make-up Systems

A high pressure coolant make-up system [either High Pressure Coolant Injection (HPCI) or RCIC] will be available for Unit 3 safe shutdown with a postulated fire in all but three fire areas. These areas are Fire Area 3 [Unit 3 Reactor Building (RB)]; Fire Area 12 [Unit 3 480 V Reactor Motor Operated Valve (RMOV) Board Room 3B]; and, Fire Area 13 [Unit 3 480 V RMOV Board Room 3A].

A postulated fire in these three areas represent the worst case safe shutdown scenario. This involves the availability of the SSDS which include only the manual control of the MSRVs, RHR systems in the LPCI mode and the RHRSW system. During depressurization there would be a short term uncovery of the reactor core with a resultant increase in the temperature at the top of the fuel prior to reflooding with RHR/LPCI. The licensee requested and was granted an exemption for temporary condition (October 21, 1988).

The availability of the high pressure coolant makeup systems would be desirable from the standpoint of minimizing the severity of reactor thermal-hydraulic transients described above. The Condensate Storage Tank (CST) would be the preferred water supply for the HPCI system and RCIC system except for a fire in Fire Area 25 which contained cabling for CST suction valve 3-FCV-2-166 resulting in valve closure and loss of suction. In this case, the redundant supply of water for the CST would be from the torus.

The inspectors examined a sample of Unit 3 high pressure coolant make-up systems for proper separation to ensure the systems would be available in the event of an Appendix R fire. The areas reviewed are discussed in Section 3.0 of this inspection report.

(2) Main Steam Relief Valves

The MSRVs are required for over pressure protection of the reactor (self-actuation mechanical mode), the manual blow down of the reactor, and to provide a path for heated reactor water to return to the suppression pool so that it can be cooled by the RHR system in the LPCI mode. In the later two cases, the MSRVs would be opened manually from either the main control room or backup control panel.

The inspectors examined a sample of Unit 3 MSRVs for proper separation to ensure the MSRVs would be available in the event of an Appendix R fire. The areas reviewed (those in which the plant was at a greater risk as a result of the failure of the MSRVs due to the lack of high pressure injection) were as follows;

Unit 3 Reactor Building, Fire Area 3, Fire Zone 3-1

*3-PCV-1-4	*3-PCV-1-19
*3-PCV-1-30	*3-PCV-1-22
*3-PCV-1-41	*3-PCV-1-31
*3-PCV-1-42	*3-PCV-1-179

Unit 3 Reactor Building, Fire Area 3, Fire Zone 3-2

*3-PCV-1-4
*3-PCV-1-5
*3-PCV-1-30
*3-PCV-1-42

Unit 3 Reactor Building, Fire Area 3, Fire Zone 3-3

*3-PCV-1-18
*3-PCV-1-19
*3-PCV-1-31
*3-PCV-1-179

In each of the above cases, the inspectors reviewed the cable block diagrams for MSRVs that depicted the routing of the MSRV control cabling from the control room to the drywell penetration. The inspectors verified that for each of the above listed Unit 3 fire zones, the control cabling for the designated MSRVs were not routed through a fire area/zone intended to be served by that MSRV. However, in the case of Fire Zone 3-1, all eight designated MSRVs had their control cabling routed through fire zone 3-1 as both penetrations utilized by MSRV cabling are located in this zone. However, the penetrations were separated by a distance of greater than 20 feet (measured to be approximately 26 feet) and were located in an area with fire detection and suppression. In addition to the cable block diagram review, the inspectors performed a field verification to the maximum extent practicable to ensure proper cable separation.

Additionally, this verification was also performed for Fire Areas 12 (Unit 3, 480 V RMOV Board Room 3B), 13 (Unit 3, 480 V RMOV Board 3A), 16 (Control Building, including control room), and 19 (Unit 3, Battery and Battery Board Room). The inspectors also verified that the four MSRVs designated for remote operation in the event of control room abandonment were able to be operated from the backup control panel. In all cases, no discrepancies were identified.

The inspectors reviewed the Appendix R safe shutdown function of the Containment Atmospheric Dilution (CAD) system. More specifically, the safe shutdown function of the CAD system in the event of a fire was to provide a long-term supply of compressed gas to the MSRVs so that they could be held open to provide a portion of the alternate shutdown cooling flow path. This air is supplied to containment via the drywell control air system. The inspectors verified that the manual actions required to align the CAD system in the event of an Appendix R fire were achievable in that: [1] the manual valves required to be operated for the two trains of the CAD system are located in different fire zones (3-1 and 3-2), and [2] at least two MSRVs utilized for safe shutdown in fire zones 3-1 and 3-2 were supplied by the CAD train located in the other fire zone thereby allowing the long term operation of a least one MSRV.

(4) The inspectors reviewed design modification DCN W30895 for Unit 3 only. The modification was being implemented to address a high-low pressure interface issue with the RWCU system. This modification installs a new flow isolation valve (3-FCV-069-094) in the RWCU suction line just downstream of the non-regenerative heat exchanger. The valve was a failed-closed pneumatically operated (Worchester Controls, twin piston, Series 39 actuator) valve designed to close upon sensing a high temperature water condition via a fusible plug (3-FUPG-32-5105) mounted directly on the low temperature piping. With this new valve installed, isolation of the RWCU system could be assured by the automatic closure of valve 3-FCV-069-094 and/or manual closure of valves 3-FCV-069-001 or 3-FCV-069-002.

The inspectors concluded that the identified changes in the SSA provided an overall improvement to the Appendix R safe shutdown capability for Unit 3. Subject to further review by NRR, the safe shutdown capability was found acceptable.

3.0 REVIEW OF THE SEPARATION FOR SAFE SHUTDOWN SYSTEMS AND COMPONENTS

Based on the licensee's FPR, they have identified 25 fire areas for Browns Ferry Nuclear Plant. Each of the unit reactor buildings was considered one fire area, but were further divided into fire zones. Fire Area 2, the Unit 2 Reactor Building, was divided into six fire zones. The Unit 3 Reactor Building, Fire Area 3, was divided into four fire zones. These are listed in Section 6.0 of the SSA and illustrated on drawings 47W216-51 through 62.

Within the above plant areas, the licensee has performed an electrical separation study to ensure that at least one train of the safe shutdown equipment and cabling was available in the event of a fire in areas that might affect these components. The equipment listed for each fire area/zone defined the "minimum" set of major components which was required to provide for safe shutdown. The availability of these components was documented by calculation ED-Q3999-940040 "Appendix R Computerized Safe Shutdown Separation Analysis." This and other related cable separation calculations have been previously reviewed by the NRC. These reviews were documented in NRC Inspection Report (IR) 50-259, 260, 296/95-20. The safe shutdown equipment and cabling were identified and traced through each fire area from the component to the power source. The results were noted on drawings identified as cable block diagrams. Associated circuits whose fire-induced spurious operation could affect safe shutdown were identified by a system review to determine those components whose improper-operation could affect the safe shutdown capability.

On the basis of the licensee's SSA and calculation ED-Q3999-940040, the inspectors made an inspection of cabling and components associated with the High Pressure Coolant Injection (system 73), Reactor Core Isolation Cooling (system 71), Condensate Storage and Supply (system 02), Main Steam Relief Valves (system 01), Residual Heat Removal (system 74), Heating, Ventilating and Air Conditioning (system 31), Standby Diesel Generators (system 82), and the Residual Heat Removal Service Water (system 23). The inspection included a review and verification of the physical separation of the licensee identified safe shutdown system functions including primary and support equipment, cables, and associated circuit locations.

During the inspection about 33 safe shutdown components were selected from the list of components required for the SSDS utilized for shutdown. The routing of selected power and control circuits for these components were verified from cable block diagram drawings to assure that none of the required circuits passed through the fire area for which the component was in the successful safe shutdown path. In addition to the cable block diagram review, a select sample of circuits were field verified to the maximum extent practicable to ensure proper cable routing and separation. The sample of circuits verified are identified in Section 2.1 and Attachment 2 of this inspection report. In all cases, no discrepancies were identified.

The licensee's electrical separation study considered availability or non-availability of offsite power. Offsite power was the normal electrical power source to the boards which feed vital safe shutdown equipment. An inspection was made to determine in what plant fire areas could a fire event have the potential for a loss of offsite power or degraded electrical system. The inspection indicated that the potential for a loss of offsite power or degraded electrical system due to a fire event existed for a minimum 19 of the 25 identified plant fire areas. Should the ability to provide normal or emergency power to vital plant equipment be in serious doubt during a fire event, the licensee would align the Auxiliary Power System (APS) to the analyzed configuration for the particular fire area. This alignment requires transfer of the required diesel generators, associated 4 kV shutdown boards, required 480 V shutdown boards, the 480 V RMOV boards, the 250 V RMOV boards, and the diesel generator auxiliary boards to local control and/or their alternate power feeds within 20 minutes. This was previously reviewed and approved for Unit 2 restart in NRC Safety Evaluation Reports of April, 1988, (NUREG-1232, Vol.3) and January, 1989, (NUREG-1232, Vol.3, Supp 2).

The results of the inspectors' reviews indicated that Fire Area 16 (Control Building - Main Control rooms, Cable Spreading rooms, Mechanical Equipment rooms, Auxiliary Instrument rooms, Process Computer room, Communications room, and MG Set rooms) was the only plant area determined to be an Appendix R III.G.3 area requiring alternate shutdown. If a fire were to disable the main control room, the alternate shutdown panels, located in separate fire areas, would be available to achieve safe shutdown conditions. There were no safe shutdown repairs required for either hot shutdown or cold shutdown for Browns Ferry Units 2/3. No entry was made, either to perform a manual operator action or for ingress/egress purpose, into a fire area where the fire event originated prior to the fire being extinguished by the fire brigade.

Based on review of the SSA, calculation ED-Q3999-940040, and inspection of the circuit separation in the plant the inspectors determined that the separation of those systems necessary to achieve safe shutdown satisfied the separation requirements of Section III.G and III. L, of Appendix R to 10 CFR Part 50.

4.0 Associated Circuits

The separation and protection requirements of 10 CFR 50, Appendix R, apply not only to safe shutdown circuits but also to "associated circuits." An associated circuit are electric cable, not itself required for safe shutdown, that are routed through the fire area of the postulated fire and has one of the following characteristics:

* shares a <u>common power source</u> (bus) with the safe shutdown circuits, and the bus does not have selective coordination, or

* shares a <u>common enclosure</u>, which means it is routed together with safe shutdown circuits, and is unprotected by an over current protective device, or

* shares a common enclosure, which means it is routed together with safe shutdown circuits, and is not flame retardant (cables coated with flame retardant material could substitute for flame retardant cable), or

* connected to circuits of equipment whose spurious operation would adversely affect the safe shutdown capability.

In general, licensees perform analysis to show that there are no cables having one of the above characteristics, and, therefore, there are no associated circuits.

4.1 Common Power Source

The concern with an associated circuit by common power source was that a fire induced short-circuit could result in deenergizing a bus needed to power safe shutdown equipment.

To verify that the licensee adequately addressed the common power source concern, the inspectors selected a sample of safe shutdown equipment, and checked that the corresponding power source buses had selective coordination. The equipment selected was RHR pump 3B and associated LPCI injection valve 3-FCV-74-67. This equipment was shown on Mechanical Control Diagram, 3-47E610-74-1, Rev. 12, Residual Heat Removal. The safe shutdown analysis indicated that this equipment, or its counterpart in trains A and C, was required to shutdown the plant for a postulated fire event in each of the fire areas. RHR pump 3B was powered from 4 kV shutdown board 3EC. The power flow path to valve 3-FCV-74-67 was from the 4 kV shutdown board 3EC, 480 V shutdown board 3B and motor control center 3E.

The licensee's coordination study for these buses was contained in Calculation ED-Q3999-910224, Cable and Bus Protection/Breaker Coordination for 4 kV Switchgear and 480 V Load Centers and Breaker Coordination for 480 V Motor Control Center Board Incoming Breakers, Rev 6. The inspectors reviewed the coordination at buses 3EC, 3B and 3E in detail and agreed that the study used standard valid methodology. The inspectors verified that data in the study agreed with published vendor information. The inspectors verified setting and style numbers for the following key devices by inspection of the installed equipment:

* At 4 kV breaker 1326, the IAC51A101A relay was set on tap 7 and time dial 4.5, which matched the design study.

* At 4 kV breaker 1338, the IAC51A101A relay was set on tap 8 and time dial 4.5, which matched the design study.

* At the 5 kV breaker for the RHR pump 3B (largest load on bus), the 12IAC66K8A relay was set on tap 4 and time dial 2, which matched the design study.

* At 480 V breaker 3B-52N, the IAC53A101A relay was set on tap 6 and time dial 5, which matched the design study.

* At the 480 V breaker for MCC-3E, the IFC66K1A relay was set on tap 5 and time dial 8. The instantaneous setting could not be observed, but the design documents called for a setting of 35. The time over current relay settings were different than the design study. The licensee presented information taken from Design Change Notice W21819 which indicated the setting will be changed to tap 3.5 and time dial 10 and instantaneous equal to 35. The inspectors concluded that both the current setting and the future setting were acceptable from the Appendix R requirements viewpoint.

Based on review of the 910224 calculation and inspection of the relays and breakers in the plant the inspectors concluded that the Appendix R common power source concern had been adequately addressed by the licensee.

During an NRC inspection conducted on January 23 - 27, 1995, the licensee's analysis of coordination in control circuit fuses was reviewed (refer Inspection Report No. 50-296/95-02). Specifically, Calculation ED-Q0211-880138, Fuse Evaluation 4 kv Shutdown Boards A/B/C/D/3EA/3EB/3EC/3ED, dated July 20, 1993, was reviewed. The report agreed with the conclusion in that calculation that selective coordination existed in the control circuit fuses. This conclusion applies to Appendix R requirements as well.

4.2 Common Enclosure

The concern with cables falling into the common enclosure category was that fire induced damage occurring on associated circuits could be transmitted to safe shutdown circuits as the result of excessive thermal energy or direct flame propagation.

The licensee's analysis to address a portion of this concern was contained in Calculation ED-Q2999-880562, Appendix R Study of Cable Auto-Ignition Protection, Rev. 11, for safety-related circuits. The purpose of this calculation was to show that cables were protected for short-circuit by the over current protective devices. The inspectors reviewed this calculation and found that it used good engineering practice as defined in IEEE Std. 242-1986, IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems. Specifically it followed the guidance in Chapter 8, Cable Protection. The licensee's study showed that, in general, cables would not exceed the maximum short-circuit temperature for the type cable being analyzed. For certain low probability high impedance faults, calculated temperatures could exceed the maximum short-circuit temperature (e.g. 250 °C for cross-linked polyethylene), but would remain well below the auto-ignition temperature. This was acceptable because the objective was to show that a cable <u>adjacent</u> to the faulted cable would not fail during the Appendix R event.

The original installation at Browns Ferry did not use all flame retardant cable. At some point in time the licensee instituted a policy of purchasing only flame retardant cable to be used for new cable installations. Therefore, there was a mix of flame retardant and non-flame retardant cable installed in the cable trays. Fire retardant coating (Flamemastic) was applied to the non-flame retardant cables. The NRC staff has reviewed the licensee's actions to resolve Flamemastic related issues identified by Corrective Action Tracking Documents (CATDs). CATD 10900-BFN-04 (NRC #29) identified a concern that Flamemastic coating on cable had not been documented for ampacity affect for all applications. This issue had been previously inspected and documented in NRC IRs 50-259, 260, 296/94-35 and 50-259, 260, 296/95-25. These reports documented that ampacity concerns for cable affected by Flamemastic coatings had been satisfactorily resolved.

4.3 Spurious Operation

The licensee's safe shutdown analysis addressed the potential for fire induced spurious operation of equipment. The analysis identified the systems and equipment that had a potential vulnerability with regard to spurious operation. The analysis then showed that while spurious operations could occur on this equipment the spurious operation would either be of no real consequence to safe shutdown or the effect of the spurious operation could be offset by operator action within the operating procedures. The inspectors reviewed the system descriptions and concluded that all potential vulnerable points were identified and that the analysis of these points were valid.

4.4 Isolation/Transfer Switches

The design covering scenarios where the control room was abandoned and the plant would be shutdown using alternate shutdown panels incorporates isolation/transfer switches. The reason for the isolation/transfer switches was that control circuits for equipment that would be operated from both the backup shutdown panel (or other local stations) and the main control panel requires a means for isolating control room wiring from the balance of the control circuit. These switches were located outside the main control room and had two positions, normal and emergency. Circuits having these switches also had two sets of fuses. The inspectors reviewed the control circuits for the RCIC Suction Valves FCV-71-17 & 18 as examples of control circuits having an Appendix R isolation/transfer switch. These circuits were 250 DC control circuits. The inspectors concluded that the type of switch and wiring of the switch into the control circuit would achieve the desired transfer between control locations and isolation of control room wiring. In addition, the inspectors noted that one contact from the switch was used to generate an alarm in the main control room should the switch be inadvertently left in the emergency position.

5.0 Emergency Lighting

The inspectors selected a sample of rooms and plant pathways to evaluate the lighting levels of the Appendix R emergency lighting system. The inspectors observed the location of the battery pack emergency lighting units in the control room and the operator access pathway to the backup shutdown panel room (electrical board room 3A). The inspectors conducted a "lights out" test in the backup shutdown panel room and the adjacent 480 V switchgear room. During the test, the inspectors verified that the emergency lighting levels were adequate for reading a procedure and panel nameplates. Based on the location and number of lighting units observed, the inspectors concluded that the lighting levels were according to Appendix R requirements.

The inspectors noted that the licensee had plant arrangement drawings depicting the operator pathways associated with the various operator actions described in the safe shutdown analysis. These drawings also depicted the location of emergency lighting fixtures provided for access and egress routes to plant locations needed for additional manual operations of safe shutdown systems identified in the drafted Safe Shutdown Instructions (SSI). The NRC intends to independently verify the lighting levels of several additional plant areas after the SSIs are finalized. Therefore, Inspector Follow-up Item 50-296/95-37-01, Verification of Emergency Lighting Levels, is identified.

6.0 Procedures and Training

The inspectors performed a review of the operational readiness of plant operations to perform safe shutdown activities and the training conducted for operations personnel to support proposed operator manual actions during any postulated fire event.

6.1 Safe Shutdown Instructions

When the licensee makes the determination that an Appendix R fire condition exists, Units 2 and 3 will be shutdown using procedures entitled SSIs. These instructions take precedence over all other procedures and instructions, including the Abnormal Operating Procedures (AOP) and Emergency Operating Procedures (EOPs), once the Appendix R fire condition declaration is made. The SSIs utilize SSDS equipment and or systems to shutdown the units during and after a fire. Currently, the SSIs in place only contain instructions to shutdown Unit 2 . With the impending restart of Unit 3, the SSIs will encompass the shutdown of both Units 2 and 3. Each fire area/zone will have a corresponding Unit 2/3 SSI containing actions required to shutdown both units including specific actions (both local and remote) required to be performed by the Unit 2 and 3 operating staff. These procedures are currently in a draft status. The licensee has stated that the SSIs will receive initial Plant Operations Review Committee (PORC) review on August 17, 1995. Following PORC review, operations personnel will receive training on the SSIs during requalification training cycle 14, scheduled for August 21 - September 29, 1995. Upon completion of requalification training, the licensee will incorporate comments received during the requalification cycle with the procedures ultimately being issued by November 29, 1995. Prior to Unit 3 restart, NRC staff inspectors will observe a simulated shutdown of Units 2 and 3 utilizing the 2/3 SSIs for a specific fire zone/area to be determined at a later date. The inspectors will verify that the required manual actions can be performed including evaluation of routes taken by operators, secondary fire conditions (i.e. smoke), adequacy of lighting, and command and control including communications. During the previously mentioned requalification training on the SSIs, the inspectors will review the implementation and training for the remote shutdown of both units. The implementation of the SSIs, including the observation of a simulated shutdown will be identified as Inspector Follow-up Item 50-296/95-37-02, Performance of Simulated Shutdown for an Appendix R Event.

6.2 Entry Conditions for an Appendix R Fire

During the inspection, the entry conditions for an Appendix R fire at Browns Ferry were examined. The entry conditions as stated in the licensee's combined Unit 2/3 SSA were as follows:

- A confirmed fire exists that cannot immediately be extinguished.
- Units 2 and/or 3 reactor pressures are greater than atmospheric.
- Unit 1 is in Cold Shutdown.
- Inability to maintain reactor water level with high pressure systems.
- Offsite power is lost or unavailable or degraded plant electrical system.

The inspectors questioned a number of aspects related to these entry conditions. Specifically, the proceduralization and transition between a "fire" and an "Appendix R fire" were discussed with the licensee. Although no specific operations procedure (with respect to the operation of plant equipment) exists for a "fire," the inspectors were satisfied that when the operations supervisor determined that an "Appendix R fire" existed, all other procedures would be exited and the SSIs entered and carried out. A second question related to the inability to maintain the reactor water level with high pressure systems was discussed. The inspectors questioned whether this criteria could be satisfied with high pressure injection available. The licensee provided clarification of this matter with a table detailing which high pressure systems would be available for each fire area/zone and at which reactor water level the entry conditions for this matter would be satisfied. Finally, a discussion on the definition of degraded electrical system was held. This matter was satisfactorily explained in that this entry condition was solely based on the judgement of the shift operations supervisor. These entry conditions will be examined further during the remaining segments of this inspection effort.

7.0 Exit Meeting

The inspection scope and results were summarized on July 21, 1995, with those individuals indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. There were no dissenting comments received from the licensee. Proprietary information is not contained in this report.

Two new Inspector Follow-up Items were identified:

(OPEN) Inspector Follow-up Item 50-296/95-37-01, Verification of Emergency Lighting Levels (Section 5.0).

(OPEN) Inspector Follow-up Item 50-296/95-37-02, Performance of Simulated Shutdown for an Appendix R Event (Section 6.0).

8.0 Acronyms and Abbreviations

ACU	Air Conditioning Unit
AOP	Abnormal Operating Procedure
APS	Auxiliary Power System
BAT	Battery
BBDR	Battery Board Room
BFN	Browns Ferry Nuclear (station)
BD	Board
CATD	Corrective Action Tracking Documents
CB	Control Building
CFR	Code of Federal Regulations
DCN	Design Change Notice
DGB	Diesel Generator Building
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FA	Fire Area
FCV	Flow Control Valve
FPR	Fire Protection Report
FZ	Fire Zone
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, and Air Conditioning
IFI	Inspection Followup Item
IP	Inspection Procedure
IR	Inspection Report
kV	kilo-Volts
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MCR	Main Control Room
MSRV	Main Steam Relief Valve
MTR	Motor
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PCV	Pressure Control Valve
PORC	Plant Operations Review Committee
RB	Reactor Building

RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMOV	Reactor Motor Operated Valve
RWCU	Reactor Water Cleanup
SDBD	Shutdown Board
SER	Safety Evaluation Report
SSDS	Safe Shutdown Systems
SSA	Safe Shutdown Analysis
SSI	Safe Shutdown Instructions
TB	Turbine Building
TVA	Tennessee Valley Authority
V	Volts

ATTACHMENT 1

Based on the review of the SSA for the combined operation of Browns Ferry Units 2 and 3, the inspectors assembled the following Appendix R post-fire safe shutdown matrix that identifies the availability of high pressure make-up coolant systems **(Denoted by: Y = YES, N = NO)** and other major equipment **(Denoted by the Equipment Identification Number)** necessary to achieve and maintain safe shutdown conditions.

BROWNS FERRY APPENDIX R SAFE SHUTDOWN MATRIX									
FIRE AREA AVAILABLITY	**U-2 HPCI**	**U-3 HPCI**	**U-3 RCIC**	**U-2 RHR LPCI**	**U-3 RHR LPCI**	**U-2 RHR SW**	**U-3 RHR SW**	**U-2 EECW**	**U-3 EECW**
1/UNIT 1 RB	Y	Y	Y	2D	3B	D2	B1	A3&C3	A3&C3
ZONE 2-1	N	Y	Y	2D	3B	D2	B1	A3&C3	A3&C3
ZONE 2-2	N	Y	Y	2C	3A	C1	A2	A3&C3	A3&C3
ZONE 2-3	N	Y	Y	2D	3C	D2	C1	A3&C3	A3&C3
ZONE 2-4	N	Y	Y	2D	3C	D2	C1	A3&c3	A3&C3
ZONE 2-5	N	Y	Y	2C	3B	C1	B2	A3&C3	A3&C3
ZONE 2-6	N	Y	Y	2D	3A	D2	A2	A3&C3	A3&C3
ZONE 3-1	Y	N	N	2D	3B	D2	B2	B3&D3	B3&D3
ZONE 3-2	Y	N	N	2D	3C	D2	C2	B3&D3	B3&D3
ZONE 3-3	Y	N	N	2D	3B	D2	B2	B3&D3	B3&D3
ZONE 3-4	Y	Y	N	2D	3A	D2	A2	B3&D3	B3&D3
4/U-1 SDBR-B	Y	Y	Y	2D	3A	D2	A2	A3&C3	A3&C3
5/U-1 SDBR-A	Y	Y	Y	2D	3B	D2	B1	A3&C3	A3&C3
6/U-1 SDBR-1A	Y	Y	Y	2D	3B	D2	B2	A3&C3	A3&C3
7/U-1 SDBR-1B	Y	Y	Y	2D	3C	D2	C1	A3&C3	A3&C3
8/U-2 SDBR-D	Y	Y	Y	2C	3A	C1	A2	A3&C3	A3&C3
9/U-2 SDBR-C	N	Y	Y	2D	3B	D2	B1	A3&C3	A3&C3
10/U-2 SDBR-2A	Y	Y	Y	2D	3B	D2	B1	A3&C3	A3&C3

11/U-2 SDBR-2B	Y	Y	Y	2C	3B	C2	B1	A3&C3	A3&C3
12/U-3 RMOV-3B	Y	N	N	2D	3A	D2	A2	B3&D3	B3&D3
13/U-3 RMOV-3A	Y	N	N	2D	3B	D2	B2	B3&D3	B3&D3
14/U-3 SDBR-3A	Y	Y	Y	2D	3B	D2	B1	B3&D3	B3&D3
15/U-3 SDBR-3B	Y	Y	Y	2D	3A	D2	A2	B3&D3	B3&D3
16/CB&MCR	N	N	Y	2C	3A	C2	A1	B3&D3	B3&D3
17/U-1 BAT/BBDR	Y	Y	Y	2D	3A	D2	A2	B3&D3	B3&D3
18/U-2 BAT/BBDR	N	Y	Y	2C	3B	C1	B1	B3&D3	B3&D3
19/U-3 BAT/BBDR	Y	N	Y	2D	3A	D2	A2	B3&D3	B3&D3
20/U-1&2 DGB	Y	Y	Y	2D	3A	D2	A2	A3&C3	A3&C3
21/U-3 DGB	Y	Y	Y	2D	3A	D2	A2	B3&D3	B3&D3
22/U-3 SDBR 3EA&3EB	Y	Y	Y	2D	3B	D2	B2	B3&D3	B3&D3
23/U-3 SDBR 3EC&3ED	Y	Y	Y	2D	3A	D2	A2	B3&D3	B3&D3
24/U-3 BUS TIE RM	Y	Y	Y	2D	3B	D2	B1	B3&D3	B3&D3
25(I)/TB, CABLE TUN. INTAKE PUMP STA.(DIV-I)	Y	Y	Y	2C	3A	C1	A1	A3&C3	A3&C3
25(II)/TB & EXTERIOR PLANT (DIV-II)	Y	Y	Y	2D	3B	D2	B2	B3&D3	B3&D3

ATTACHMENT 2

SEPARATION FOR SAFE SHUTDOWN SYSTEMS AND COMPONENTS

The routing of selected power and control circuits for the following post fire safe shutdown components were either verified from cable block diagram drawings and/or field verified for proper routing to assure none of the required circuits passed through the fire area for which the component was in the successful safe shutdown path.

Safe Shutdown System-Function/Component	Cable/Raceway/Conduit No./Function	Cable Route Location By Fire Area/Zone
Condensate Storage Tank 2 - Discharge Isolation Valve/ 3-FCV-2-166	PL 1825/Power Cable PL 1828/Control Cable	FA 25/Plant Yard-480 V. Water and Oil Storage Board,Compartments 1A and 4B. FA 16/ Control Room,Panel 9-20.
HPCI- Condensate Storage Tank - Suction Valve/ 3-FCV-[illegible]-40	3ES 2800-II/JB 3153/Power Cable 3ES 2812-II/JB 3153/Control Cable	FZ 3-2/Reactor Building. FZ 3-3/Reactor Building. FA 13/RMOV Board Room 3A, Compartment 1D2.
RCIC- Condensate Storage Tank - Suction Valve/ 3-FCV-71-19	3ES 1307-I/JAI-I/KAI-I/LAI-I/JAM-I/Control Cable	FZ 3-4/Reactor Building, Panel 25-31. FZ 3-3/Reactor Building. FA 16/ Control Room/Panel 9-3A.
HPCI- Suppression Pool Supply - Suction Valve/ 3-FCV-73-26	3ES 2888-II/JB 3166/LOA-II/KOA-II/3A-3ES3232-II/Power Cable	FZ 3-2/Reactor Building. FZ 3-3/Reactor Building. FA 13/RMOV Board Room 3A, Compartment 4D.

ATTACHMENT 2

RCIC- Suppression Pool Supply - Outboard Suction Valve/ 3-FCV-71-17	3ES1283-I/3ES1291-I/JB 3077/3ES1312-I/JOA-I/3ES1531-I/Power Cable. 3ES1286-I/JB 3077/3ES1291-I/Control Cable. 3ES1287-I/3ES1291-I/JB 3077/3ES1312-I/JOA-I/JMS-I/SZ-I/Control Cable	FZ 3-1/Reactor Building. FZ 3-1/RMOV Board Room 3C, Compartment 8B, 3-HS-71-17C. FZ 3-1/Local Control Station, 3-HS-71-17B. FA 16/Control Room/Panel 9-3A, 3-HS-71-17A.
RCIC- Suppression Pool Supply - Inboard Suction Valve/ 3-FCV-71-18	3ES1296-I/3ES1530-I/JGA-I/OP-I/JMS-I/Control Cable	FZ 3-1/RMOV Board Room 3C, Compartment 7D, 3-HS-71-18C. FZ 3-1/Reactor Building. FZ 3-3/Reactor Building. FA 16/ Control Room/Panel 9-3A, 3-HS-71-18A.
HPCI- Steam Admission Valve to Turbine/ 3-FCV-73-16	3ES2788-II/JB3587/ 3ES3211-II/EGH-II/GJ-II/GS-II/3ES3230-II/Power Cable	FZ 3-2/Reactor Building. FZ 3-3/Reactor Building. FA 13/RMOV Board Room 3A, Compartment 3D.
RCIC- Steam Admission Valve to Turbine/ 3-FCV-71-08	3ES1376-I/JAI-I/KAI-I/LAI-I/NAH-I/RO-I/JMS-I/JAM-I/SZ-I/Control Cable	FZ 3-4/Unit 3 Reactor Building/ Panel 25-31. FZ 3-3/Reactor Building. FA 16/ Control Room/Panel 9-3A, 3-HS-71-8A.

ATTACHMENT 2

RHR- Pump 3A/ 3-MTR-74-5	3ES1560-I/JB3593/CE-I/Power Cable	FZ 3-1/ Unit 3 Reactor Building. FZ 3-2/ Unit 3 Reactor Building. FA 22/Diesel Generator Building, Unit 3 4KV SDBR 3EA, Compartment 12.
RHR- Pump 3C/ 3-MTR-74-16	3ES1570-I/JB3593/CE-I/Power Cable	FZ 3-1/Reactor Building. FZ 3-2/ Reactor Building. FA 22/ Diesel Generator Building, Unit 3 4KV SDBR 3EB, Compartment 4.
HVAC- Shutdown Board Room Air Conditioning Unit 3B/ 3-ACU-031-7206	3PL6071-II/3PL6092-II/Power Cable	FA 12/ Unit 3 RMOV Board Room 3B, ACU-3B Main Control Panel 15C (3-LPNL-25-541).
Emergency Diesel Generators- Fuel Oil Supply Transfer Pumps 3A/ 3-MTR-18-70A & -72A	3ES1615-IA/3ES1609-IA/JB4203/3ES1608-IA/Power Cable to Motors	FA 21/ Unit 3 Diesel Generator Building
Emergency Diesel Generators- Fuel Oil Supply Transfer Pumps 3B/ 3-MTR-18-70B & -72B	3ES1655-IB/3ES1649-IB/JB4206/3ES1648-IB/Power Cable to Motors	FA 21/ Unit 3 Diesel Generator Building
RHRSW- Pump A1/ 3-MTR-23-01	ES75-I/JB4859/JB4915/AT-I/AZ-I/Power Cable to Motor	FA 25/ Intake Pump Station, Yard, and Turbine Building FZ 1-1/ Unit 1 Reactor Building. FA 5/ Unit 1 4KV SDBR A, Compartment 10.

ATTACHMENT 2

RHRSW- Pump B2/ 3-MTR-23-19	ES2588-II/CB-II/AO-II/BU-II/BK-II/Power Cable to Motor	FA 25/ Intake Pump Station, Yard, and Turbine Building FZ 2-2/ Unit 2 Reactor Building. FZ 2-4/ Unit 2 Reactor Building. FA 9/ Unit 2 4KV SDBR C, Compartment 16.