UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

August 8, 2000

Tennessee Valley Authority
ATTN: Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INSPECTION REPORT 50-259/00-08, 50-260/00-08, AND 50-296/00-08

Dear Mr. Scalice:

This refers to the inspection conducted on June 26-30, 2000, at your Browns Ferry Nuclear facility. This was a fire protection triennial baseline inspection which was performed using Procedure 71111.05 under the revised reactor oversight process. The enclosed report presents the results of that inspection.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license as they related to implementation of your NRC-approved Fire Protection Program. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Based on the results of this inspection, no findings were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publically Available Records* (PARS) *component* of the NRC's document system (ADAMS). ADAMS *is accessible from* the NRC web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Kerry D. Landis, Chief
Engineering Branch
Division of Reactor Safety

Docket Nos. 50-259, 50-260, 50-296
License Nos. DPR-33, DPR-52, DPR-68

Enclosure: NRC Inspection Report
Nos. 50-259/00-08, 50-260/00-08, 50-296/00-08

Attachment: NRC's Revised Reactor

Oversight Process Summary

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-259, 50-260, 50-296
License Nos:	DPR-33, DPR-52, DPR-68
Report Nos:	50-259/2000-08, 50-260/2000-08, 50-296/2000-08
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, & 3
Location:	Corner of Shaw and Browns Ferry Roads Athens, AL 35611
Dates:	June 26 - 30, 2000
Inspectors:	E. Brown, Resident Inspector, Brunswick, Region II P. Fillion, Reactor Inspector, Region II A. Fresco, Contractor, Brookhaven National Laboratories R. Schin, Senior Reactor Inspector, Region II G. Wiseman, Senior Reactor Inspector (Lead Inspector), Region II
Approved By:	Kerry D. Landis, Chief Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Browns Ferry Nuclear Plant, Units 1, 2, and 3
NRC Inspection Report 50-259/00-08, 50-260/00-08, 50-296/00-08

This report covers a one-week period of inspection onsite by a team to perform the triennial fire protection baseline inspection using procedure 71111.05, Fire Protection. There were no findings identified.

REPORT DETAILS

Summary of Plant Status: Unit 3 was at 100 percent power and Unit 1 was defueled during this inspection. Unit 2 started this inspection week at 100 percent power and ended the week shut down. Unit 2 was shut down on June 29 to address drywell unidentified leakage.

REACTOR SAFETY

CORNERSTONES: INITIATING EVENTS and MITIGATING SYSTEMS

1R05 FIRE PROTECTION

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

a. Inspection Scope

The team selected five risk significant fire areas/zones identified in the licensee's individual plant examination for external events (IPEEE) to verify that the post-fire safe shutdown (SSD) capability and the fire protection features provided for ensuring that at least one post-fire SSD success path was maintained free of fire damage. For each of these fire areas, the team focused its inspection on the fire protection features, and on the systems and equipment necessary for the licensee to achieve and maintain SSD conditions. The fire areas chosen for review during this inspection were:

(1) Unit 1, 4KV Shutdown Board A Room [Fire Area 5]

NOTE: Although Unit 1 is permanently defueled and not in operation, an Appendix R fire in this area would involve alternative shutdown of both Units 2 and 3, coordinated from the control rooms.

(2) Unit 3, 480V Reactor MOV Board Room 3B [Fire Area 12]

NOTE: An Appendix R fire in this area would involve alternative shutdown of both Units 2 and 3, coordinated from the control rooms, and a rapid depressurization of Unit 3.

(3) Control Building Elevation 593 Through 617 (including main control rooms, Unit 2 auxiliary instrument room, and cable spreading rooms for Units 2 and 3) [Fire Area 16]

NOTE: An Appendix R fire in this area would involve alternative shutdown of both Units 2 and 3, evacuation of both control rooms, locally powering all safety-related equipment from the emergency diesel generators (EDGs), and a rapid depressurization of Unit 2.

(4) Unit 2, Battery and Battery Board Room [Fire Area 18]

NOTE: An Appendix R fire in this area would involve alternative shutdown of both Units 2 and 3, coordinated from the control rooms, and a rapid depressurization of Unit 2.

(5) Unit 3, 4160V Shutdown Board 3EA & 3EB Room [Fire Area 22]

NOTE: An Appendix R fire in this area would involve alternative shutdown of both Units 2 and 3, coordinated from the control rooms.

For each of these fire areas, the team reviewed the licensee's Fire Hazard Analysis (FHA), 10 CFR 50, Appendix R Report, Safe Shutdown Analysis (SSA), Appendix R Safe Shutdown Program, and Updated Final Safety Evaluation Report (UFSAR) to determine the identified components and systems necessary to achieve and maintain SSD conditions. The objective of this evaluation was to verify that the licensee's shutdown methodology had properly identified the components and systems necessary to achieve and maintain SSD and assure the post-fire SSD analytical approach was consistent with and satisfied the 10 CFR 50, Appendix R reactor performance criteria for a boiling water reactor (BWR).

b. Issues and Findings

There were no findings identified.

.2 Fire Protection of Safe Shutdown Capability

.21 Fire Detection Systems

a. Inspection Scope

The team walked down the accessible portions of the fire detection and alarm systems in the auxiliary instrument room (fire area 16), 480V reactor MOV board room 3B (fire area 12), battery and battery board rooms (fire area 18), and 4160V shutdown board 3EA & 3EB room (fire area 22) to evaluate the engineering design and operation of the installed configurations. The team also reviewed engineering evaluations for the detection design, spacing criteria, and detector locations for the installed detection systems in the selected plant areas to verify effectiveness of the systems and compliance with the National Fire Protection Association (NFPA) code.

b. Issues and Findings

There were no findings identified.

.22 Fire Protection Water Supply System

a. Inspection Scope

The team reviewed flow and wiring diagrams, and cable routing information associated with the fire pumps and fire protection/raw service water supply system. These systems were necessary for manual fire fighting activities and/or water-based fire suppression systems which protected redundant trains of systems for hot shutdown. The review was to determine whether the common fire protection water delivery and supply components could be damaged or inhibited by fire-induced failures of electrical power supplies or control circuits.

b. Issues and Findings

There were no findings identified.

.23 Fixed Fire Suppression Systems

a. Inspection Scope

The team reviewed the adequacy of the design and installation of the carbon dioxide (CO_2) fire suppression systems for fire area 18 and the sprinkler systems located in fire areas 12, 16, and 22. Team members performed a walk down of the selected areas to assure proper placement and spacing of sprinkler heads and the lack of obstructions. The team reviewed CO_2 controls to assure accessibility and functionality of the system and associated ventilation system fire dampers. Vendor design calculations were verified to ensure that the required quantity of CO_2 for each area was available. Quality Assurance/Quality Control audits, fire protection program self-assessments, and the adequacy of licensee problem identification and resolution for identified findings were reviewed. Selected 10 CFR 50, Appendix R exemptions and engineering evaluations for NFPA code deviations were reviewed and compared against the physical configuration of the selected fire areas. Additionally, the team reviewed flow diagrams, and engineering evaluations associated with floor drain and heating ventilating and air conditioning (HVAC) systems to verify that systems and operator actions required for post-fire safe shutdown would not be inhibited by leakage or flooding from fire suppression activities or rupture of fire suppression systems.

b. Issues and Findings

There were no findings identified.

.24 Fire Barrier Enclosures

a. Inspection Scope

The team reviewed the five selected fire areas to evaluate the adequacy of the fire resistance of fire area barrier enclosure walls, ceilings, floors, structural beam support protection, fire barrier penetration seals, fire doors, and fire dampers by observing the material condition and configuration of the installed fire barrier features, as well as, construction details and supporting fire endurance tests for the installed fire barrier features. The team also reviewed the FHA to verify the fire loading used by the licensee to determine the fire resistive rating of the fire barrier enclosures. In addition, the team reviewed the licensing documentation, 10 CFR 50, Appendix R exemptions, generic letter (GL) 86-10 engineering evaluations of fire barrier features, engineering calculations, and NFPA code deviations to verify that the fire barrier installations met licensing commitments.

b. Issues and Findings

There were no findings identified.

.25 Fire Brigade Equipment

a. Inspection Scope

The team performed a walk down of the fire brigade house and response vehicle to assess the condition of fire fighting equipment. Fire brigade personal protective equipment was reviewed to evaluate equipment accessibility and functionality. The adequacy of the fire brigade self-contained breathing apparatus (SCBAs) was reviewed as well as the availability of supplemental breathing air tanks.

b. Issues and Findings

There were no findings identified.

.26 Fire Brigade Drill Program

a. Inspection Scope

The fire brigade is a dedicated group which is independent of the control room staff. The team reviewed the fire brigade drill program and observed fire brigade response associated with an announced fire brigade drill in fire area 22. The team observed the drill to verify that: 1) the fire brigade properly donned their protective clothing and turnout gear; 2) SCBAs were properly worn and used; 3) fire hoses were capable of reaching the location and properly laid out; 4) the fire brigade made a controlled fire area entry; 5) the fire brigade leaders directions were clear; 6) radio communications were effective; and 7) the brigade’s response and drill performance met the established drill objectives. The team also verified that the fire brigade performed a search for smoke and/or fire propagation, as well as search activities for fire victims. Additionally, the team observed the conduct of smoke removal activities simulated by the fire brigade and control room operators. Previous critiques of other operating shifts’ drill performances and the fire brigade training/ drill records were reviewed to determine if the fire brigade personnel qualifications and drill participation met the requirements of the licensee’s approved fire protection program.

b. Issues and Findings

There were no findings identified.

.27 Compensatory Measures

a. Inspection Scope

The team reviewed the administrative controls for out-of-service, degraded, and/or inoperable, fire protection systems and post-fire SSD systems and components. The review was performed to verify that the risk associated with removing fire protection and/or post-fire systems or components was properly assessed and adequate compensatory measures were implemented in accordance with the licensee's technical specifications (TS) and approved fire protection program.

b. Issues and Findings

There were no findings identified.

.3 Post-fire Safe Shutdown Circuit Analysis

a. Inspection Scope

On a sampling basis, the team reviewed drawings, schematics, wiring diagrams, and cable routing information associated with systems and components required for post-fire safe shutdown identified in the licensee's SSA to verify that power and control cables associated with post-fire safe shutdown equipment and associated non-safety related (NSR) circuits and cables in the selected fire areas had been identified by the licensee and had been analyzed to show that they would be free of fire damage as required by 10 CFR 50.48 and Section III.G of Appendix R to 10 CFR Part 50. Selected for review were circuits for equipment associated with the fire protection system pumps and valves, main steam isolation valves (MSIVs), low pressure coolant injection (LPCI) system, main steam relief valves (MSRV), residual heat removal (RHR) system, residual heat removal service water (RHRSW) system, and the EDGs.

The team evaluated the licensee's circuit analysis, system piping and instrumentation drawings, schematics, and wiring diagrams for components in each of the safe shutdown systems necessary for system success whose inadvertent operation or mal-operation due to a fire could initiate a transient, result in a loss of coolant accident, or adversely affect the post-fire safe shutdown capability by flow diversion or system isolation. This inspection also included a review of the licensee's fuse/breaker coordination analysis for the electrical distribution system.

b. Issues and Findings

There was one issue identified during the inspection of post-fire safe shutdown circuit analysis involving the spurious closure of a RHR minimum flow control valve. The issue is identified as an unresolved item pending further NRC review to determine the risk significance of this condition.

The licensee's SSA was based on protecting a minimum set of SSD systems and equipment for a fire in any given fire area. Section 4.3.a. of the SSA discusses the consequences of loss of power and spurious closure of the minimum flow valve for the RHR pumps during the LPCI mode of operation. One minimum flow bypass line is provided for each pair of the RHR pumps to protect the pump from overheating at low flow rates. The bypass line routes water from the pump discharge to the suppression pool. An orifice at each pump limits flow to 500 gpm. One motor operated valve controls the bypass line for the pump pair. This valve is normally open, but closes on sufficient flow. The potential to lose electric power and have the minimum flow control valve remain open is not important because the diverted flow would be small compared to the pump capacity and system requirements. The SSA makes the following statements concerning the potential for spurious closing of the minimum flow valve:

> Therefore, the RHR pump start is limited to those two signals (manual initiation or low water level) both of which will occur at approximately 20 minutes into an event. At this time, the operator would have gained sufficient control of the system that the dead-headed pump would only operate in this mode for 1 or 2 minutes. This limited time of dead-headed operation is not likely to cause pump failure (reference 30). This assessment is based on observations from several other plants which experienced running the RHR pump without minimum flow line.

Reference 30 mentioned above is a letter from the vendor dated September 5, 1985, which essentially makes the same statements as the licensee's SSA. The team reviewed the vendor letter.

Based on the reasons stated above, the licensee excluded the RHR minimum flow valves from the required minimum SSD equipment and did not perform a circuit analysis for the RHR pump minimum flow valve.

The team ascertained through discussions with licensee operations specialist that the RHR pump dead-head time for a representative fire scenario would be about eight minutes for the case of no damage in the main control room. In the event of a need to implement a rapid depressurization where loss of high pressure makeup capability had been damaged due to a fire or for any other reason, the plant emergency operating instructions directed the operator to start the RHR pump prior to the actual depressurization to ensure that the pump was ready to make up the lost inventory. Since starting of an RHR pump under such conditions would occur with the LPCI discharge isolation valve closed, the RHR pump minimum flow line path to pump suction must be available to ensure minimum flow through the RHR pump to prevent overheating and damaging the pump. If the normally open minimum flow valve were to spuriously close during this time, the RHR pump would be operating in a dead headed condition without minimum flow requirements being met.

It was estimated that three minutes could elapse between starting the RHR pump and opening the specified number of safety relief valves for depressurization. Based on simulator exercises conducted during the inspection, the team determined that about five minutes would elapse between opening the safety relief valves and the point at which reactor pressure was reduced to 330 pounds per square inch gauge (psig) and injection flow commenced. For the case of the control room fire, shutdown would be controlled from the remote shutdown board; and, assuming the minimum flow valve could be affected by this fire, the RHR pump dead-head time could be 15 minutes due to the time necessary for communications in the period after starting the pump and opening the safety relief valves. Therefore, the dead-head time could be greater than that stated in the SSA evaluation. Even during fire incidents where a high pressure makeup system remains available, after two (2) hours, the operator must first start an RHR pump and then perform a rapid depressurization for the plant to make the transition from hot shutdown to cold shutdown.

In view of the importance of the RHR pump minimum flow valves remaining in the open position, the team was concerned that the acceptability of dead-heading the pump was based on events at other plants, but no detailed comparison of pump characteristics and system conditions was included in the discussion. There was no statement of what fluid temperatures within the pump could be expected at the end of the estimated dead-head time as compared to pump component temperature limits nor whether the flash point of the fluid could be reached.

For Unit 2, the RHR minimum flow valve was powered and controlled from a motor control center in the reactor building. The SSA subdivided the reactor building into six fire zones. For each of these zones, the analysis stated that a rapid depressurization would be necessary. The analysis credited only one RHR pump, either 2C or 2D depending on the zone. Since the licensee excluded the minimum flow valve from circuit analysis, it was not known through which zones the relevant cables were routed. However, there is a high probability that the cables are vulnerable for a fire in at least one or more of the zones. The situation at Unit 3 would be similar.

On July 24, 2000, the licensee reported, pursuant to 10 CFR 50.72, that the issue of spurious closure of the RHR minimum flow valve constituted a condition not covered by the plant operations and emergency procedures in noncompliance with Appendix R, Section III.L.2. Fire watches were posted.

This issue remains unresolved pending further licensee evaluation of the issue, NRC review of the licensee’s evaluation, and NRC determination of the risk significance. It is identified as Unresolved Item 50-260, 296/00-08-01, Determination of the Risk Significance of Dead-Heading the RHR Pump.

.4 Alternative Shutdown Capability

a. Inspection Scope

The team performed a review of the licensee's abnormal procedures for fire response, alternative safe shutdown procedures/instructions (SSI), emergency plan procedures, and the Appendix R manual action requirements analyses for the five selected fire areas. The team also walked down selected portions of the SSIs. The reviews focused on ensuring that all required functions for post-fire safe shutdown and the corresponding equipment necessary to perform those functions were included in the procedures. The walk downs focused on ensuring that the procedures could reasonably be performed within the required times given the minimum staffing level of operators specified in the SSA, with or without offsite power available. The team reviewed the licensee's smoke control procedures, ventilation systems, and SCBA availability to verify that smoke would not prevent operators from performing the safe shutdown instructions. In addition, the team reviewed the EDG control circuitry to verify that transfer from the control rooms to the alternative shutdown locations was not affected by fire-induced circuit faults. Also, the team reviewed the 10 CFR 50.65 reliability data for the RHR pumps, RHRSW pumps, and the emergency equipment cooling water (EECW) headers to verify that there had not been an excessive number of maintenance preventable failures. The objective of these reviews was to assure that the post-fire safe shutdown analytical approach, safe shutdown equipment, and procedures were consistent and satisfied the Appendix R reactor performance criteria for safe shutdown.

b. Issues and Findings

There were no findings identified.

.5 Operational Implementation of Alternative Shutdown Capability

a. Inspection Scope

The team reviewed the operational implementation of the alternative shutdown capability for the selected fire areas to verify that: (1) the training program for licensed personnel included alternative or dedicated SSD capability; (2) seven operators were available from normal onsite staff to achieve and maintain the plant in cold shutdown following a fire using the alternative shutdown system, exclusive of the fire brigade; (3) the licensee had incorporated the operability of alternative shutdown transfer and control functions into the plant TSs; and (4) the licensee periodically performed operability testing of the alternative shutdown instrumentation and transfer and control functions. The team also reviewed the licensee's design modification and maintenance programs to verify that Appendix R considerations were incorporated.

b. Issues and Findings

There were no findings identified.

.6 Communications for Performance of Alternative Shutdown Capability

a. Inspection Scope

During walk downs of the SSIs, the team inspected remote shutdown equipment and verified that adequate radio communications equipment was available for the personnel performing alternative safe shutdown. The team also observed selected sound powered phone jack locations to verify that the jacks were in good condition, free of foreign material, and installed at the proper locations to support required shutdown actions identified in the procedures.

b. Issues and Findings

There were no findings identified.

.7 Emergency Lighting for Performance of Alternative Shutdown Capability

a. Inspection Scope

The team reviewed the design and operation of the 8-hour battery powered emergency lighting systems. The team performed a walk down of the remote shutdown equipment identified in the SSIs for selected fire areas and verified that emergency lighting unit lamps were operational and the lighting heads aimed to provide adequate illumination to perform the required shutdown actions required by the procedures. The team also observed operation of emergency lighting at the 4160V Shutdown Board C during a blackout of normal lighting in the area.

b. Observations and Findings

There were no findings identified.

.8 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed the licensee’s Appendix R emergency maintenance ventilation procedure and inspected the portable equipment and ventilation ducts stored at a special Appendix R storage area and other locations on site used for cooling an electrical equipment room required to reach cold shutdown.

b. Observations and Findings

There were no findings identified.

OTHER ACTIVITIES

4OA5 Management Meetings

.1 Briefings and Exit Meeting Summaries

The lead inspector presented the inspection results to Mr. J. Herron, Site Vice President, and other members of the licensee's staff at the conclusion of the onsite inspection on June 30, 2000. The licensee acknowledged the issue involving the spurious closure of a RHR minimum flow control valve.

The team asked the licensee whether any of the material examined during the inspection should be considered proprietary. Proprietary information was reviewed by the team but is not included in this inspection report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Abbas, Fire Protection Engineer
T. Abney, Licensing Manager
A. Bhatnagar, Site Support Manager
P. Chadwell, Operations Support Supervisor
T. Cornellus, Emergency Preparedness Manager
M. Heatherly, Corporate Fire Protection Engineer
J. Herron, Site Vice President
R. Jones, Plant Manager
G. Little, Operations Manager
L. Long, Fire Protection
D. Matherly, Programs Manager
R. Norton, Site Quality Manager
R. Sampson, Electrical Engineer
D. Sanchez, Training Manager
R. Wiggall, Site Engineering Manager
R. V. White, Fire Protection
R. Wright, Design Engineering Manager

Other licensee employees contacted included engineers, operations personnel, fire services personnel, and administrative personnel.

NRC

K. Landis, Chief, Engineering Branch, Division of Reactor Safety
W. Smith, Senior Resident Inspector
J. Starefos, Resident Inspector

ITEMS OPENED, CLOSED, OR DISCUSSED

Opened

50-260, 296/00-08-01	URI	Determination of the Risk Significance of Dead-Heading the RHR Pump. (Section 1R05.3)

APPENDIX

LIST OF DOCUMENTS REVIEWED

PROCEDURES

Abnormal Operating Instruction 0-AOI-26-1, Fire Response, Revision 2

Abnormal Operating Instruction 2-AOI-100-2, Control Room Abandonment - Revision 45

EOI Program Manual - EOIPM Section 0-II-C, EOI-1, RPV Control Guideline - Revision 1

Emergency Plan Implementing Procedure EPIP-6, Activation and Operation of the Technical Support Center (TSC), Revision 20A

Emergency Plan Implementing Procedure EPIP-21, Fire Emergency Procedure, Revision 3

Mechanical Section Instruction MSI-0-000-PRO005, Electrical Equipment Room Emergency Ventilation Following an Appendix R Fire Event, Revision 1

Operating Instruction 0-OI-26, High Pressure Fire Protection System, Revision 52

Operating Instruction 0-OI-82, Standby Diesel Generator System, Revision 72

Safe Shutdown Instruction 2/3-SSI-001, Safe Shutdown Instructions, Revision 3

Safe Shutdown Instruction 2/3-SSI-5, Unit 1, 4KV Shutdown Board A Room, Revision 3

Safe Shutdown Instruction 2/3-SSI-12, Unit 3, 480V Reactor MOV Board Room 3B, Revision 3

Safe Shutdown Instruction 2/3-SSI-16, Control Building Fire El. 593 Through El. 617, Revision 4

Safe Shutdown Instruction 2/3-SSI-18, Unit 2 Battery and Battery Board Room, Revision 4

Safe Shutdown Instruction 2/3-SSI-22, Unit 3, 4160V Shutdown Board 3EA and 3EB Room, Revision 3

Site Standard Practice, SSP-9.4, Plant Modifications and Design Change Control, Revision 28

TVAN Standard Department Procedure, MMDP-1, Maintenance Management System, Revision 2

TVAN Standard Programs and Processes, SPP-9.3, Plant Modifications and Engineering Change Control, Revision 1B1

TVAN Standard Programs and Processes, SPP-9.4, 10CFR50.59 Evaluations of Changes, Tests, and Experiments, Revision 2

TVAN Standard Programs and Processes, SPP-9.5, Temporary Alterations, Revision 2

TVAN Standard Programs and Processes, SPP-10.9, Control of Fire Protection Impairments, Revision 0

Fire Brigade Pre-plan No. RX1-621, Unit 1 Reactor Building, Elevation 621'-0, Revision 3

Fire Brigade Pre-plan No. RX3-593, Unit 3 Reactor Building, Elevation 593'-0, Revision 3

Fire Brigade Pre-plan No. CB1-593, Unit 1 Control Building, Elevation 593'-0, Revision 3

Fire Brigade Pre-plan No. CB1-606, Unit 1 Control Building, Elevation 606'-0, Revision 2

Fire Brigade Pre-plan No. CB2-593, Unit 2 Control Building, Cable Spreading Room, Elevation 606'-0, Revision 2

Fire Brigade Pre-plan No. CB2-606, Unit 2 Control Building, Elevation 593'-0, Revision 3

Fire Brigade Pre-plan No. DG3-565, Unit 3 Standby Diesel Generator Building, Elevation 565'-0, Revision 2

Fire Brigade Pre-plan No. DG3-583, Unit 3 Standby Diesel Generator Building, Elevation 583'-0, Revision 2

CALCULATIONS

BFN-67-D053, Diesel Generator Temperature Rise Evaluation at Full Load, dated April 11, 1986
BFN-ED-N0224-890050, Appendix R Analysis for Intraplant Communication System, Revision 3, dated April 29, 1992

BFN-ED-Q0999-870077, Analysis of the Auxiliary and Control Power System to Identify Associated Circuits - 10 CFR 50, Appendix R, Revision 19, dated January 29, 1997 [a coordination study].

BFN-ED-N0999-880700, Normal DC Control Power for Associated Circuit Analysis U1, U2 & U3, Revision 12, dated December 18, 1995 [demonstrates that dc control power is available, i.e. not damaged by fire, for the required equipment].

BFN-ED-Q0999-940040, Appendix R Computerized Safe Shutdown Separation Analysis, Revision 9, dated September 3, 1999 [covers Units 2 & 3, uses INDMS]

BFN-ED-Q0999-940040, “Appendix E - (A) Availability of Scram and Isolation Evaluation; (B) Three-phase Hot Shorts for Non High-Low Pressure Interfaces,” B22 911018 007, 9 pages.

BFN-ED-Q2999-880574, Class 1E Electrical Boards Margin Study for 4kV, 480 V, 120 VAC and 250 V, 125 V, 24 VDC Systems, Revision 5, dated February 5, 1996 [addresses high multiple impedance faults, Unit 2].

BFN-MD-Q0026-890024, Appendix R Fire Suppression Flooding and Drainage Damage Evaluation Report, dated June 30, 1989

BFN-MD-Q0100-890003, Evaluation of Building Gaps and Seal Details, dated February 14, 1989

BFN-MD-Q0100-980006, Evaluation of Penetration Seals, dated April 15, 1998

BFN-MD-Q0303-880381, Fire Hazards Analysis for Structural Steel, dated June 15, 1989

BFN-ND-Q0999-920116, Appendix R Manual Action Requirements, dated July 23, 1998.

Chemetron Corporation, Low Pressure Carbon Dioxide Flow Calculations, FL-15745, dated April 20, 1976

PLC, Browns Ferry Nuclear Plant Fire Detection and Alarm System Design Spacing , dated June 1992

DRAWINGS

0-45E400-RW-02, Revision 4
0-45E400-RW-04, Revision 2
0-45E400-RW-09, Revision 3
0-46E403-17 & 18, Revision 1
0-46E610-1, Revision 22
0-46E644-1, 2 & 3, Revision 12
3-47E817-1, Revision 037
2-47E817-1, Revision 046
3-47E813-1, Revision 042
2-47E813-1, Revision 040
3-47E812-1, Revision 050
3-47E811-1, Revision 048
2-47E812-1, Revision 044
2-47E811-1, Revision 052
3-47E837-1, Revision 006
2-47E837-1, Revision 017
3-47E810-1, Revision 032
2-47E810-1, Revision 026
3-47E804-1, Revision 038
2-47E804-1, Revision 046
3-47E801-1, Revision 019
2-47E801-1, Revision 019
3-47E862-1, Revision 026
1-47E862-1, Revision 013
2-47E862-1, Revision 018
3-47E859-2, Revision 022
3-47E859-1, Revision 035
2-47E859-1, Revision 024
2-47E818-1, Revision 020
3-47E858-1, Revision 024
2-47E858-1, Revision 017
3-47E844-2, Revision 036
2-47E844-2, Revision 023
3-47E844-1, Revision 022
2-47E844-1, Revision 026
3-47E818-1, Revision 021
0-47W391-14, Revision 1
0-47W930-1, Revision 3
0-47W931-7, Revision 0
0-47W600-256, Revision 0
2-47W491-39 & 43, Revision 0
2-47W2392-320, Revision 3

ENGINEERING EVALUATIONS

Browns Ferry Nuclear Power Plant Fire Area/Zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Fire Area 5, Revision 7

Browns Ferry Nuclear Power Plant Fire Area/Zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Fire Area 6, Revision 7

Browns Ferry Nuclear Power Plant Fire Area/Zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Fire Area 7, Revision 7

Engineering Evaluation of the Seismic Gaps , dated July 27, 1988

Engineering Evaluation of Bus Duct Penetrations, dated July 27, 1988

Engineering Evaluation of Fire Barrier Walls Separating the Battery and Battery Board Rooms From the Control Bay Corridor, dated July 11, 1989

Engineering Evaluation of Appendix R Fire Dampers and Summary Report, dated September 8, 1989

Engineering Evaluation of Appendix R Unducted Ventilation Openings, dated February 24, 1989

Engineering Evaluation of Fire Door Integrity for Fire Rating, dated September 1, 1993

10 CFR PART 50 APPENDIX R, EXEMPTION REQUESTS

Browns Ferry Units 2 and 3 Safe Shutdown Analysis, Section 9.0, Exemptions, Revision 15

OTHER DOCUMENTS

Fire Protection Report Vol. 1, Browns Ferry Nuclear Plant Units 2 and 3 - Fire Hazard Analysis, Section 2, Revision 0015

Fire Protection Report Vol. 1, Browns Ferry Nuclear Plant Units 2 and 3 - Safe Shutdown Analysis, Section 3, Revision 0015

Fire Protection Report Vol. 1, Browns Ferry Nuclear Plant Units 2 and 3 - Appendix R Safe Shutdown Program, Section 4, Revision 0015

Browns Ferry Nuclear Plant Updated Final Safety Analysis Report, Section 4.7, Reactor Core Isolation Cooling System

Browns Ferry Nuclear Plant Updated Final Safety Analysis Report, Section 4.8, Residual Heat Removal System

Browns Ferry Nuclear Plant Updated Final Safety Analysis Report, Section 6.4, Description, Emergency Core Cooling System

Browns Ferry Nuclear Plant - Individual Plant Examination for External Events (IPEEE) - Internal Fires, High Winds, Floods, Transportation and Nearby Facility Accidents, July 1995, Tennessee Valley Authority, PDR 9508020141 950724

Response to Generic Letter 88-20, Supplement 4 - Individual Plant Examination for External Events (IPEEE - Fire) - Browns Ferry Nuclear Plant Unit 3 - Fire Induced Vulnerability Evaluation (FIVE) Methodology, (undated ca. 1999).

Response to Request for Additional Information on Individual Plant Examination for External Events: Seismic; Fire; High Winds, Floods, and Other External Events - November 1998, Revision 0, Fire Questions, page 5 ff., PDR 9812080124 981125.

BNFP 10 CFR 50.65 Maintenance Rule Reliability and Availability Data:
RHR Service Water System 023 - Attachment 5
Emergency Equipment Cooling Water System 067 (Pump) - Attachment 15
Residual Heat Removal System 074 - Attachment 18
Emergency Diesel Generator System 082 - Attachment 24

Licensed Operator Training Lesson Plan OPL171, Safe Shutdown Instruction 001 for Unit 2/3 (Appendix R Fire), Revision 31

Fire Brigade Drill Data Sheets, 1998, 1999, and 2000

Fire Brigade Training and Physical Examination Data Sheets, 1998, 1999, and 2000

Browns Ferry Nuclear Plant Design Criteria, No. BFN-50-7026, High Pressure Fire Protection System, Revision 3

Browns Ferry Nuclear Plant Design Criteria, No. BFN-50-7039, CO_2 Storage, Fire Protection, and Purging System, Revision 3

Browns Ferry Nuclear Plant Design Criteria, No. BFN-50-7308, Fire Alarm and Detection System, Revision 0

Browns Ferry Nuclear Plant - Summary of Deviations from NFPA Code for BFN, dated August 3, 1988

W79 980424-001, Power Uprate Evaluation Report for Tennessee Valley Authority - Browns Ferry Units 2 and 3 - Appendix R Fire Protection, GE-NE-B13-01866-16, dated April 1998

LIST OF ACRONYMS USED

AOI	Abnormal Operating Instruction
BFN	Browns Ferry Nuclear Plant
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
EECW	Emergency Equipment Cooling Water
EDG	Emergency Diesel Generator
GL	Generic Letter
HVAC	Heating, Ventilation and Air Conditioning
IPEEE	Individual Plant Examination External Events
KV	Kilovolt
LPCI	Low Pressure Coolant Injection
MCR	Main Control Room
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSRV	Main Steam Relief Valve
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NSR	Non-Safety Related
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
SDP	Significance Determination Process
SPP	Standard Programs and Processes
SRA	Senior Reactor Analyst
SSA	Safe Shutdown Analysis
SSD	Safe Shutdown Systems
SSI	Safe Shutdown Instruction
SSP	Site Standard Practice
TS	Technical Specification
TSC	Technical Support Center
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volts
VAC	Volts Alternating Current
VDC	Volts Direct Current

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.