UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

October 30, 2006

Tennessee Valley Authority
ATTN: Mr. Karl W. Singer
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 NO. 05000260/2006014 AND 05000296/2006014

Dear Mr. Singer:

On September 15, 2006, the United States Nuclear Regulatory Commission (NRC) completed a triennial fire protection inspection at your operating Browns Ferry Units 2 and 3 reactor facilities. The enclosed report documents the inspection results which were discussed at an exit meeting on that date, with Mr. R. G. Jones and other members of your staff.

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

This report documents two NRC-identified findings of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section IV.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this report, with the basis of your denial, to the Nuclear Regulatory commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Browns Ferry Nuclear Plant.

In accordance with 10 CFR 2.390 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 Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

D. Charles Payne, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000260/2006014 and 05000296/2006014
w/Attachment; Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-260, 50-296
License Nos.:	DPR-52, DPR-68
Report Nos.:	05000260/2006014, 05000296/2006014
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 2 and 3
Location:	Athens, Alabama
Dates:	August 28 - September 1, 2006 (Week 1) September 11- 15, 2006 (Week 2)
Inspectors:	G. MacDonald, Senior Reactor Inspector (Lead Inspector) R. Fanner, Reactor Inspector N. Merriweather, Senior Reactor Inspector R. Rodriguez, Reactor Inspector G. Wiseman, Senior Reactor Inspector D. Mas-Penaranda, Reactor Inspector
Accompanying	T. Harrison, Nuclear Safety Professional
Approved by:	D. Charles Payne, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000260/2006-014, 05000296/2006-014; 08/28-9/1/2006 and 09/11-15/2006; Browns Ferry Nuclear Plant, Units 2 and 3; Fire Protection.

This report covers a two-week triennial fire protection inspection by six regional inspectors. Two Green findings, both of which were non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process" Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation (NCV) of Unit 2 Operating License Condition 2.C.14 for an inadequate Safe Shutdown Instruction (SSI) which directed the operator to align credited 4 kV Shutdown Board (SD BD) "A" to its alternate supply (Shutdown Bus 2). This could connect 4kV SD BD "A" to a fire-induced fault and result in a lockout of the "A" emergency diesel generator (EDG), one of two required EDGs for Unit 2 to complete Safe Shutdown (SSD) for a fire in fire area 8 (FA 8). The licensee established compensatory measures for the issue and entered this performance deficiency into their corrective action program (CAP) for resolution.

  The finding is more than minor because this performance deficiency is associated with the reactor safety mitigating system cornerstone attribute of protection against external events, i.e., fire. It also affected the cornerstone objective of ensuring availability of systems that respond to events in that 4kV SD BD "A" could have been de-energized and locked out in response to a postulated fire in FA 8. The inspectors determined that the issue was of very low safety significance (Green) because the finding was judged to have a low degradation impact on safe shutdown in that the deficiency would not have caused a failure of the SSD strategy for FA 8. There was a very short period of time when the fault could have affected 4 kV SD BD "A" and there was significant recovery time available (approximately 2 hours) due to the required SSD loads not being powered from 4 kV SD BD "A". (Section 1R05.01.b.2)

- Green. The team identified a non-cited violation (NCV) of Unit 2 Operating License Condition 2.C.14 and Unit 3 Operating License Condition 2.C.7 for failure to have adequate communications to implement alternate shutdown for a fire in fire area (FA) 16 using procedure 2/3-SSI-16.

  This issue is a performance deficiency because the cell phone system was unreliable and the F4 portable radio system was not credited for a fire in FA 16. The finding is greater than minor because it affected the ability of the licensee to

maintain communications for a fire in FA 16 and is associated with the mitigating systems cornerstone and respective attribute of protection against external factors, i.e., fire in that degraded communications would impact the ability to achieve SSD following a fire. This finding was determined to be a finding of very low safety significance (Green) because it only affected the ability to reach and maintain cold shutdown conditions due to the availability of alternate communications measures (F4 radios) for a time period sufficient to achieve hot shutdown conditions. (Section 1R05.08)

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R05 Fire Protection

This report presents the results of a triennial fire protection inspection conducted in accordance with NRC Inspection Procedure (IP) 71111.05T, "Fire Protection." The objective of the inspection was to assess whether the licensee has implemented an adequate fire protection program and that post-fire safe shutdown capabilities have been established and are being properly maintained at the Browns Ferry Nuclear facility. The following fire areas (FAs) and/or fire zones (FZs) were selected for detailed review based on both risk insights from the licensee's Individual Plant Examination of External Events and in-plant tours by the inspectors:

- FZ 2-5, Unit 2 Reactor Building Elevations 621 and 639 feet
- FA 8, Unit 2 4160 Volt Electric Board Room 2B
- FA 16, Control Building Elevation 593 feet

Section 71111.05-05 of the IP specifies a minimum sample size of three fire areas. Inspection of these areas/zones fulfills the procedure completion criteria. The inspection team evaluated the licensee's fire protection program (FPP) against applicable requirements which include plant Technical Specifications (TS); Units 1 and 2 Operating License Condition 2.C.(14) for Unit 2 and 2.C.(7) for Unit 3; NRC Safety Evaluation Reports (SERs); Appendix R and Section 50.48 to Title 10 of the Code of Federal Regulations (CFR) Part 50 (hereafter referred to as 10 CFR 50); and NRC Approved Exemptions to 10 CFR 50, Appendix R (hereafter referred to as Appendix R). The team also reviewed related documents that include the Fire Hazards Analysis (FHA), Post-Fire Safe Shutdown Analysis (SSA) Report; and the Browns Ferry Fire Protection Program Plan.

Those specific documents reviewed by the team are listed in the Attachment.

.01 Post-Fire Safe Shutdown From Main Control Room

a. Inspection Scope

Methodology

The team reviewed the FHA, SSA, operating procedures, piping and instrumentation drawings (P&IDs), electrical drawings, the FPP and other supporting documents for fires in FA 8, and FZ 2-5 to verify that the shutdown methodology properly identified the components and systems necessary to achieve and maintain safe shutdown conditions. This review included verification that shutdown from the main control room (MCR) could be performed both with and without the availability of offsite power. Plant walkdowns were also performed to verify that the plant configuration was consistent with that described in safe shutdown and fire hazards analyses. These inspection activities focused on ensuring the adequacy of systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring instrumentation and support

systems functions. The team reviewed the systems and components credited for use during this shutdown method to verify that they would remain free from fire damage.

Operation Implementation

The team reviewed the training program for licensed and non-licensed operators to verify that the training reinforced the shutdown methodology in the SSA and fire procedures for the selected FAs and FZs.

The team reviewed the adequacy of procedures utilized for post-fire safe shutdown (SSD) and performed an independent walk through of procedure steps to ensure the implementation and human factors adequacy of the procedures. The team also reviewed select operator actions to verify that the operators could reasonably be expected to perform the specific actions within the time required to maintain plant parameters within specified limits. Specific actions which were verified included: remote confirmatory reactor scram, recirculation pump trip and main steam isolation valve closure; restoration of alternating current (AC) electrical power and electrical distribution system alignment; remote starting of residual heat removal (RHR) and RHR service water (RHRSW) pumps; remote operation of the low pressure coolant injection (LPCI) injection valve, and manual operation of RHRSW outlet valve and containment atmosphere dilution (CAD) valves. Communications and emergency lighting necessary to support accomplishment of the manual actions were reviewed. The team reviewed the licensee's walkdown validation packages for FZ 2-5, FA 8 and FA 16.

Specific procedures and attachments reviewed are listed in the Attachment.

The team also examined operator manual actions to ensure that they had been properly reviewed and approved by NRC, as applicable, and that the actions could be implemented in accordance with plant procedures in the time necessary to support the SSD method for each fire area. The team reviewed the manual actions to verify that those actions met the criteria in Attachment 2 of NRC Inspection Procedure 71111.05T and that problem evaluation report (PER) 101631, Appendix R Section III.G.2 Operator Manual Actions, remained open to track resolution of the manual action issues at the Browns Ferry facility.

b. Findings

1) Postulated Fire-Induced Circuit Failures Could Prevent the Operator from Opening LPCI Injection Valve 2-FCV-074-053 from the Main Control and Result in Failure to Establish LPCI Flow Into the Reactor Vessel

Introduction: The team identified an unresolved item (URI) associated with potential fire-induced electrical circuit failures in the control circuit for LPCI Injection Valve 2-FCV-074-053. The team postulated a fire in Fire Area 8, which could result in fire-induced electrical circuit faults in two control cables associated with the pressure interlock bypass circuit of the LPCI injection valve which, if damaged by fire, could result

in failure to open the valve from the main control room and loss of LPCI flow capability. The cables supply power to the relay logic for a pressure permissive that allows opening and closing the valve from the main control room if reactor pressure is less than 450 psig.

Description: The licensee's Safe Shutdown Instruction (SSI) 2/3-SSI-8 credits operator action to open LPCI Injection Valve 2-FCV-074-053 from the main control room (MCR) to establish LPCI flow into the reactor vessel during hot shutdown for a fire in FA 8. The inspectors concluded that fire-induced failure of two control power circuits could result in a loss of power to the relay logic that provides the interlock bypass for LPCI Injection Valve 2-FCV-074-053. If this occurred, the operator would not be able to open LPCI Injection Valve 2-FCV-074-53 from the main control room if the other LPCI Injection Valve 2-FCV-074-052 was open. The operator would have to open the valve locally at the valve or at the 480 Volt Reactor Motor Operated Valve (MOV) Board in the Reactor Building. However, the SSI did not address this potential failure to open LPCI Injection Valve 2-FCV-074-053 from the main control room in the event of fire damage to the control cables.

The team reviewed cable routing information for LPCI Injection Valve 2-FCV-074-053 in FA 8 and determined that two cables associated with the control logic for the valve were routed in FA 8 without protection from potential fire damage. Both thermoplastic type cables had been routed in the same conduit above the 250 Volt DC Motor Control Center (MCC) in FA 8. These control cables, 2ES1050-I and 2ES769-I, are associated with the interlock bypass logic for LPCI injection valve 2-FCV-074-053 such that the failure of either cable will prevent LPCI Injection Valve 2-FCV-074-053 from being opened from the main control room. The operator would have to open the valve locally at the valve or at the 480 Volt Reactor MOV Board in the Reactor Building. The SSD analysis requires that both LPCI Injection valves 2-FCV-074-052 and -053 be opened simultaneously. The purpose of the pressure interlock is to prevent the simultaneous opening of both LPCI valves 2-FCV-74-0052 and -53 from the main control room when reactor pressure is greater than 450 psig. The interlock will be bypassed automatically when reactor vessel pressure is less than 450 psig allowing both valves to be opened simultaneously.

Analysis: The failure to protect control circuit cables 2ES1050-I and 2ES769-I in FA 8 is a performance deficiency due to potential fire damage to the control circuit for the LPCI injection valve which could impact the ability to open 2-FCV-074-53 from the MCR as required by SSI 2/3-SSI-8. The licensee should have identified this potential circuit problem in both the SSD analysis and the manual action calculation of record, which forms the basis for the safe shutdown procedures. The performance deficiency is associated with the reactor safety mitigating system cornerstone attribute of protection against external events, i.e., fire. It also affects the cornerstone objective of ensuring the availability of systems that respond to events in that the LPCI injection valve could not be operated from the main control room as described in the fire response procedure as a result of postulated fire damage.

Enforcement: 10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R, Section III.G. Section III.G.1 specifies that fire protection features shall be provided for SSD components and these features shall be capable of limiting fire damage such that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station is free of fire damage.

Contrary to the above, on September 15, 2006, control cables associated with a train of systems necessary to achieve and/or maintain hot shutdown conditions from the main control room were not protected from fire damage. Lack of cable protection could result in a failure to open LPCI Injection Valve 2-FCV-074-053 from the main control room and subsequent failure to establish LPCI flow into the reactor vessel.

Pursuant to NRC Manual Chapter 0305, Section 06.06.a.2, "Violations in Selected Areas of Interest Qualifying for Enforcement Discretion," the NRC may exercise enforcement discretion for violations involving fire protection circuits if the licensee acknowledges the violation, enters it into the corrective action program, takes appropriate compensatory actions and the finding is not evaluated to be Red. The licensee has acknowledged the violation, entered the findings into the corrective action program (PER 110612) and completed timely corrective action. This item is open pending further NRC review to determine whether discretion for fire induced circuit problems is applicable and to complete fire modeling and risk evaluation to determine the credibility of the potential fire scenarios. The item is identified as URI 05000260/2006014-001, Postulated Fire-Induced Circuit Failures Could Prevent the Operator From Opening LPCI Injection Valve 2-FCV-074-053 From The Main Control and Result in Failure to Establish LPCI Flow Into the Reactor Vessel.

2) Safe Shutdown Instructions Direct the Operators to Connect Credited 4KV SD BD "A" to a Potentially Fire-Induced Fault Which Could Result in a Lockout of the "A" EDG.

Introduction: The team identified a Green NCV of Unit 2 Operating License Condition 2.C.14 associated with SSI 2/3-SSI-8 which directed the operators to align credited 4KV Shutdown Board (SD BD) "A" to its alternate supply (Shutdown Bus 2). This could connect credited 4KV SD BD "A" to a fire induced fault and result in a lockout of the "A" EDG, one of the two required EDGs necessary for Unit 2 to complete SSD for a fire in FA 8.

Description: The licensee's Fire Protection Report SSA credits 4KV SD BDs "A" and "B" and their respective EDGs along with SSI 2/3-SSI-8 to shutdown Unit 2 for a fire in FA 8. Step 2.7 of procedure 2/3-SSI-8 directs the Unit 2 Operator to transfer 4KV SD BD "A" to its alternate supply (SD Bus 2) by closing the 4KV SD BD "A" Alternate Feeder Breaker 1716 and then opening the 4KV SD BD "A" Normal Feeder Breaker 1614. This action electrically momentarily connects the credited 4KV SD BD "A" to 4KV SD BD "D" through SD Bus 2.

While in this configuration, a postulated fire in FA 8 could result in a fire-induced electrical fault in 4KV SD BD "D". The fault current would be sensed by the overcurrent relays at 4KV SD BD "A" tripping all feeder breakers and sending a signal to the lockout relay resulting in a bus lockout.

When 4KV SD BD "A" is de-energized, the "A" EDG would start, come up to rated speed and voltage, but the lockout relay would prevent its output breaker from closing. In order to close the "A" EDG output breaker and energize 4KV SD BD "A", the operators would have to manually reset the lockout relay locally at 4KV SD BD "A". The Unit 2 SSD strategy requires 4KV SD BD "A" to power the EDG's auxiliaries and the vital instrumentation bus. The EDG's auxiliaries are only needed to maintain the EDG ready to start. Once it starts, they are no longer needed. The vital instrumentation bus is backed up by the station vital batteries which can power the bus for up to two hours. Compensatory action has been established and PER 110536 was initiated for resolution.

Analysis: The issue is a performance deficiency due to SSI 2/3-SSI-8 directing the operators to connect the credited 4KV SD BD "A" to a potentially fire-induced fault which could result in a bus lockout. The performance deficiency is associated with the reactor safety mitigating system cornerstone attribute of protection against external events, i.e., fire. It also affects the cornerstone objective of ensuring the availability of systems that respond to events in that 4KV SD BD "A" could have been de-energized and locked out as a result of implementing SSI 2/3-SSI-8 in response to a postulated fire in FA 8. Therefore, the finding is greater than minor.

The inspectors determined that the issue was of very low safety significance (Green) because the finding was judged to have a low degradation impact on SSD in that the deficiency would not have caused a failure of the SSD strategy for FA 8. Some of the factors causing the issue to be of very low safety significance were:

- In order for the overcurrent relays on 4KV SD BD "A" to sense a fault current on 4KV SD BD "D", both buses must be electrically connected through SD Bus 2, i.e., when breakers 1716 and 1614 are closed. There is a very small period of time during which this situation occurs as directed by procedure 2/3-SSI-8.

- If 4KV SD BD "A" is de-energized and locked out, the operators would have approximately two hours (vital instrumentation bus powered by the station vital batteries) to reset the lockout relay and close the "A" EDG output breaker. EDG auxiliaries would not be required as the EDG would already be operating.

Enforcement: Operating License Condition 2.C.14 for Unit 2 requires that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program (FPP) as described in the FSAR for BFN as approved in the SE dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989. The Fire Protection Program is described in the Fire Protection Report. Section 8.3 of the Fire Protection Report Vol. I, Revision 36 established that

safe shutdown procedures are available in the event a fire occurs in safety-related areas of the plant to accomplish SSD.

Contrary to the above, on September 15, 2006, SSI procedure 2/3-SSI-8 directed the operators to align the credited 4KV SD BD "A" to its alternate supply (SD Bus 2). This could connect the credited 4KV SD BD "A" to a fire-induced fault which could result in a lockout of the "A" EDG, one of the two required EDGs necessary for Unit 2 to complete SSD for a fire in FA 8. Because the inadequate safe shutdown procedure finding was of very low safety significance and has been entered into the corrective action program as PER 110536, this violation is being treated as an NCV consistent with Section IV.A.1 of the NRC Enforcement Policy : NCV 05000260/2006014-002, Safe Shutdown Instruction Directs Operators to Connect 4KV SD BD "A" to a Potentially Fire-Induced Fault in FA 8.

3) Unapproved Operator Manual Actions (OMAs) in Appendix R Section III.G.2 Fire Area 2 Fire Zone 2-5 (FA 2/FZ 2-5)

Introduction: The team identified an unresolved item (URI) involving reliance on operator manual actions (OMAs) in lieu of the cable protection required by Appendix R, Section III.G.2. Examples of this problem applied in FA 2/FZ 2-5, however the extent of this problem went beyond the fire areas/zones selected for this inspection. Use of OMAs is a generic industry issue discussed in Federal Register Notice 71 FR 11169, dated March 6, 2006. Notice 71 FR 11169 provides for enforcement discretion if specified conditions are met. Existing URI 05000259/2006012-001 is open to monitor OMA resolution for Unit 1, and a new URI was opened to monitor resolution of the OMA issue and determine whether enforcement discretion can be applied for Units 2 and 3.

Description: The licensee's strategy for accomplishing post fire SSD for a postulated fire in FZ 2-5 utilized OMAs outside of the MCR which were not approved by the NRC. The OMAs were:

- Remote confirmatory verification of reactor scram, recirculation pump trip and main steam isolation valve closure
- Remote restoration of electrical power (starting EDGs), electrical distribution system alignment and resetting battery chargers
- Remote starting of RHR and RHRSW pumps
- Remote operation of the RHR LPCI injection valve
- Manual operation of the RHRSW outlet valve and CAD supply valves and aligning and starting control bay and electric board room air handling units

The team concluded that the OMAs met the criteria of IP 71111.05T for an acceptable compensatory measure. The licensee's SSI verification and validation review only

documented the ability to complete the procedures within the required time, it did not document verification of the other elements of NRC IP 71111.05T Attachment 2. PER 110658 was initiated regarding SSI validation. The team noted that all unapproved OMAs had been entered into the corrective action program as PER 101631.

Analysis: The issue is a performance deficiency, because even though it involves a generic industry issue, the licensee should not have used OMAs in lieu of protecting cables (i.e., fixed fire protection features) important to safe shutdown from potential fire damage without prior NRC approval. The performance deficiency is associated with the reactor safety mitigating system cornerstone attribute of protection against external events, i.e., fire. It also affects the cornerstone objective of ensuring reliability of systems that respond to events in that OMAs are less reliable than fixed fire protection features.

Enforcement: 10 CFR 50.48(b)(1) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R, Section III.G. Section III.G.2 applies to the ability to achieve and maintain hot SSD from the control room during a fire. It states, in part, that where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of three means of protecting cables to ensure that one of the redundant trains is free of fire damage shall be provided. The three means involve physical protection or separation of cables to preclude fire damage - III.G.2 does not allow MOAs in lieu of protection.

Contrary to the above, on September 15, 2006, cables of systems necessary to achieve and/or maintain hot standby conditions were not protected from fire damage. Lack of cable protection could result in de-energizing or losing remote control over EDGs, SD BD feeder breakers, RHR pumps, RHRSW pumps, battery chargers, ventilation equipment and valves important to achieving and/or maintaining hot standby conditions. Instead the plant relied on OMAs to recover from the potential adverse effects. However, in the March 6, 2006, Federal Register Notice (71 FR 11169) that withdrew the proposed rulemaking on manual actions, the NRC stated that, for cases involving feasible manual actions, the licensee would be eligible for enforcement discretion if they initiated corrective actions within six months of the issue date of the notice and completed all corrective actions within three years. The licensee has entered this item into their corrective action program as PER 101631 and indicated that they plan to require that all manual actions identified in manual action calculation NDQ0999920116 be re-evaluated for compliance with 10 CFR 50 Appendix R, Section III.G.2. The licensee has indicated that they plan to either submit exemption requests to 10 CFR 50, Appendix R, for any unapproved manual actions or implement modifications to eliminate the need for the manual action.

The URI was opened to monitor resolution of the OMA issue and determine whether enforcement discretion can be applied.

It is identified as URI 05000260, 296/2006014-003, Unapproved Local Manual Operator Actions in Lieu of Cable Protection for a Fire Area Subject to the Requirements of Appendix R Section III.G.2.

.02 Protection of Safe Shutdown Capabilities

a. Inspection Scope

The team reviewed the FHA, post-fire SSA, and supporting drawings and documentation to verify that SSD capabilities were properly protected. The team verified that separation requirements of the FPP were maintained for the credited safe shutdown equipment and their supporting power, control and instrumentation cables. This review included an assessment of the adequacy of the selected systems for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring, and associated support system functions.

For the selected fire areas/zones, the team evaluated the potential for fires, the combustible fire load characteristics, and the potential exposure fire severity. The team reviewed the licensee procedures, plant smoking policy, and programs for the control of ignition sources and transient combustibles. These reviews were conducted to assess their effectiveness in preventing fires and in controlling combustible loading within limits established in the FPP. The documents reviewed are listed in the Attachment.

The team performed plant walkdowns to verify (1) the material condition of fire protection systems and equipment, (2) the storage of permanent and transient combustible materials, and (3) the administrative controls for limiting fire hazards, combustible waste collection, housekeeping practices, and cleanliness conditions were being implemented consistent with administrative procedures, and other FPP procedures.

b. Findings

No findings of significance were identified.

.03 Passive Fire Protection

Title 10 CFR Part 50, Appendix R, Section III.G.1, required that Structures, Systems, and Components (SSCs) important to safe shutdown be provided with fire protection features capable of limiting fire damage to ensure that one train of systems necessary to achieve and maintain hot shutdown conditions remained free of fire damage. Options for providing this level of FP were delineated in 10 CFR Part 50, Appendix R, Section III.G.2.

a. Inspection Scope

The team inspected the material condition of accessible passive fire barriers surrounding and within the fire areas selected for review. Barriers in use included walls, ceilings, floors, structural steel, mechanical and electrical penetration seals, doors, and dampers. Construction details and fire endurance test data which established the ratings of fire barriers were reviewed by the team.

Engineering evaluations and relevant exemptions described in NRC Safety Evaluation Reports related to fire barriers were reviewed. Where applicable, the team examined installed barriers to compare the configuration of the barrier to the rated configuration to verify that the as-built configurations met design requirements, license commitments, standard industry practices and were either properly evaluated or qualified by appropriate fire endurance tests. In addition, a sample of completed surveillance and maintenance procedures for selected fire doors, dampers, and penetration seals in the selected fire areas/zones was reviewed to ensure that these passive fire barrier features were properly inspected and maintained. The fire protection features included in the review are listed in the Attachment.

b. Findings

No findings of significance were identified.

.04 Active Fire Protection

a. Inspection Scope

The team reviewed flow diagrams, cable routing information, system operating instructions, operational valve lineup procedures, and vendor documentation associated with the fire pumps and fire protection water supply system. Using operating and valve alignment procedures, team members toured selected fire pumps and portions of the fire main piping system to evaluate material condition, consistency of as-built configurations with engineering drawings, and to verify correct system valve lineups. The common fire protection water delivery and supply components were reviewed to assess if they could be damaged or inhibited by fire-induced failures of electrical power supplies or control circuits. In addition, the team reviewed periodic surveillance and operability flow test data for the electric and diesel fire pumps and fire main loop to assess whether the test program was sufficient to validate proper operation of the fire protection water supply system in accordance with its design requirements.

Through in-plant observation of systems, design document review and reference to the applicable National Fire Protection Association (NFPA) codes and standards, the team evaluated the material condition and operational lineup of fire detection and suppression systems. The appropriateness of detection and suppression methods for the category of fire hazards in the selected fire areas was evaluated. The pre-action sprinkler and local application fire hose stations FA 2/FZ 2-5 were inspected.

During plant tours, the team observed placement of the fire hoses and extinguishers to verify they were not blocked and were consistent with the pre-fire plans and FPP documents. Fire suppression systems were evaluated from source to discharge device including hydraulic calculations performed by the licensee to demonstrate adequate flow, pressure and water distribution. The team reviewed the fire brigade staging and dress-out areas to assess the operational readiness of fire fighting and smoke control equipment. The fire brigade personal protective equipment, self-contained breathing apparatuses (SCBAs) and SCBA cylinder refill capability were reviewed for adequacy and functionality. The team also reviewed operator and fire brigade staffing, fire brigade response, offsite fire department communications and staging procedures, fire fighting pre-plans, fire brigade qualification training, and the fire brigade drill program procedures. Four fire brigade response-to-drill scenarios and associated brigade drill evaluations/critiques that transpired over the last 12 months were reviewed. The team walked down fire brigade pre-fire plans and smoke removal procedures for the selected fire areas to determine if appropriate information was provided to fire brigade members and plant operators to identify safe shutdown equipment and instrumentation, and to facilitate suppression of a fire that could impact post-fire safe shutdown (SSD). The documents included in the review are listed in the Attachment.

b. Findings

No findings of significance were identified.

.05 Protection From Damage From Fire Suppression Activities

a. Inspection Scope

The team performed document reviews and in-plant walkdowns to verify that redundant trains of systems required for hot shutdown were not subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems. Specifically, the team:

- Reviewed fire damper location and vendor detail drawings, and heating, ventilation, and air conditioning (HVAC) system drawings to verify that a fire in one of the selected fire areas would not directly, through production of smoke, heat or hot gases, inhibit access to alternate shutdown equipment or performance of alternate safe shutdown operator actions by smoke migration through duct work from the area of a fire to adjacent plant areas/zones.

- Reviewed the physical configuration of electrical raceways and safe shutdown components in the selected fire areas to verify water from a pipe rupture, actuation of the automatic suppression system, or manual fire suppression activities would not directly cause damage to all redundant trains within the fire area or an adjacent plant area that could inhibit SSD (e.g., fire suppression caused flooding of other than the locally affected train).

- Reviewed floor drain locations and building drain system drawings to verify that adequate drainage is provided in areas protected by water suppression systems.

The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.06 Post-Fire Alternative Shutdown Capability From Outside MCR

a. Inspection Scope

Methodology

The team reviewed the FHA, SSA, SSIs, P&IDs, electrical drawings, the FPP and other supporting documents for a postulated fire in the Control Building Communications Room, FA 16, to verify that the shutdown methodology properly identified the components and systems necessary to achieve and maintain post fire SSD conditions. This review included verification that alternate shutdown from outside the MCR could be performed both with and without the availability of offsite power. Plant walkdowns were also performed to verify that the plant configuration was consistent with that described in the SSA and the FHA analyses. These inspection activities focused on ensuring the adequacy of systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring instrumentation and support systems functions. The team reviewed the systems and components credited for use during alternate shutdown to verify that they would remain free from fire damage.

Operation Implementation

The team reviewed the training program for licensed and non-licensed operators to verify that the training reinforced the shutdown methodology in the SSA and SSIs for a fire in FA 16 requiring MCR evacuation and implementation of alternate shutdown. Shift turnover records (dayshift and nightshift) for selected dates were reviewed to determine if minimum staffing requirements were met for SSD of both units irrespective of assigned fire brigade personnel. The actual staffing was compared to the minimum criteria defined in Section 5.2.2 of the Unit 2 and 3 Technical Specifications.

The team reviewed the adequacy of procedures utilized for post-fire alternate shutdown and performed an independent walk through of procedure steps to ensure the implementation and human factors adequacy of the procedures. The team also reviewed select operator actions to verify that the operators could reasonably be expected to perform the specific actions within the time required to maintain plant parameters within specified limits.

Actions verified included, restoration of AC electrical power, reactor pressure and level control, and implementation of alternate shutdown cooling operation from the Backup Control Panel. Communications and emergency lighting necessary to support accomplishment of alternate shutdown actions were reviewed.

The team also reviewed the periodic test procedures and test records of the alternative shutdown transfer capability and instrumentation and control functions to ensure the tests are adequate to verify the functionality of the alternative shutdown capability. Specific documents reviewed for alternative safe shutdown are listed in the Attachment.

b. Findings

No findings of significance were identified.

.07 Circuit Analyses

a. Inspection Scope

The team verified that the licensee performed a post-fire SSA for the selected fire areas and that the analysis appropriately identified the structures, systems and components important to achieving and maintaining safe shutdown. Additionally, the team verified that the licensee's analysis ensured that necessary electrical circuits were properly protected and that those circuits that could adversely impact safe shutdown due to hot shorts, shorts to ground or other failures were identified, evaluated and dispositioned to ensure spurious actuations would not prevent safe shutdown. The team's review considered fire and cable attributes, potential undesirable consequences and common power supply/bus concerns. Specific items included the credibility of the fire threat, cable insulation attributes, cable failure modes, and spurious operations resulting in flow diversion paths or loss of coolant events. The team reviewed system flow diagrams, electrical schematics, and cable routing data for a select sample of components, including a sample of instrumentation circuits required for post-fire SSD to verify that the licensee's program appropriately evaluated the adequacy of cable routing by fire area as described in the cable routing database files.

The team also reviewed breaker and fuse coordination data results and associated data sheets to confirm that the fuses and/or breakers were properly selected to electrically isolate SSD circuits from fire damage due to unprotected associated circuits.

The specific components reviewed are listed in the Attachment.

b. Findings

An URI involving unprotected control cables in the interlock bypass logic for LPCI Injection Valve 2-FCV-074-053 was identified and is discussed in Section 1R05.01.b.1.

.08 Communications

a. Inspection Scope

The team reviewed SSI procedures and associated documents (electrical drawings and calculations) to verify an adequate method of communications would be available to plant operators following a fire. During this review the team considered the effects of ambient noise levels, clarity of reception, reliability and coverage patterns.

An inventory list was reviewed to determine if an adequate amount of portable radios and cellular phones were available. The team reviewed the communications available at different locations. The team also verified that communications equipment such as repeaters, transmitters, etc. would not be affected by a fire in the selected fire areas / zones. Communications via portable radios and cellular phones were observed at the control building and other locations required by SSI procedure 2/3-SSI-16.

The team reviewed the plant communications systems that would be relied upon to support fire event notification and fire brigade fire fighting activities. The team also reviewed selected fire brigade drill evaluation/critique reports to assess proper operation and effectiveness of the fire brigade command post portable radio communications during fire drills and identify any history of operational or performance problems with radio communications during fire drills. In addition, the team verified the radio battery usage ratings for the fire brigade radios stored and maintained on charging stations.

b. Findings

Introduction: A Green NCV of Unit 2 Operating License Condition 2.C.14 and Unit 3 Operating License Condition 2.C.7 was identified for failure to have adequate communications to implement alternate shutdown for a fire in FA 16 using procedure 2/3-SSI-16.

Description: The Appendix R calculation Analysis for Intraplant Communication (EDN0244890050) directed the use of cellular phones for a fire in FA 16, Control Building/ Communications Room. The team performed a walkdown of the Unit 2 Backup Control Panel and other critical locations required by procedure 2/3-SSI-16 to verify the adequacy of the cellular phone communications system. During the walkdown the team found that the cellular phone system had poor or no coverage at these locations. With the unreliability of the cellular phone system at several alternate safe shutdown locations, the team concluded it would be extremely difficult to implement the procedure 2/3-SSI-16 within the allowed time frame.

As a result of the above finding, the licensee performed an evaluation and determined that the F4 portable radio system would be available for operations use during a fire in FA 16. The licensee determined that the F4 radio system would have some minor fire damage, but it would not make the system inoperable. The licensee indicated that the F4 repeater is located in the reactor building and would not be affected by a fire in FA 16.

Additionally, the radio system vendor had performed a test of a similar radio communication system to demonstrate that the system will function with the repeaters only. The licensee indicated that the F4 radio system has three hours of battery back up for the power supply which will be sufficient to reach cold shutdown plant conditions. The inspectors noted that the licensee has been aware of problems with cellular phone coverage and reliability as evidenced by PER 98547, initiated March 2, 2006 that documented problems with cellular phone use identified during Appendix R lighting and communications testing.

Analysis: This issue is a performance deficiency because the cell phone system was demonstrated to be unreliable and the F4 radio system was not credited in the SSA for use during a fire in FA 16. The team determined that the finding is greater than minor because it affected the ability of the licensee to maintain communications for a fire in FA 16 and is associated with the Mitigating Systems cornerstone and respective attribute of protection against external factors, i.e., fire in that degraded communications would impact the ability to achieve SSD following a fire. The team determined finding to be of very low safety significance (Green) because the finding only affected the ability to reach and maintain cold shutdown conditions due to the availability of alternate communications measures (F4 radios) for a time period sufficient to achieve hot shutdown conditions.

Enforcement: Operating License Conditions 2.C.14 (Unit 2) and 2.C.7 (Unit 3) require that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program as described in the FSAR for BFN as approved in the SE dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989. The Fire Protection Program is described in the Fire Protection Report (FPR). Section 4.8 of the FPR Revision 36, established that the communication system has been evaluated as part of Appendix R analysis (Calculation EDN0244890050 Analysis for Intraplant Communication) to ensure that adequate communication systems are available for safe shutdown for a fire in any plant area. The Analysis for Intraplant Communications established the Cellular Phone System as the credited system for achieving Unit 2 and Unit 3 SSD for a fire in FA16.

Contrary to the above, on September 15, 2006, the team identified that the credited communication system for achieving Unit 2 and Unit 3 SSD for a fire in FA 16 would not be available at critical locations required by procedure 2/3-SSI-16. Procedure 2/3-SSI-16 is used to implement safe shutdown of Units 2 and 3 for a fire in FA 16. Because the failure to identify adequate communications equipment to achieve and maintain Unit 2 and Unit 3 in a SSD condition for a fire in FA 16 was of very low safety significance and has been entered into the corrective action program as PER 110657. This violation is being treated as a Green NCV consistent with Section IV.A.1 of the NRC Enforcement Policy: NCV 05000260,296/2006014-004, Cellular Phone Communications Unreliable for Alternate Shutdown in FA 16.

.09 Emergency Lighting

a. Inspection Scope

The team observed the placement and coverage area of emergency lights throughout the selected fire areas to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation and/or instrumentation monitoring for post-fire SSD. The team also verified that the battery power supplies were rated with at least an eight-hour capacity.

Preventive maintenance procedures and various documents, including the vendors manual and completed surveillance tests were reviewed to ensure adequate surveillance testing and periodic battery replacements were in place to ensure reliable operation of the eight-hour emergency lights and that the emergency lighting units (ELUs) were being maintained consistent with the manufacturer's recommendations and accepted industry practices.

The team also observed whether emergency exit lighting was provided for personnel evacuation pathways to the outside exits as identified in the NFPA 101, Life Safety Code, and the Occupational Safety and Health Administration (OSHA) Part 1910, Occupational Safety and Health Standards. This review also included examination of whether backup ELUs were provided for the primary and secondary fire emergency equipment storage locker locations and dress-out areas in support of fire brigade operations should power fail during a fire emergency.

b. Findings

No findings of significance were identified.

.10 Cold Shutdown Repairs

a. Inspection Scope

The team verified that the licensee had evaluated the need for any dedicated repair procedures, equipment, and materials to accomplish repairs of components required for cold shutdown which might be damaged by the fire to ensure cold shutdown could be achieved within the time frames specific in their design and licensing bases.

b. Findings

No findings of significance were identified.

.11 Compensatory Measures

a. Inspection Scope

The team reviewed the administrative controls for out-of-service (OSS), degraded, and/or inoperable, fire protection features (e.g., detection and suppression systems and equipment, passive fire barriers, or pumps, valves or electrical devices providing SSD functions or capabilities). The team reviewed selected items on the fire protection impairment (FPIP) report and compared them with the fire areas/zones selected for inspection. The compensatory measures that had been established in these areas/zones were compared to those specified for the applicable fire protection feature to verify that the risk associated with removing the fire protection feature from service was properly assessed and adequate compensatory measures were implemented in accordance with the approved FPP. Additionally, the team reviewed the licensee's short term compensatory measures (compensatory fire watches) to verify that they were adequate to compensate for a degraded function or feature until appropriate corrective action could be taken and that the licensee was effective in returning the equipment to service in a reasonable period of time.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed Corrective Action Program (CAP) documents, including completed corrective actions documented in selected PERs to verify that fire protection problems potentially or actually affecting Browns Ferry Nuclear Plant were appropriately entered into, and resolved by, the corrective action program process. In addition, the inspectors reviewed Self-Assessment Report # BFN-ENG-06-012 which was a review of the plant's readiness for the NRC Triennial Fire Protection Inspection. The inspectors evaluated the effectiveness of the corrective actions for the identified issues. The documents reviewed are listed in the Attachment.

CAP problem evaluation reports related to the Browns Ferry FPP, and the capability to successfully achieve and maintain the plant in a SSD condition following a plant fire, as well as selected fire brigade response, emergency / incidents, and fire safety inspection reports were reviewed. This review was conducted to assess the frequency of fire incidents and effectiveness of the fire prevention program and any maintenance-related or material condition problems related to fire incidents.

The team also reviewed other CAP documents, including completed corrective actions documented in selected PERs, and operating experience program (OEP) documents to verify that industry-identified fire protection problems potentially or actually affecting Browns Ferry were appropriately entered into, and resolved by, the corrective action program process. Items included in the OEP effectiveness review were NRC Regulatory Issue Summaries (RIS), Information Notices (INs), industry or vendor-generated reports of defects and noncompliance under 10 CFR Part 21, and vendor information letters. In addition, the inspectors reviewed a sample of the fire protection program audits and self-assessments which the licensee performed in the previous one-year period to assess the types of findings that were generated and that the findings were appropriately entered into the licensee's CAP.

The inspectors evaluated the effectiveness of the corrective actions for a sample of identified issues. The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On September 15, 2006, the lead inspector presented the inspection results to Mr. R.G. Jones and other members of his staff. Proprietary information is not included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:
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NRC Personnel:
R. Fanner, RII Reactor Inspector
T. Harrison, RII Reactor Inspector (Training)
R. Holbrook, RII Resident Inspector Contractor
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D. Mas-Penaranda, RII Reactor Inspector
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D. Payne, RII Chief, Engineering Branch 2, Division of Reactor Safety
R. Rodriguez, RII Reactor Inspector
T. Ross, RII Senior Resident Inspector
G. Wiseman, RII Senior Reactor Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000260/2006014-001	URI	Postulated Fire-Induced Circuit Failures Could Prevent the Operator From Opening LPCI Injection Valve 2-FCV-074-053 From The Main Control and Result in Failure to Establish LPCI Flow Into the Reactor Vessel. (Section 1R05.01.b.1)
05000260,296/2006014-003	URI	Unapproved Local Manual Operator Actions in Lieu of Cable Protection for a Fire Area Subject to the Requirements of Appendix R Section III.G.2.(Section 1R05.01.b.3)

Opened and Closed

05000260/2006014-002	NCV	Safe Shutdown Instruction Directs Operators to Connect 4KV SD BD "A" to a Potentially Fire-Induced Fault in FA 8. (Section 1R05.01.b.2)
05000260,296/2006014-004	NCV	Cellular Phone Communications Unreliable for Alternate Shutdown in FA16. (Section1R05.08)

Closed
None.

Discussed
None.

Section 1R05.03 List of Fire Barrier Features inspected in Relation to Safe Shutdown Separation Requirements

Fire Protection Feature	Description
Fire Doors	Nos. 464, 500, 541, 640
Fire Dampers	Nos. 31-2018, 31-2019, 31-2009, 31-2515, 31-2527, 31-2532
Fire Barrier Penetration Seals	Nos. S25933885, S25933886, S25933887, R26045868, R26045606

Section 1R05.01 List of SSI Procedures / Attachments / Sections Reviewed

Procedure/Attachment	Sections
2/3-SSI-8/1	Section 1
2/3-SSI-8/2	Section 1
2/3-SSI-8/3	Section 1-2
2/3-SSI-2-5/1	Section 1-2
2/3-SSI-2-5/2	Section 1-4

2/3-SSI-2-5/3	Section 1-3
2/3-SSI-16/1	Section 1-2
2/3-SSI-16/2	Section 1
2/3-SSI-16/3	Section 1-3
2/3-SSI-16/4	Section 1-4
2/3-SSI-16/5	Section 1

Section 1R05.07.a: Components Reviewed for Cable Failure Modes

Component No.	Description
2-FCV-73-2	HPCI Steam Supply Line Inboard Containment Isolation Valve (CIV)
2-FCV-73-3	HPCI Steam Supply Line Outboard CIV
2-FCV-73-16	Steam Admission Valve To HPCI Turbine
2-FCV-74-60	Drywell Spray Line Outboard CIV
2-FCV-74-61	Drywell Spray Line Inboard CIV
2-FCV-74-57	Shutoff Valve In Return Line To Suppression Pool
2-FCV-74-59	Suppression Pool Cooling Line CIV
2-FCV-74-58	Wetwell Spray Line CIV
0-PMP-23-85	RHRSW Pump A3
0-PMP-23-0012	RHRSW Pump C2
0-PMP-23-0091	RHRSW Pump C3
2-PMP-74-0016	RHR Pump 2C
2-LT-003-0058A	Reactor Water Level Transmitter
2-LT-003-0058B	Reactor Water Level Transmitter
2-LT-003-0058C	Reactor Water Level Transmitter
2-LT-003-0058D	Reactor Water Level Transmitter
2-LI-003-0058B	Reactor Water Level Indicator
2-LI-064-0159A	Suppression Pool Level Indicator
2-TI-064-0162	Suppression Pool Temperature Indicator
2-PI-003-0074B	Reactor Vessel Pressure Indicator
2-PI-064-0160A	Drywell Pressure Indicator
2-PCV-001-0005	Group C Main Steam Relief Valve (MSRV)
2-PCV-001-0018	Group B MSRV
2-PCV-001-0030	Group A MSRV
2-FCV-067-0051	RBCCW Heat Exchanger Supply Line Isolation Valve
2-FCV-067-0050	RBCCW Heat Exchanger Supply Line Isolation Valve
2-FCV-074-0053	LPCI Injection Outboard CIV
2-FCV-074-0067	LPCI Injection Outboard CIV
2-TS-073-0002A	Steam Leakage Detection Temperature Switch
2-TS-073-0002B	Steam Leakage Detection Temperature Switch
2-TS-073-0002C	Steam Leakage Detection Temperature Switch
2-TS-073-0002D	Steam Leakage Detection Temperature Switch
2-PS-073-0020A	Turbine Exhaust Over Pressure Relief Line Pressure Switch
2-PS-073-0020B	Turbine Exhaust Over Pressure Relief Line Pressure Switch
2-PS-073-0020C	Turbine Exhaust Over Pressure Relief Line Pressure Switch
2-PS-073-0020D	Turbine Exhaust Over Pressure Relief Line Pressure Switch

2-PS-073-0001A	Steam Supply Line Pressure Switch
2-PS-073-0001B	Steam Supply Line Pressure Switch
2-PS-073-0001C	Steam Supply Line Pressure Switch
2-PS-073-0001D	Steam Supply Line Pressure Switch
2-LT-003-0208B	Reactor Vessel Water Level Transmitter
2-LT-003-0208D	Reactor Vessel Water Level Transmitter
2-PS-073-0022A	Turbine Exhaust Line Pressure Switch
2-PS-073-0022B	Turbine Exhaust Line Pressure Switch
2-PDT-73-1A	Steam Line Flow Differential Pressure Transmitter
2-PDT-73-1B	Steam Line Flow Differential Pressure Transmitter
2-PT-73-29-1	HPCI Pump Suction Pressure Transmitter

LIST OF DOCUMENTS REVIEWED

Applicable Design Criteria, Codes and Standards

BFN-50-7026, High Pressure Fire Protection System, Rev. 6
BFN-50-747, Fire Protection of Safe Shutdown, Rev. 5
NFPA 13, Standard for the Installation of Sprinkler Systems, 1976 Edition
NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 1974 Edition
NFPA 20, Standard for the Installation of Centrifugal Fire Pumps, 1987 Edition
NFPA 72E, Standard on Automatic Fire Detectors, 1990 Edition
NFPA 80, Standard on Fire Doors and Windows, 1975 Edition
NUREG-1552, Supplement 1, Fire Barrier Penetration Seals in Nuclear Power Plants, dated January 1999
OSHA Standard 29 CFR 1910, Occupational Safety and Health Standards, Underwriters Laboratory, Fire Resistance Directory, January 1998

Licensing Basis Documents

BTP Chemical and Material Engineering Branch CMEB 9.5-1 Letter, dated July 1981
Fire Protection Report Volume 1, Fire Protection Plan, Rev. 36
Fire Protection Report Volume 1, Section 2, Fire Hazards Analysis, Rev. 23
Fire Protection Report Volume 2, Section I-D, Smoking Restrictions, Rev. 0
Fire Protection Report Volume 2, Section III, Fire Brigade Training and Fire Drill Evaluations/ Critiques, Rev. 2
NRC Safety Evaluation Report dated April 14, 1989
NRC Safety Evaluation Report dated March 31, 1993
NRC Safety Evaluation Report dated November 2, 1995
TVA Letter to NRC, Summary of Deviations from NFPA Code for BFN, dated August 3, 1988

Procedures

2/3-SSI-001, Safe Shutdown Instructions, Rev. 10
2/3-SSI-8, Unit 2, 4kV Electric Board Room 2B, Rev.8
2/3-SSI-8, Unit 2, Reactor Building Fire El. 621 and 639 North of Column Line R, Rev. 11
2/3-SSI-16, Control Building Fire El. 593 through El. 617, Rev. 13
2-SR-3.3.3.2.1 (74), Backup Control Panel Testing, Rev. 14
2-SR-3.3.3.2.1 (73), Backup Control Panel Testing, Rev. 2

2-SR-3.3.3.2.1 (71), Backup Control Panel Testing, Rev. 9
2-SR-3.3.3.2.1 (1 MSIV), Backup Control Panel Testing Main Steam Isolation Valves, Rev. 3
2-SR-3.3.3.2.1 (1 MSRV), Backup Control Panel Testing Main Steam Relief Valves, Rev. 2
Emergency Plan Implementation Procedure EPIP-3, Alert , Rev. 30
Emergency Plan Implementing Procedure EPIP-17, Fire Emergency Procedure, Rev. 29
Fire Protection Training Procedure TRN-31, Fire Brigade Training, Rev. 7
General Operating Instruction, 0-GOI-200-1, Freeze Protection Inspection, Rev. 57
Operating Instruction 0-OI-26, High Pressure Fire Protection System, Rev. 81
Standard Department Procedure FPDP-1, Conduct of Fire Protection, Rev. 0
Standard Department Procedure FPDP-2, Administration of Pre-Fire Plans, Rev. 0
Standard Department Procedure FPDP-4, Fire Emergency Response, Rev. 0
Standard Programs and Processes, SPP-10.9, Control of Fire Protection Impairments, Rev. 2
Standard Programs and Processes, SPP-10.10, Control of Transient Combustibles, Rev. 4
Standard Programs and Processes, SPP-10.11, Control of Ignition Sources (Hot Work), Rev. 4
Alarm Response Procedure 2-ARP-9-8B, Rev. 10
Alarm Response Procedure 1/2-ARP-9-23A, Rev. 13

Completed Surveillance Test Procedures and Test Records

Completed Surveillance Test (4-11-2005) 2-SR-3.3.3.2.1 (1 MSRV) Backup Control Panel Testing Main Steam Relief Valves, Rev.2
Completed Surveillance Test (4-12-2005) 2-SR-3.3.3.2.1 (1 MSIV) Backup Control Panel Testing Main Steam Isolation Valves, Rev. 2
Completed Surveillance Test (4-11-2005) 2-SR-3.3.3.2.1 (74) Backup Control Panel Testing, Rev. 13
FP-0-247-INS004, Appendix R Battery Operated Emergency Lighting Quarterly Test, Rev. 22, completed 06/01/06, 05/31/06, 08/29/06
FP-2-247-INS003B, Second Period Appendix R Battery Operated Emergency Lighting 18 Month Test, Rev. 14, completed 04/24/06
0-SI-4.11.A.1(1), Control Bay El. 593' Local Control Panel Test, Rev. 11, completed 05/11/2006
0-SI-4.11.A.2, Fire System Circuit Operability Test, Rev. 16, completed 04/05/2006
0-SI-4.11.B.1.b, High Pressure Fire Protection System Valve Position Verification (Inside Loop), Rev. 43, completed 08/03/2006
0-SI-4.11.B.1.c, High Pressure Fire Protection System Flushes, Rev. 33, completed 04/17/2006
0-SI-4.11.B.1.e, High Pressure Fire Protection System Valve Cycling, Rev. 18, completed 09/30/2005
0-SI-4.11.B.3.a, Weekly Check for Diesel Fire Pump Batteries 1 & 2, Rev. 18, completed 01/04/2006, 01/11/2006, 01/18/2006, 01/25/2006
0-SI-4.11.E.1.a, Fire Hose Station Inspection and Reracking, Rev. 13, completed 09/17/2004
0-SI-4.11.E.1.b(1), Fire Hose Station Operability / Flow Test, Rev. 3, completed 08/26/2004
2-SI-4.11.A.1.(3), Unit 2 Reactor Building Local Fire Control Panel Operability Test, Rev. 10, completed 12/27/2005
2-SI-4.11.C.1.c, Simulated Automatic Actuation of the Fire Protection Sprinkler Systems, Rev. 29, completed 3/24/2006
3-SI-4.11.C.1.c, Simulated Automatic Actuation of the Fire Protection Sprinkler Systems, Rev. 23, completed 09/26/2005

Calculations, Evaluations, and Specifications

EDQ0-999-2003-0055, Rev. 3, BFN Appendix R Auxiliary Power System Loading Table
ND-Q0999-920116, Rev. 17 (?19?), Appendix R Manual Action Requirements
ED-Q0999-940040, Rev. 8 (?17?), Appendix R Computerized Safe Shutdown Separation Analysis
ED-Q0999-880489, Appendix R Tabulation of Equipment Power Supplies and Detailing Criteria for the Auxiliary Power system, Rev. 27
Engineering Planning and Management, Inc. Fire Area/zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Rev. 2, Fire Area 2-5
Engineering Planning and Management, Inc. Fire Area/zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Rev. 2, Fire Area 8
Engineering Planning and Management, Inc. Fire Area/zone Detailed Appendix R Safe Shutdown Separation Analysis Record, Rev. 2, Fire Area 16
TVA General Design Criteria, BFN-50-747, Fire Protection of Safe Shutdown Systems, Rev. 5
Fire Protection Report Volume 1, Fire Protection Plan, Rev. 36
Fire Hazard Analysis Vol. 1, Rev. 36
BFNP Appendix R Safe Shutdown Program, Vol. 1, Rev. 36
EDN0244890050, Appendix R Analysis for Intraplant Communication System, Rev. 5
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110479, Loss of U2/U3 Fire Pump Start
40532, NRC Letter Secy 03-100 Was Recently Issued On Rulemaking For Manual Actions Used For 10 CFR 50 Appendix R III.G.2 Compliance
53289, SSIs Have Not Described What Tool Is Used To Open Tamper Proof Covers
77358, Nuclear Assurance Audit
73268, Communication System Discrepancies
95151, The Fire Protection Report Should State Compensatory Actions
102638, Smoke Coming from Clean Lube Oil Tank, 05/08/2006
105831, Smoke Coming from Panel 25-18 Power Supply, 06/25/2006
104090, Smoke Appeared from Temporary Test Counter, 05/30/2006

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110657, Nextel Cell Phone Coverage, 9/13/06
110612, 2-FCV-74-53 Control Cables (2ES769-1 and 2ES1050-1) Are Routed In Fire Area 8 Such That Operation Of This Valve From Control Room Cannot Be Ensured, 9/13/06
110536, Procedure 2/3-SSI-8 Aligns 4KV SHDN Bd A To Its Alternate Feed (4KV SHDN Bus 2) Which Does Not Ensure 4KV SHDN Bd A Availability, 9/12/06
110617, Emergency Light Head Misdirected (App. R light #404), 9/13/2006
108103, Unacceptable Housekeeping in the Unit 2 Board Room and Unit 3 Computer Room 08/03/06
110658, SSI Walkdowns During the NRC TFP noted that SSI Validation Does Not Appear To Consider All Parameters in NRC IP 71111.05T, 9/13/06
109515, The 4160 V Board is Incorrectly Labeled on Pre-Fire Plan No. RX2-593, Rev.4, 8/24/06
109661, Appendix R Spurious Valve Opening (0-FCV_25-32 & 70), 9/1/06

LIST OF ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
AI	Action Item
APCSB	Auxiliary and Power Conversion Systems Branch
ATWS	Anticipated Transient Without Scram
BFN	Browns Ferry Nuclear
BTP	Branch Technical Position
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CMEB	Chemical and Material Engineering Branch
CPSC	Consumers Product Safety Commission
DC	Direct Current
ELU	Emergency Lighting Unit
EPRI	Electric Power Research Institute
FA	Fire Area
FHA	Fire Hazards Analysis
FPIP	Fire Protection Impairment
FP	Fire Protection
FPP	Fire Protection Program
FPR	Fire Protection Report
FZ	Fire Zone
GL	Generic Letter
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, and Air Conditioning
INs	Information Notices
IP	Inspection Procedure
JPM	Job Performance Measures
MCC	Motor Control Center
MCR	Main Control Room
MOA	Manual Operator Action
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NRC	U. S. Nuclear Regulatory Commission
OEP	Operating Experience Program
OOS	Out Of Service
OSHA	Occupational Safety and Health Administration
PER	Problem Evaluation Report
P&ID	Piping and Instrumentation Drawings
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIS	Regulatory Issue Summaries
RPV	Reactor Pressure Vessel
RSDP	Remote Shutdown Panel
SCBA	Self-Contained Breathing Apparatus

SER	Safety Evaluation Report
SR	Surveillance Requirement
SSA	Safe Shutdown Analysis
SSC	Structures, Systems, and Components
SSD	Safe Shutdown
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volt