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

November 13, 2006

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INSPECTION REPORT 05000259/2006016

Dear Mr. Singer:

On September 29, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection associated with recovery activities at your Browns Ferry Unit 1 reactor facility. The enclosed inspection report documents the inspection results, which were discussed on September 29, 2006, with Mr. B. O'Grady, Mr. M. Bajestani and other members of your staff.

This inspection examined activities conducted under your Unit 1 license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license and also with fulfillment of Unit 1 Regulatory Framework Commitments. The inspection focused on the Special Program for Fire Protection. The inspectors reviewed selected procedures and records, observed activities, walked-down facilities, and interviewed personnel.

Based on the results of this inspection, no violations or findings of significance were identified. However, the report includes one unresolved item related to the incomplete evaluation of an approved 10 CFR 50, Appendix R, Section III.G.2.b exemption concerning intervening combustibles in all three reactor building separation zones. The Special Program for Fire Protection remains open pending final NRC review of fire protection modifications on Unit 1 and review of the approved three-unit safe shutdown instructions scheduled for January 2007.

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 No. 50-259
License No. DPR-33

Enclosure: Inspection Report 05000259/2006016
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2006016
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant (BFNP), Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	September 25 - 29, 2006
Inspectors:	G. MacDonald, Senior Reactor Inspector - Team Leader R. Fanner, Reactor Inspector N. Merriweather, Senior Reactor Inspector R. Rodriguez, Reactor Inspector G. Wiseman, Senior Reactor Inspector T. Harrison, Reactor Safety Nuclear Specialist
Approved by:	D. Charles Payne, Chief Engineering Branch 2 Division of Reactor Safety

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2006016

This inspection included aspects of licensee engineering activities associated with the Unit 1 Recovery Special Program for Fire Protection. The inspection program for the Unit 1 Restart Program is described in NRC Inspection Manual Chapter 2509. Information regarding the Browns Ferry Unit 1 Recovery and NRC Inspections can be found at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/bf1-recovery.html.

Inspection Results - Engineering

- The licensee's modification activities to date, associated with four ongoing fire protection permanent plant modifications, were performed in accordance with documented requirements. The modification design packages associated with fire protection systems and features met applicable code requirements and licensing commitments. These fire protection modifications adequately addressed the changes needed to restore Unit 1 to current fire protection plan (FPP) requirements. No violations were identified during this ongoing review of the licensee's FPP for the restart of Browns Ferry Unit 1. (Section E1.1.)

- The inspectors identified a fire risk vulnerability in the Unit 1 Reactor Building. The vulnerability involved the fire-induced maloperation (failure to close or spuriously open subsequent to closure) of the raw service water head tank isolation valves upon an automatic fire pump start signal. The licensee indicated that procedural changes to the appropriate reactor building pre-fire plans would be incorporated to require closure of a manual valve in order to resolve the problem and prevent the loss of fire water suppression flow and pressure. No violations or deviations were identified. (Section E1.2.)

- For the sample of safe shutdown (SSD) components examined, the inspectors determined that the licensee's SSD analysis had properly assessed the affects of fire damage from associated cables. The licensee had identified feasible operator manual actions (OMAs) or work arounds for the cable failures postulated in the current draft three-unit safe shutdown instructions (SSIs). The work arounds would require the operator to locally start and stop some equipment from areas not affected by the postulated fire. A final assessment of OMA feasibility will be established based on review of the approved three-unit SSIs during an inspection scheduled for January 2007. (Section E1.3)

- The inspectors determined that the draft three-unit SSI sample identified adequate equipment and guidance to achieve and maintain hot and cold shutdown. However the procedures, which incorporated local OMAs to accomplish SSD in Appendix R Section III.G.2 areas, were still in a draft status. Several discrepancies were noted in the draft three-unit procedures in the sample reviewed. The licensee had not yet completed modifications for Unit 1 restart and the fire protection design calculations and SSIs were still undergoing changes. Until the fire protection related modifications are complete and the three-unit SSIs approved, the NRC cannot make a final determination on the feasibility and reliability of the OMAs and the SSIs used to implement SSD for BFNP. The inspection to determine acceptability of the approved three-unit SSIs to implement SSD and fire protection related modification completion is scheduled for January 2007. (Section E1.4)

Inspection Results - Plant Support

- NRC approved an exemption from 10 CFR 50, Appendix R, Sections III.G.2.b concerning intervening combustibles in the form of cables in trays in all three reactor building 20-foot separation zones in 1988. The TVA exemption request did not address some in-situ fire hazards such as a portion of a 480V motor control center (MCC) that extended into the separation zones. The inspectors also determined that the exemption does not currently consider combustibles in ongoing Unit 1 modifications. Implementation of Unit 1 Design Change Notice (DCN) 61563 introduced additional intervening combustibles (Thermo-Lag) other than cable insulation in trays within the 20-foot separation zones located on the 565' and 593' elevations of the Unit 1 Reactor Building which need to be included in the exemption. On October 26, 2006, TVA submitted to the NRC a revision to the existing exemption from 10 CFR 50, Appendix R, Section III.G.2.b originally approved by NRC in 1988. This item is unresolved pending further NRC review of the licensee’s October 26, 2006, revision to the exemption concerning intervening combustibles in the form of cables in trays in the 20-foot separation zones and to complete fire modeling and risk evaluation to determine the credibility of the potential fire scenarios. The licensee initiated problem evaluation reports 93306 and 111744 for tracking and resolution of this issue. (Section F1.1)

REPORT DETAILS

II. Engineering

E1 Conduct of Engineering

E1.1 Unit 1 Restart Special Program - Fire Protection - Appendix R (37550)

a. Inspection Scope

The inspectors reviewed the scope of the licensee's corrective actions that were developed for resolution of issues associated with the Browns Ferry fire protection plan (FPP) and to meet the requirements of Appendix R. The corrective actions were evaluated for conformance with the objectives of the BFN Fire Protection Improvement Plan which was used for resolving fire protection concerns prior to the re-start of Units 2 and 3. See Inspection Report 05000259/2004-009 for documentation of previous inspections in this area. During the current inspection, the inspectors reviewed planned Design Change Notice (DCN) packages associated with modifications to passive fire protection features to meet Appendix R separation requirements in the Unit 1 Reactor Building. These included modifications for addition of fire dampers to the Unit 1 Reactor Building Ventilation System, replacement of several fire doors, the addition of fire barrier penetration seals, and installation of a one-hour fire rated Electrical Raceway Fire Barrier System (ERFBS) on portions of cable raceways located in the Unit 1 Reactor Building. In addition, the inspectors reviewed a DCN package associated with a modification to active fire protection features that consisted of enhancing the Unit 1 Reactor Building pre-action sprinkler system.

b. Observations and Findings

b.1 DCN Package Review

The inspectors reviewed the scope of corrective actions developed for resolving Fire Protection Program and Appendix R issues and determined the Unit 1 Fire Protection Improvement Plan was accomplished by the following four design change notices:

- DCN 51190 - Ventilation system modifications
- DCN 51208 - Penetration seal modifications (includes fire doors)
- DCN 61563 - Fire Wrapped Identification Raceways to Meet Appendix R Requirements
- DCN 51180 - Suppression system modifications

This review was conducted to verify that the design change packages contained adequate design information and supporting analyses to allow plant personnel to properly implement the desired change, update plant documentation, and resolve the identified condition.

The inspectors evaluated the adequacy of the modification packages and observed a selected sample of completed field work to verify that modifications would not adversely affect the design basis of the system or interfacing systems.

DCN 51190

The inspectors reviewed and observed activities associated with a selected portion of permanent plant modification DCN 51190, Ventilation System Modifications. This DCN implemented fire damper modifications for the ventilation systems in the reactor building. Planned changes included the installation of 27 fire damper/sleeve assemblies in floor supply and exhaust duct penetrations at designated fire zone boundaries within the building. The DCN also required the performance of ventilation air flow tests to confirm that the new damper assemblies did not adversely affect design basis ventilation system air flow requirements. This DCN addresses problem evaluation report (PER) 03-003181-000 concerns that the ventilation duct penetrations did not contain fire dampers to prevent the potential spread of fire and hot gases between fire zones. In these installations, the fire damper is installed external to the floor/ceiling fire barrier within a heavy gauge metal enclosure mounted on a curb above the vertical duct penetration. The fire dampers are curtain-type spring-operated Underwriters Laboratory (UL) listed fire dampers actuated to close against air flow by non-electrical thermal sensitive fusible links. Part of DCN 51190 included revising 1-47E865- Drawing series, "Flow Diagram - Heating and Ventilating Air Flow, System 64 - Powerhouse, Reactor Building Unit 1" to include the new fire dampers. The inspectors performed walkdowns of the accessible fire dampers in penetration numbers F-215, F-303, F-216, F-250, F-255, and F-263 to confirm that the damper/sleeve field assemblies were mounted consistent with manufacturers design drawings, and UL 555 listing requirements; and met National Fire Protection Association (NFPA) code requirements and licensing commitments. The inspectors also reviewed the fire damper/sleeve installation drawings that correlated to each penetration number, as well as reviewing construction details, engineering evaluations and fire endurance tests for the damper sleeve assemblies, to verify that the as-built configurations were either properly evaluated or qualified by appropriate fire endurance tests.

DCN 51208

The inspectors reviewed and observed portions of permanent plant modification activities associated with modification DCN 51208, Penetration Seals Modification (including Fire Doors). This DCN implemented the fire door modifications recommended for Penetrations and Sleeves System, in the Unit 1 Reactor Building. The DCN consisted of two stages and included updating fire penetration seal information and fire door replacement to assure Appendix R fire barrier integrity between fire areas/zones. Stage 1 involved inspection, repair, and/or replacement of all penetration seals between Unit 1 fire areas/zones to ensure penetration seals are fire rated and approved for use and supported by calculation MD-Q0100-980006, Engineering Evaluation of Penetration Seals. During plant walkdowns the inspectors observed the material condition and configuration of several installed penetration seals identified on penetration seal location drawings for the selected fire areas/zones boundaries. The inspectors also reviewed associated construction detail drawings, engineering evaluations and fire endurance tests for a sample of the installed seals, to verify that the as-built configurations met design requirements, license commitments, and standard industry practices. Stage 2 of DCN 51208 instructs that doors, 490, 635, and 670, located between reactor building elevations and the stairwell/elevator lobby be replaced with UL approved one-hour rated fire doors to provide the required fire rated barrier separation between fire zones. The existing doors were damaged and were not labeled with fire-rated certification. Each fire door consisted of two leaves, associated hardware, and a frame. At the time of this inspection the replacement doors were not yet installed, therefore, the inspectors observed the doors, hardware, and frames, currently stored in a TVA warehouse. The inspectors verified that the required UL listing labels were attached to the door leaves and frames indicating they were one-hour fire-rated doors to satisfy the applicable NFPA 80 requirements.

<u>DCN 61563</u>

The inspectors reviewed Unit 1 permanent plant modification DCN 61563, Unit 1 Reactor Building Fire Wrap Raceways to Meet Appendix R Requirements (systems 023, 211, 244). The DCN was developed to address concerns identified in the Unit 1 baseline Fire Safe Shutdown Analysis (ED-Q0999-2003-0037). This calculation identified cables located in the Unit 1 Reactor Building which required protection with a fire rated barrier material to satisfy 10 CFR 50, Appendix R separation requirements for emergency safe shutdown cables. This DCN provided a one-hour fire rated ERFBS for Appendix R safe shutdown (SSD) cables ES88-I, ES125-I, ES2673-II, ES2625-II, PP459-IA and PP460-IA in the Unit 1 Reactor Building prior to restart of Unit 1. The DCN also identified that some of these raceways pass through or within the designated 20 ft. separation zones on the 565' elevation (between fire zones 1-1 and 1-2) and on the 593' elevation (between fire zones 1-3 and 1-4). This review consisted of evaluation of test observations and methodology given in a calculation of the fire endurance and qualification of fire-rated material used for cable protection.

<u>DCN 51180</u>

The inspectors reviewed permanent plant modification DCN 51180, Suppression System Modifications. This DCN upgraded the Unit 1 Reactor Building pre-action sprinkler system to achieve general area-wide floor coverage consistent with applicable NFPA requirements, including providing air supervision (supplied by the service air system) of the dry sprinkler piping. In addition, the DCN added pre-action sprinkler water curtains beneath unsealed vertical stairway openings, residual heat removal (RHR) system door and room openings, and open hatches between fire zones at several elevations of the Unit 1 Reactor Building. The inspectors performed walkdowns of the accessible portions of the pre-action sprinkler system on the 565' and 593' elevations of the Unit 1 Reactor Building, including stairway and hatch water curtains to evaluate proper type, placement, spacing of the sprinkler heads, and the extent of the sprinkler head obstructions for effectiveness to prevent a fire from spreading to adjacent fire zones.

In addition, the inspectors examined the sprinkler system hydraulic design calculations to verify that the system could supply sufficient pressure and flow volume to produce the required water density for the protected area.

c. Conclusions

The inspectors determined that the licensee's modification activities to date, associated with four ongoing fire protection permanent plant modifications, were performed in accordance with the documented requirements. The modification design packages associated with fire protection systems and features met applicable code requirements and licensing commitments. For the sample of modification packages examined, the inspectors determined that the modifications adequately addressed the changes needed to restore Unit 1 to current FPP requirements. No violations were identified during this ongoing review of the licensee's FPP for the restart of Browns Ferry Unit 1.

E1.2 Review of Circuit Analysis Data for the Fire Water Pumps and Distribution System to Verify that the System Will Be Free of Fire Damage and Available in Selected Unit 1 Fire Areas (37550)

a. Inspection Scope

The inspectors reviewed the Fire Hazards Analysis (FHA), Safe Shutdown Analysis (SSA) calculations, the FPP, mechanical fire protection flow diagrams, piping and instrumentation drawings (P&IDs), cable routing information, fire brigade pre-fire plans, fire protection system valve lineup procedures, electrical drawings, and other supporting documents associated with the fire pumps and common fire protection water delivery system. The purpose of the review was to assess if 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. Observations and Findings

The BFN FPP, Section 4.4.1, states that pressure is maintained on the high-pressure raw water fire protection system (System 26) through an interconnection to the raw service water (System 25) at approximately 50 psig. The raw service water system pumps provide enough pressure and quantity of water to fight small fires throughout the powerhouse below Elevation 617'. The fire pumps are automatically actuated upon a fire signal from any of the fixed water fire fighting systems. When the fire pumps are actuated the raw service water head tanks are isolated and the raw service water pumps are shut off.

10 CFR 50, Appendix R, Sections III.G.2 does not require that fire protection systems be treated as safe shutdown systems, however, the licensee's Appendix R Computer Separation Analysis evaluated fire protection pump availability by postulating an Appendix R fire in any area of the plant. Each fire area was evaluated for potential loss of fire pumps due to fire in that area. Inspectors' review of the fire protection P&IDs found that the raw service water head tanks are isolated upon a fire pump start by a

closure signal from contacts to air operated valve (AOV) FCV-25-32 (0-FCV-25-0032) and motor operated valve (MOV) FCV-25-70 (0-FCV-25-0070). The inspectors determined that the raw service water head tank isolation functions (equipment, components, and cables) were not included in the fire pump availability study. Additionally, a review of cable routing information for the isolation valves found that the control logic and power cables to these two valves were routed in common cable trays KAH, LY, and MM and are susceptible to fire damage within the Unit 1 Reactor Building. The current configuration is of concern because fire-induced cable failures or hot shorts can prevent these two valves from closing, or spuriously open these two valves once closed. This failure will result in fire protection water flow diversion through the raw service water head tank three-inch overflow and vent lines. This event could possibly reduce the effectiveness of fire protection water flow and pressure for fixed and fire brigade fire fighting systems. The inspectors noted that were the isolation valves to remain open, the system was equipped with a manual valve that could be closed to prevent the loss of fire water suppression flow and pressure.

The inspectors determined that this issue was not a violation of NRC requirements, however the licensee initiated PER 109961 to evaluate the need to isolate the fire protection water flow diversion should the automatic valves fail open. The licensee indicated that procedural changes to the appropriate reactor building pre-fire plans would be incorporated to require closure of the manual valve. This action was documented into their corrective action program.

c. Conclusions

No violations or deviations were identified. However, the inspectors identified a fire risk vulnerability in the Unit 1 Reactor Building. The vulnerability involved the fire-induced maloperation (failure to close or spuriously open subsequent to closure) of the raw service water head tank isolation valves upon an automatic fire pump start signal. The licensee indicated that procedural changes to the appropriate reactor building pre-fire plans would be incorporated to require closure of a manual valve in order to resolve the problem and prevent the loss of fire water suppression flow and pressure.

E1.3 Review of Circuit Analysis Data For A Sample of SSD Components to Verify That One Train of SSD Components Will Be Available and Free of Fire Damage in the Selected Fire Areas of Interest (37550)

a. Inspection Scope

The inspectors reviewed the results of the licensee's circuit analysis for a sample of SSD components to verify that the components would be available to perform their hot and/or cold shutdown functions during a fire in the selected fire areas. The inspectors also reviewed the feasibility of operator manual actions that would be implemented to mitigate potential spurious operation of SSD components due to postulated fire damage to cables in the selected areas. Documents reviewed are listed in the Attachment.

b. Observations and Findings

The inspectors reviewed various design output documents including cable routing data sheets, electrical schematics, connection diagrams, and fuse coordination studies to verify that the SSD components selected would be available to perform their hot and/or cold shutdown functions during a fire in the selected fire areas. The effects of hot shorts, shorts to grounds, or open circuits were evaluated for the selected SSD components that had unprotected cables routed through the fire areas. In cases where double fusing was the method used to protect the SSD circuits from fire damage, the inspectors reviewed the fuse coordination results and associated fuse data sheets to verify that the upstream and downstream fuses were properly sized and selected to protect the SSD circuit from fire damage. For the sample of components examined, the inspectors verified that the licensee's SSD analysis had properly assessed the effects of fire damage from associated cables and had identified feasible operator manual actions (OMAs) or work arounds for the cable failures postulated. The work arounds would require the operator to locally start and stop some equipment from the 480V Reactor MOV Board or 4KV Shutdown Board. For example, the residual heat removal service water (RHRSW) pump A1 was credited for a fire in fire area 1 fire zone 1-4 (FA 1/FZ1-4). A control cable associated with this pump was routed in FA 1/FZ 1-4 and could be damaged by a fire in this area. Therefore, a manual action is required to isolate the cable failure with transfer switch 0-XS-23-1 at 4KV Shutdown Board A located in FA 5. This FA would not be affected by a fire in FA 1-4. Two other components requiring local manual action due to potential fire damage to associated cables in FA 1/FZ 1-4 were the RHR Pump 1A and low pressure coolant injection (LPCI) Injection valve 1-FCV-74-53. RHR Pump 1A has two control cables routed in FA 1/FZ 1-4 that could be damaged by a fire. A manual action is required to isolate the failed cables by operating transfer switch 1-XS-74-5 located at the 4KV Shutdown Board in FA 5. As a result of fire damage to the RPV Low Pressure Permissive, the operator may not be able to open the LPCI Injection Valve 1-FCV-74-53 from the MCR. Failure of this permissive is due to cable failures that support the RHR Train A Logic. A modification is planned to provide a remote emergency control scheme at the Reactor MOV Board. This will require a local manual action to open valve 1-FCV-74-53 at the Reactor MOV Board in FA 4. The use of operator manual actions in lieu of cable protection is not allowed by 10 CFR 50, Appendix R, Section III.G.2. Section III.G.2 only allows for physical protection or separation of cables as an acceptable method for ensuring that one train is free of fire damage. FA 1/FZ 1-4 is an Appendix R Section III.G.2 fire area requiring the ability to achieve and maintain hot SSD from the control room during a fire. The above OMAs are considered to be additional examples of URI 05000259/2006012-001, Local Operator Actions in Lieu of Cable Protection for a Fire Area Subject to the Requirements of III.G.2.

c. Conclusions

For the sample of SSD components examined, the inspectors determined that the licensee's SSD analysis had properly assessed the affects of fire damage from associated cables. The licensee had identified feasible OMAs or work arounds for the cable failures postulated in the current draft three-unit SSIs. The work arounds would require the operator to locally start and stop some equipment from areas not affected by the postulated fire. A final assessment of OMA feasibility will be established based on review of the approved three-unit SSIs during an inspection scheduled for January 2007.

E1.4 Operational Implementation of SSD For FA 8, FA 16 and FA 1/FZ 1-4 (37550)

a. Inspection Scope

The inspectors reviewed selected sections of the three-unit Safe Shutdown Instructions to be used to achieve and maintain SSD following the transition to three unit operation at Browns Ferry. The scope included the Unit 1 SSD portions of the SSIs for FA 8 (U2 4KV Electric Board Room 2B), FA 16 (Control Building El 593 Through El617) and FA 1/FZ 1-4 (Unit 1 Reactor Building El 593 South of Column Line R and RHR Heat Exchanger Rooms from El 565 Through 593). Procedures 0-SSI-8 and 0-SSI-1-4 were reviewed to determine if the licensee had implemented a SSD strategy to achieve and maintain hot and cold shutdown from the main control room both with and without the availability of offsite power for a postulated fire in FA 8 or FA 1/FZ 1-4. Procedure 0-SSI-16 was reviewed to determine if the licensee had implemented a SSD strategy to achieve and maintain hot and cold shutdown both with and without the availability of offsite power utilizing alternative shutdown for a postulated fire in FA 16. The inspectors reviewed the reliability and feasibility of local OMAs utilized to achieve and maintain SSD for a postulated fire in FA 1/FZ 1-4 as described in procedure 0-SSI-1-4 using the criteria identified in NRC Inspection Procedure 71111.05T paragraph 11B. The inspectors reviewed the licensee validation walkdown packages for procedures 0-SSI-8, 0-SSI-1-4 and 0-SSI-16, and independently walked down selected SSI attachments and sections identified in the Attachment. Proposed licensee staffing for three-unit operation was reviewed and backup control panel surveillance testing was reviewed for selected alternative shutdown equipment.

b. Observations and Findings

The inspectors noted that the fire protection SSIs to be used for three-unit operation at BFNP were still in a draft non-approved status. The licensee had performed timed walkthroughs of the draft SSIs. The inspectors reviewed the walkdown packages for 0-SSI-1-4, 0-SSI-8 and 0-SSI-16 and noted that the time documented by the auxiliary operators met the procedure time requirements. The inspectors performed independent walkthroughs with licensee auxiliary operators of selected attachments and sections of draft SSIs 0-SSI-1-4, 0-SSI-8 and 0-SSI-16 and did not identify any instances where the required time in the procedure was not achieved. The licensee's walkdown packages did not document parameters other than time. Environmental and human factors considerations were not incorporated into the walkthrough validation and PER 110658 was initiated for this concern. During the NRC walkdowns, the inspectors observed several discrepancies with the draft three-unit SSIs. In SSI 0-SSI-16, the inspectors noted a labeling discrepancy for the switch to the normal feeder breaker for transformer TS2E. In SSI 0-SSI-1-4 the inspectors observed that the procedure lacked guidance to direct the operator to don a self contained breathing apparatus prior to entering the reactor building to perform a local trip of the recirculation pumps. This action would be performed in a potential smoke affected area. During the walkdown of procedure 0-SSI-1-4 for Unit 1 Reactor Building, a 10 CFR 50, Appendix R Section, III.G.2 fire area, the inspectors determined that the procedure contained local OMAs. These OMAs were reviewed and judged to be feasible and reliable per the criteria of NRC Inspection Procedure 71111.05T paragraph 11.B, however the procedure was draft. A final assessment of OMA feasibility will be established based on review of the approved three-unit SSIs during an inspection scheduled for January 2007. The inspectors reviewed the proposed staffing for three-unit operation and determined that sufficient staff would be assigned to meet the procedure manpower requirements (not counting fire brigade personnel).

c. Conclusions

The inspectors determined that the review of selected draft three-unit SSIs identified adequate equipment and guidance to achieve and maintain hot and cold shutdown conditions, however, the procedures had not been approved in final form. The inspectors identified that the SSIs incorporated local OMAs to accomplish SSD in Appendix R, Section III.G.2 areas. Several discrepancies were noted in the draft three-unit procedures in the sample reviewed. The licensee was still working on the modifications for Unit 1 restart and the fire protection design calculations, and SSIs, were still undergoing changes. Until the fire protection-related modifications are complete and the final SSIs approved, the NRC cannot make a final determination on the feasibility and reliability of the OMAs and the SSIs used to implement SSD for BFNP. The inspection to determine acceptability of the final approved SSIs to implement SSD and fire protection related modification completion is scheduled for January 2007.

Plant Support

F1 Fire Protection Program

F1.1 Review of the Fire Protection Report and Selected Appendix R Exemptions (64704)

a. Inspection Scope

The inspectors reviewed the licensee's methodology for meeting the requirements of 10 CFR 50.48 and the bases for the NRC's acceptance of this methodology as documented in NRC Safety Evaluation (SEs). The inspectors reviewed the Fire Protection Report (FPR) for Units 2 & 3 and the draft FPR for combined three-unit operation. In addition, the inspectors reviewed the SSA, FHA, NRC SERs, and license documentation, such as submittals made to the NRC by the licensee in support of the NRC's review of their FPP, and deviations from NRC regulations. Plant walkdowns were also performed to verify that the plant configuration was consistent with that described in SSA and FHA analyses to verify that the licensee met fire protection-related license commitments for the Unit 1 Reactor Building.

b. Observations and Findings

The licensee's SSA relies on the reactor building being separated into analysis zones, with 20-foot separation between redundant safe shutdown trains located in each of the zones. The Unit 1 Reactor Building, Fire Area 1, contains all four Unit 1 RHR LPCI injection valves. Division 1 RHR LPCI injection valves are located in Fire Zone 1-1 and the redundant Division 2 valves are in Fire Zone 1-2. The RHR system provides reactor makeup and decay heat removal functions under transient and accident conditions for a fire. Fire Zones 1-1 and 1-2, and the 20-foot separation zone are protected by area-wide fire detection and sprinkler systems.

In letters dated January 31, and November 21, 1986, the licensee requested an exemption from the specific provisions of 10 CFR Part 50, Appendix R, Section III.G.2.b to the extent that it requires no intervening combustibles within the 20-foot separation zone provided between redundant safe shutdown train components in each reactor building of all three units. Specifically, the request stated that each reactor building had open ladder-type cable trays located between redundant cables and equipment with insulation on cables in these trays that was considered an intervening combustible material and that there were no significant in-situ fire hazards present except for the cable insulation in the cable trays.

The NRC, in a letter dated October 21, 1988, granted an exemption from 10 CFR 50, Appendix R, Section III.G.2.b for all three units allowing combustibles in the form of cable trays containing cables coated with fire retardant material in the 20-foot separation zone between redundant trains of safe shutdown equipment provided: 1) the locations in the reactor building had no other in-situ fire hazards or fire loads except for cable insulation which had been coated with a fire retardant material; and, 2) no other combustibles would be permitted in the 20-foot separation space; and, 3) the reactor buildings had area-wide fire detection and sprinkler systems, coupled with manual extinguishers and hose stations, to protect the intervening cables in the affected areas. During walkdowns of the 20-foot separation zone on the 565' elevation of the Unit 1 Reactor Building, the inspectors observed that, in addition to intervening cable trays, the 20-foot separation zone included intervening ignition sources located in 480V Reactor Vent Board 1B. The inspectors observed that the board extended several feet into the separation zone. More importantly, the inspectors were concerned that the 480V Reactor Vent Board that intervened in the 20-foot separation zone represented combustible fire hazards and also did not appear to meet the basis for the approved Section III.G.2.b exemption granted in 1988. TVA's current NRC approved exemption addresses intervening cable trays located in the 20-foot separation zone but does not mention in-situ combustible fire hazards such as the 480V Reactor Vent Board 1B.

The concern was that a fire originating in one fire zone (Fire Zone 1-1 or 1-2, or within the 20-foot separation zone) may spread to another fire zone via the intervening combustible hazards, thus affecting redundant divisions of SSD cables or equipment.

The licensee was questioned as to whether the identified combustible fire hazards were included in the engineering analysis that supported the 1986 exemption request concerning intervening combustibles in the form of cables in trays. The licensee subsequently determined that the 480V Reactor Vent Board existing in the 20-foot separation zone was not specifically addressed in the 1986 exemption request; that 480V Reactor Vent Boards extended into the 20-foot separation zones in similar areas on Units 2 and 3; and that the licensee no longer conformed with the Section III.G.2.b exemption concerning the intervening combustibles in the form of cables in trays. PER 111744 was generated to address this issue. The inspectors also discussed with the licensee a similar issue concerning implementation of DCN 61563 that would introduce additional intervening combustibles (Thermo-Lag) within the 20-foot separation zones located on the 565' and 593' elevations of the Unit 1 Reactor Building. This was identified and discussed in NRC IR 05000259/2005016.

10 CFR 50.48(b) states, that "Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979." Section III.G.2 of Appendix R describes three methods acceptable for ensuring that at least one train of redundant safe shutdown equipment is free of fire damage: (a) redundant trains be located in different fire areas separated by 3-hour rated fire barriers; (b) redundant trains in the same fire area be separated by 20 feet of horizontal distance with no intervening combustibles or fire hazards, and the fire area be equipped with area-wide detection and suppression; or (c) one redundant train be separated from the other redundant trains by enclosing it in a 1-hour fire-rated barrier, and the fire area be equipped with area-wide detection and suppression.

Section III.G.2.b was the method chosen by the licensee to ensure that one division of SSD equipment including the RHR LPCI injection valves would be free of fire damage in FA 1 of the Unit 1 Reactor Building. The inspectors determined that both divisions of RHR LPCI injection valves may be damaged by a fire in the 20-foot separation zone because there are intervening ignition sources and fire hazards such as the 480V Reactor Vent Boards that were not part of the 1986 TVA exemption request.

c. Conclusions

The inspectors identified an URI involving BFN's 1988 NRC-approved exemption from Section III.G.2.b concerning intervening combustibles in the form of cables in trays in all three reactor buildings. The engineering analysis that supported the 1988 exemption was not complete. As such, the exemption request did not address in-situ fire hazards such as the 480V Reactor Vent Board that extended into the separation zones. The exemption request also needed to consider implementation of DCN 61563 that introduced additional intervening combustibles (Thermo-Lag) other than cable insulation in trays within the 20-foot separation zones located on the 565' and 593' elevations of the Unit 1 Reactor Building. The issue is identified as URI 05000259,260,296/2006016-001, Failure to Meet License Basis for an Approved Exemption to 10 CFR 50, Appendix R, Section III.G.2.b for Combustible Fire Hazards Intervening in 20-foot Separation Zones.

On October 26, 2006, TVA submitted to the NRC a revision to the existing 1988 exemption from 10 CFR 50, Appendix R, Sections III.G.2.b. approved by NRC in 1988. This item is unresolved pending further NRC review of the October 26, 2006, revision and to complete fire modeling and risk evaluation to determine the credibility of the potential fire scenarios. PERs 93306 and 111744 were initiated for tracking and resolution of this issue.

V. Management Meetings

X1 Exit Meeting Summary

On September 29, 2006, the inspectors presented the inspection results to Mr. Brian O'Grady and Mr. Masoud Bajestani and other members of their staff, who acknowledged the findings. The inspectors confirmed that proprietary information is not included in this inspection report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Abbas, BFNP Engineering - Mechanical
S. Austin, BFNP Site Licensing
M. Bajestani, BFNP U1 Vice President
R. Baron, BFNP U1 Nuclear Assurance Manager
C. Boschetti, BFNP U2/3 Design Engineering Electrical / Instrumentation & Controls
D. Burrell, BFNP Unit1 Electrical Engineering Lead
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G. Little, BFNP U1 Restart Manager
L. Long, BFNP Fire Protection
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J. McCrary, BFNP U1 Operations Support
R. Moll, BFNP U1 Engineering
B. O'Grady, BFNP Site Vice President
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M. Shulman, BFNP U1 Nuclear Assurance
E. Smith, BFNP Fire Operations
J. Symonds, BFNP U1 Modifications Manager
J. Valente, BFNP U1 Engineering Manager

NRC personnel

W. Bearden, Senior Resident Inspector, Unit 1
R. Fanner, Region II (RII), Reactor Inspector, Division of Reactor Safety (DRS)
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R. Holbrook, RII Resident Inspector Contractor
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L. Mellen, RII Senior Project Engineer BFNP U1 Restart, Division of Reactor Projects (DRP)
N. Merriweather, RII Senior Reactor Inspector,DRS
C. Payne, RII Chief, Engineering Branch 2, DRS
R. Rodriguez, RII Reactor Inspector,DRS
C. Stancil, RII Resident Inspector, DRP
G. Wiseman, RII Senior Reactor Inspector,DRS

INSPECTION PROCEDURES USED

IP 37550 Engineering
IP 64704 Fire Protection Program Inspection Procedure
IP 92701 Followup

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

None.

Opened

05000259,260,296/2006016-01	URI	Failure to Meet License Basis for an Approved Exemption to 10 CFR 50, Appendix R, Section III.G.2. b. for Combustible Fire Hazards Intervening in 20-foot Separation Zones. (Section F1.1)

Closed

None.

Discussed

05000259/2006012-01	URI	Feasibility and Reliability of Local Manual Operator Actions to Achieve Safe Shutdown (Sections E1.3, E1.4)

LIST OF DOCUMENTS REVIEWED

Section E1.3: Review of Circuit Analysis Data For A Sample of SSD Components to Verify That One Train of SSD Components Will Be Available and Free of Fire Damage in the Selected Fire Areas of Interest (37550)

Safe Shutdown Components Examined
0-PMP-023-0001, RHRSW Pump A1
0-PMP-23-0019, RHRSW Pump B2
0-PMP-023-0085, EECW Pump A3
0-PMP-023-0088, RHRSW Pump B3
0-PMP-023-0091, EECW Pump C3
0-PMP-023-0094, RHRSW Pump D3
1-PMP-074-005, RHR Pump 1A
1-PMP-74-0028, RHR Pump 1B
1-FCV-074-0053, LPCI Injection Valve
1-FCV-074-0067, LPCI Injection Valve
0-PMP-026-0001, Motor Driven Fire Pump A
0-PMP-026-0003, Motor Driven Fire Pump C
0-PMP-026-0118, Diesel Driven Fire Pump
0-FU2-211-AAG, 10 ampere fuses (A2Y10)
0-FU1-211-25-45AB, 3 ampere fuse (A6Y3-11)
3-FU2-211-3EAR, 10 ampere fuses (A2Y10-1)
3-FU2-3EAT, 3 ampere fuse (A6Y3-1)

Section E1.4 SSI Attachments / Sections Reviewed

0-SSI-8	Attachment 3	Sections 1-3
0-SSI-1-4	Attachment 1	All sections
0-SSI-1-4	Attachment 2	All sections
0-SSI-1-4	Attachment 3	All sections
0-SSI-1-4	Attachment 4	All sections
0-SSI-16	Attachment 1	Sections 1-2
0-SSI-16	Attachment 5	Sections 1-3

Licensing Bases Documents
BTP Chemical and Material Engineering Branch CMEB 9.5-1 Letter, dated July 1981
Fire Protection Report Volume 1, Section 2, Fire Hazards Analysis, Rev. 23
NRC SERs dated December 8, 1988; March 6, 1991; March 31, 1993; November 2, 1995; and SER Supplement dated November 3, 1989
Browns Ferry Nuclear Plant Units 2 and 3 Fire Protection Report, Volume 1, Rev. 36
Browns Ferry Nuclear Plant Units 1, 2, and 3 draft Fire Protection Report Vol. 1, Section 2, Fire Hazards Analysis, Rev. 35 draft
TVA Letter to NRC, Summary of Deviations from NFPA Code for BFN, dated August 3, 1988
TVA letter, R. Gridley to NRC, Browns Ferry Nuclear Plant (BFN) - Fire Protection Report, dated April 4, 1988
6. NRC letter, S. Black to TVA, Appendix R Exemptions for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC 61124, 61125 and 61126),dated October 21, 1988

Procedures

MAI-1.3, General Requirements for Modification, Rev. 21
SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 9
0-SSI-8, Unit 2, 4KV Electric Board Room 2B, Rev. 0000E
0-SSI-1-4, Unit 1 Reactor Building Fire El 593 South of Column Line R and RHR Heat Exchanger Rooms From El 565 Through 593, Rev. 0000E
0-SSI-16, Control Building Fire El 593 Through El 617, Rev. 0000G
Unit 1 Surveillance Procedure 1-SR-3.3.3.2.1 (74) Backup Control Panel Testing Rev. 0000B

Calculations

ED-Q0999-2003-037, Appendix R Computerized Separation Analysis, Rev. 002
ED-Q0999-940040, Appendix R Computer Separation Analysis (Units 1, 2, and 3), Rev. 17
MDQ110020050013, TVA Browns Ferry Nuclear Plant Reactor Building Thermo-Lag 330-1 Fire Endurance Qualification Calculation, Rev. 0
ED-Q0999-2003-0048 Unit 1,2 and 3 Appendix R Manual Action Requirements, Rev.2

DCNs

DCN 51222 - System 074, Residual Heat Removal, Reactor Building
DCN 60546A - Recirculation Pump Trip for LPCI Injection During an Appendix R Event
DCN 51190 -Ventilation System Modifications, Fire Dampers
DCN 51208 -Penetration seal modifications (includes fire doors), Fire Doors
DCN 61563 -Fire Wrapped Identification Raceways to Meet Appendix R Requirements
DCN 51180 -Suppression system modifications, Pre-action Sprinkler System Unit 1 Reactor Building

Drawings

0-45E765-5, 4160V Shutdown Auxiliary Power Schematic Diagram, Rev. 039
0-45E766-23, 4160V Shutdown Auxiliary Power Schematic Diagram, Rev. 037
0-731E761-10, Elementary Diagram Emergency Equipment, Rev. 020
0-731E761-11, Elementary Diagram Emergency Equipment, Rev. 022
3-45E768-1, Emergency Equipment Diesel Generator 3A Schematic Diagrams, Rev. 019
3-45E768-2, Emergency Equipment Diesel Generator 3A Schematic Diagrams, Rev. 020
3-45E768-3, Emergency Equipment Schematic Diagram Diesel Generator 3B, Rev. 012
3-45E768-4, Emergency Equipment Schematic Diagrams, Rev. 014
3-45E768-5, Emergency Equipment Schematic Diagram Diesel Generator 3C, Rev. 022
3-45E768-6, Emergency Equipment Schematic Diagrams, Rev. 020
3-45E768-7, Emergency Equipment Schematic Diagrams, Rev. 012
3-45E768-8, Emergency Equipment Diesel Generator 3D Schematic Diagram, Rev. 014
0-807E243T, 4KV Shutdown Board A Logic Panel 25-45A, Rev. 006
0-807E243T, Logic Panel 25-45B Connection Diagram, Rev. 004
0-807E245T, Connection Diagram Panel 25-45C, Rev. 008
0-807E246T, Logic Panel 25-45D Connection Diagram, Rev. 005
1-45N1641-4, Wiring Diagrams Unit Control Board Panel 9-3, Rev. 002
0-0106D8860, 4KV Shutdown Board A Connection Diagram, Rev. 005
0-0106D8864, 4KV Shutdown Board C Connection Diagram, Rev. 001
0-0106D8865, 4KV Shutdown Board D Connection Diagram, Rev. 001
1-47E813-1, Flow Diagram Reactor Core Isolation Cooling System, Rev. 21
1-47E812-1, Flow Diagram High Pressure Coolant Injection System, Rev. 20
1-47E811-1, Flow Diagram Residual Heat Removal System, Rev. 28

1-47E801-1, Flow Diagram Main Steam System, Rev. 8
1-47E801-2, Flow Diagram Main Steam System, Rev. 5
1-47E858-1, Flow Diagram RHR Service Water System, Rev. 55
1-47E859-1, Flow Diagram Emergency Equipment Cooling Water System, Rev. 74
1-47E670-Wiring Diagram ECCS Div I Analog Trip Units Schematic Diagram, Rev. 3
2-730E928-1 Sheet 1, HPCI Elementary Diagram, Rev. 24
2-730E928-2 Sheet 2, HPCI Elementary Diagram, Rev. 23
2-730E928-3 Sheet 3, HPCI Elementary Diagram, Rev. 22
2-730E928-4 Sheet 4, HPCI Elementary Diagram, Rev. 10
2-730E928-5 Sheet 5, HPCI Elementary Diagram, Rev. 24
2-730E928-7 Sheet 7, HPCI Elementary Diagram, Rev. 11
2-730E928-8 Sheet 8, HPCI Elementary Diagram, Rev. 21
2-45E670-19, Wiring Diagram ECCS Div II Analog Trip Units Schematic Diagram, Rev. 18
2-730E929-1 Sheet 1, Elementary Diagram Automatic Blowdown System, Rev. 23
2-730E929-2 Sheet 2, Elementary Diagram Automatic Blowdown System, Rev. 23
2-730E929-3 Sheet 3, Elementary Diagram Automatic Blowdown System, Rev. 16
2-730E929-4 Sheet 4, Elementary Diagram Automatic Blowdown System, Rev. 17
2-730E929-5 Sheet 5, Elementary Diagram MSRV Auto Actuation Logic, Rev. 2
0-45E732-1, Wiring Diagram 480V Diesel Auxiliary Bd A Single Line, Rev. 37
0-45E732-2, Wiring Diagram 480V Diesel Auxiliary Bd A Single Line, Rev. 23
0-47W216-51, Appendix R Fire Area Compartmentation and Zone Drawings, Index and Notes, Rev. 5
0-47W216-52, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 519.0, Rev. 1
0-47W216-54, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 565.0, Rev. 3
0-47W216-56, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 593.0, Rev. 4
0-47W216-57, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 621.0, Rev. 1
1-47E610-26-1, Mechanical Control Diagram, High Pressure Fire Protection System, Rev. 28
0-47W216-51, Appendix R Fire Area Compartmentation and Zone Drawings, Index and Notes, Rev. 5
0-47W216-56, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 593.0, Rev. 4
0-47W216-57, Appendix R Fire Area Compartmentation and Zone Drawings, Plan El. 621.0, Rev. 1
1-47B1920-658 & 659, Heating, Ventilating, & Air Conditioning Fire Damper Details, Rev. 4
1-47B1920-677 & 668, Heating, Ventilating, & Air Conditioning Fire Damper Details, Rev. 5
1-47E491-2-1, Mechanical Fire Protection Plan-El. 621', Rev. 2
1-47E491-3-1, Mechanical Fire Protection Plan-El. 593', Rev. 3
1-47E491-3-3, Mechanical Fire Protection Plan-El. 593', Rev. 1
1-47E865-1, Flow Diagram Heating and Ventilating Air Flow, System 64 - Powerhouse, Reactor Building Unit 1, Rev. 39
1-47E1392-367, Appendix R Penetration Seal Location Drawings, El. 593.0, Rev. 1
1-47E1392-377, Appendix R Penetration Seal Tabular Drawings, El. 593.0, Rev. 1
1-47E1392-713, Appendix R Penetration Seal Location Drawings, El. 639.0, Rev. 0

1-47E1392-717, Appendix R Penetration Seal Tabular Drawings, El. 639.0, Rev. 1

Other Documents

Final Report - Coordination Testing Of Circuit Breaker/Fuse And Fuse/Fuse Combinations For Browns Ferry Nuclear Plant, Rev. 0, Dated October 16, 1990
Attachment 6, Calculation ED-Q2000-870550, Sheet 1 of 7, Form 600 Fuses Data
Calculation ED-Q2000-8705, Appendix C Sheets 6A and 6B of 29, Rev. 14
BFN-50-7026, High Pressure Fire Protection System, Rev. 6
NFPA 20, Standard for the Installation of Centrifugal Fire Pumps, 1987 Edition
NFPA 13, Installation of Sprinkler Systems, 2002 Edition
NFPA 80, Standard for Fire Doors and Fire Windows, 1999 Edition
NFPA 90A, Standard for Fire Dampers, 1981 Edition
Underwriters Laboratory 555, Standard for Fire Dampers and Ceiling Dampers, 3rd Edition
PSI-001, Technical Evaluation of Fire Rated Material Applied to Damper Sleeves, 3M INTERAM E-50 Series Flexible Wrap, for Preferred Metal Technologies, Rev. 0, 10/21/2003
Letter D. Priest, Omega Point Laboratories, to M. Murphy, PCI-Promatec, "Fire Penetration Seal Design for Dampers," dated 10/14/2003
Completed Procedure Walkdown Package for 0-SSI-1-4, Rev. 0000E
Completed Procedure Walkdown Package for 0-SSI-8, Rev. 0000E
Completed Procedure Walkdown Package for 0-SSI-16, Rev. 0000G

Condition Reports Reviewed During This Inspection

93306, NRC Concern on Thermo-Lag
101631, Appendix R Section III.G.2 Operator Manual Actions
110479, Loss of U2/U3 Fire Pump Start
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

Condition Reports Generated as a Result of This Inspection

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
111744, Appendix R Zone of Separation Issue

ACRONYMS

10 CFR 50	Title 10, Part 50 of the Code of Federal Regulations
AOV	Air Operated Valve
BFNP	Browns Ferry Nuclear Plant
DCN	Design Change Notice
DRS	Division of Reactor Safety
EB	Engineering Branch
ERFBS	Electrical Raceway Fire Barrier System
FPP	Fire Protection Plan
FPR	Fire Protection Report
IMC	Inspection Manual Chapter
MCR	Main Control Room
NFPA	National Fire Protection Association
MOV	Motor Operated Valve
NRC	United States Nuclear Regulatory Commission
OMA	Operator Manual Action
PER	Problem Evaluation Report
RHR	Residual Heat Removal System
RHRSW	Residual Heat Removal Service Water System
RII	Region II
SE	Safety Evaluation
SSD	Safe Shutdown
SSI	Safe Shut-down Instructions
TVA	Tennessee Valley Authority
UL	Underwriters Laboratory
URI	Unresolved Item
V	Volts