UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

February 9, 2010

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000259/2009008, 05000260/2009008, AND 05000296/2009008

Dear Mr. Krich:

On December 28, 2009, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Browns Ferry Nuclear Plant, Units 1, 2, and 3. The enclosed inspection report documents the inspection findings which were discussed on November 20, 2009 with Mr. Rusty West, Site Vice President, and on December 28, 2009, via telephone, with Mr. Jim Randich, Plant Manager as well as 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.

Based on the results of this inspection, the inspectors identified two findings of very low safety significance (Green), which involved violations of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of their 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 VI.A.1 of the NRC's Enforcement Policy. If you contest any of these NCVs you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Browns Ferry Nuclear Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at Browns Ferry Nuclear Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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/

Binoy B. Desai, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000259/2009008, 05000260/2009008, and 05000296/2009008 w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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cc w/encl:
K. J. Polson
Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

J.J. Randich
General Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

F.R. Godwin
Manager, Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

E. J. Vigluicci
Assistant General Counsel
Tennessee Valley Authority
6A West Tower
400 West Summit Hill Drive
Knoxville, TN 37902

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 30317
Montgomery, AL 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Browns Ferry Nuclear Plant
U.S. Nuclear Regulatory Commission
10833 Shaw Road
Athens, AL 35611-6970

Letter to Preston D. Swafford from Binoy B. Desai dated February 9, 2010.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC COMPONENT DESIGN BASIS INSPECTION REPORT NOS. 05000259/2009008, 05000260/2009008, AND 05000296/2009008

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Eva Brown, NRR
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2009008, 05000260/2009008, and 05000296/2009008
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	October 19, 2009 through December 28, 2009
Inspectors:	R. Berryman, P.E., Senior Reactor Inspector (Lead) R. Aiello, Senior Operations Engineer C. Even, Reactor Inspector R. Patterson, Reactor Inspector J. Leivo, Contractor H. Campbell, Contractor
Accompanying Personnel:	J. Kent, Construction Inspector (trainee)
Approved by:	Binoy B. Desai, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2009008, 05000260/2009008, and 05000296/2009008; 10/19/09 - 12/28/09; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Component Design Basis Inspection.

This inspection was conducted by a team of four NRC inspectors, one NRC inspector who was in training, and two NRC contract 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 IMC 0609, "Significance Determination Process" (SDP). Finding for which the SDP does not apply may be Green or is 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," (ROP) Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation (NCV) of TS 5.4.1 for the failure to have an adequate procedure that would ensure tornado depressurization protection of the emergency diesel generators (EDGs). Abnormal Operating Instruction, 0-AOI-100-7, "Severe Weather," did not provide guidance on how to provide pressure equalization of the EDG building for mitigating atmospheric depressurization associated with tornado conditions that could impact the EDG building ventilation system. The design of the EDG ventilation system intake and exhaust dampers requires the dampers to be manually opened prior to a tornado depressurization event to ensure the EDG building and ventilation system remain intact and operable during and after a tornado. This finding was entered into the licensee's corrective action program as problem evaluation report (PER) 206919. As an immediate corrective action, the licensee added steps to procedure 0-AOI-100-7 to station an operator in the EDG building to perform required manual actions in the event of a tornado warning in the area.

  This finding is more than minor because it affects the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and operability of the EDGs to perform the intended safety function during a design basis event and the cornerstone attribute of Procedure Quality, i.e. Operating (Post Event) Procedures (AOPs). The inspectors assessed the finding using a Phase I Significance Determination Process (SDP) screening which determined a Phase III SDP evaluation was required due to the fact that the finding involved the loss or degradation of equipment specifically designed to mitigate a severe weather initiating event (e.g., tornado doors). The loss of this equipment by itself, during the external initiating event it was intended to mitigate, would degrade two or more trains of a multi-train safety system. A Phase III SDP evaluation was performed in accordance with NRC Inspection Manual Chapter 0609 Appendix A by a regional Senior Reactor Analyst using the NRC Standardized Plant Analysis

Risk (SPAR) model. The analysis determined that the performance deficiency resulted in a core damage frequency (CDF) risk increase of less than 1E-6. Therefore, the finding was characterized as having very low safety significance (Green). The large early release frequency (LERF) result was less than 1E-7 which would not override the CDF risk characterization. The dominant sequence was a tornado which caused damage to the EDG dampers for all EDGs and resulted in a loss of offsite power (LOOP). The risk was mitigated by the low initiating event likelihood for tornadoes and the probability for EDG recovery which was proceduralized.

The inspectors determined that the use of operating experience (OE) information was a significant cause of this performance deficiency. Regulatory Information Summary 2006-03, "Post Tornado Operability of Ventilation and Air Conditioning Systems" as well as an internal licensee OE had raised a similar concern. The licensee was unaware of the vulnerability of the EDG ventilation system to tornado depressurization events until it was brought to their attention by the inspectors. The licensee's failure to use available OE is directly related to the OE component of the cross-cutting area of Problem Identification and Resolution and the aspect of implementing OE through changes to procedures (P.2.(b)). (Section 1R21.2.12)

- Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to provide adequate guidance in existing procedures utilized for flow balancing of the emergency equipment cooling water (EECW) system. The EECW system provided the heat sink for station safety-related heat loads including cooling for the residual heat removal (RHR) and core spray (CS) room coolers. The installed strainers on the EECW system are capable of filtering debris greater than 1/8 inch (.125 inches), potentially allowing debris less than 1/8 inch to pass through and clog downstream throttle valves. A clog in the throttle valves would prevent adequate flow from reaching safety-related heat exchangers unless procedural guidance or limitations prevented throttling valves to disk-to-seat clearances of less then 1/8 inch. The existing EECW flow balance procedure was inadequate in that it made no provision in the acceptance criteria to limit or evaluate minimum throttle valve seat/disc clearance, and the subsequent potential for increased flow obstruction, resulting from system flow balancing. This finding was entered into the licensee's corrective action program as problem evaluation reports (PERs) 208374 and 208636. Planned corrective actions included a revision to EECW flow balancing procedures. The inspectors verified and discussed with the licensee existing indications that are available to alert the operator of potential clogging.

  This finding is more than minor because it affects the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and operability of the RHR and CS pump room coolers to perform the intended safety function during a design basis event and the cornerstone attribute of Procedure Quality, i.e. Maintenance and Testing (Pre-event) Procedures. The team assessed this finding using the Significance Determination Process (SDP) and determined that

the finding was of very low safety significance (Green) because the inspectors found no documented history of an actual loss of safety system function. This finding was reviewed for cross-cutting aspects and none were identified. (Section 1R21.2.15)

B. Licensee-Identified Violations

- One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk-significant components and operator actions for review using information contained in the licensee's Probabilistic Risk Assessment (PRA). In general, this included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1 $X10^{-6}$. The components selected were located within the residual heat removal (RHR) system, plant and drywell control air systems, main steam isolation valves (MSIVs), emergency diesel generator (EDG) support and electrical subsystems, 4160 VAC electrical system, common station service transformers (CSSTs), 125/250 VDC battery system, and main steam line radiation monitors. The sample selection included 17 components, five operator actions, and five operating experience items. Additionally, the team reviewed one modification related to extended power uprate activities documented in Section 4OA5 of this report.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases had been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modification, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results, significant corrective action, repeated maintenance, maintenance rule (a)1 status, RIS 05-020 (formerly GL 91-18) conditions, NRC resident inspector input of problem equipment, system health reports, industry operating experience and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. An overall summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 <u>RHR Pump B and D Minimum Flow Valve (MOV FCV-74-30)</u>

a. <u>Inspection Scope</u>

The team reviewed design and licensing documentation for motor-operated valve (MOV) FCV-74-30, including the Updated Final Safety Analysis Report (UFSAR), Design Basis Documents (DBDs), and drawings to verify the capability of the valves to meet design basis requirements. The design function of these valves is to protect the RHR Loop II pumps from possible damage during low flow conditions; at high pump flow the valve

closes while low pump flow results in a signal to open the valve thereby providing a minimum flow path. Orifice sizing, setpoint, and related flow calculations were reviewed to verify that they were appropriate. Additionally, discussions with design engineering personnel were conducted and completed surveillance test results were reviewed to verify that the recirculation flow path was adequate.

b. Findings

No findings of significance were identified.

.2.2 RHRSW to A Train RHR Heat Exchanger Check Valve (1-CKV-23-510)

a. Inspection Scope

The team reviewed portions of the RHR service water (RHRSW) and RHR system DBDs and drawings to determine the functional requirements for valve 1-CKV-23-510. This valve allows forward flow through RHRSW coolant heat exchanger and prevents potential reverse flow from the A train RHR heat exchanger. The inspectors also conducted a walkdown of the RHRSW pumps, piping and valves, with particular focus on CKV-23-510. Additionally, the inspectors verified that a program was in place to appropriately monitor and address any valve degradation.

b. Findings

No findings of significance were identified.

.2.3 A and B Train RHR Supply to Torus Cooling (MOV FCV-74-57 and -59)

a. Inspection Scope

The team reviewed applicable DBDs, drawings, and calculations to verify the capability of these valves to perform the design basis functions. American Society of Mechanical Engineers (ASME) Section XI stroke-timing test data was reviewed for the valves to verify that requirements were satisfied and potential adverse trends were monitored. Test procedures were also reviewed to evaluate the valve throttling capability during RHR pump flow testing to verify it was adequate. Additionally, calculations documenting the design requirements of maximum differential pressures, actuator requirements and weak-link evaluations were reviewed to verify that they were appropriate.

b. Findings

No findings of significance were identified.

.2.4 RHR Pumps A, B, C and D

a. Inspection Scope

The team reviewed portions of the UFSAR, Technical Specifications (TS), and the RHR DBD to verify the capability of the pumps to perform the required function during postulated design basis events. Calculations establishing pump performance requirements during design basis events were reviewed to verify that the conclusions were appropriate. In-service testing (IST) and comprehensive test results were reviewed to assess potential pump degradation and to verify that pump performance was sufficient to satisfy design basis accident requirements. In addition, the team reviewed the system health reports, corrective action documentation, and the station response to Bulletin 88-04, "Potential Safety-Related Pump Loss," to verify that any potential degradation was being tracked and addressed appropriately. Additionally, the inspectors interviewed the RHR system engineer to discuss the overall health and performance history of the RHR system. The inspectors also conducted a walkdown of the Unit 1 loop II pumps and associated piping and valves to verify that any adverse material conditions were being appropriately addressed.

b. Findings

No findings of significance were identified.

.2.5 EDG-Mechanical (Unit 1-2 EDGs A, B, C, D)

a. Inspection Scope

The team reviewed portions of the UFSAR, EDG DBD, calculations, System health reports, related problem evaluation reports (PERs), operating procedures, and a selection of completed TS surveillances to verify the capability of the EDGs to supply onsite AC electrical power during design basis events. A walkdown of the B EDG was conducted to verify that adverse material conditions, if any, were being appropriately addressed. The inspectors also reviewed the EDG fuel oil transfer pump test procedures and completed test results to verify that the fuel oil transfer pumps were capable of performing their required function during design basis events.

b. Findings

No findings of significance were identified.

.2.6 Unit 1 Inboard Main Steam Isolation Valves (MSIVs), (1-FCV-1-14, -26, -37, and -51)

a. Inspection Scope

The team reviewed applicable portions of the UFSAR, TS, calculations, and drawings to verify the capability of the MSIVs to perform the required functions during design basis events. The inspectors interviewed the system engineer to discuss the overall health of the MSIVs and associated PERs to verify that adverse material conditions, if any, were

being appropriately addressed. The inspectors also reviewed design change notice (DCN) 51143, “Main Steam and Standby Liquid Control System.” This DCN was reviewed by the team, with focus on the technical evaluation of the major changes to the MSIVs required for a planned power uprate. The associated 10 CFR 50.59 Evaluation and related thermal hydraulic calculations for this DCN were also reviewed by the team to verify that the results were appropriate. MSIV stroke-timing test results were also reviewed to verify that any degradation was being appropriately monitored and addressed.

b. Findings

No findings of significance were identified.

.2.7 250 VDC Battery Board 1

a. Inspection Scope

The team selectively reviewed electrical one-line diagrams, loading calculations, voltage drop calculations, short-circuit calculations, and associated electrical protection to determine the capability of the battery board to serve the required power to 250 VDC loads in accordance with the design and licensing basis as well as for station blackout (SBO) events. For a sample of limiting conditions identified for the battery boards in the design calculations, the team reviewed operating procedures to confirm that the procedures were consistent with the actions prescribed by the calculations for selected out-of-service configurations. The team reviewed recent system health reports, a sample of available preventive maintenance and surveillance records, and associated corrective action history. The team also performed a non-intrusive visual inspection of the battery board to assess the installation configuration, material condition, and potential vulnerability to external hazards to verify that any degraded conditions were being appropriately addressed.

b. Findings

Introduction. The team indentified an Unresolved Item (URI) regarding safety-related molded-case circuit breakers (MCCBs) in safety-related applications. TVA had not implemented a test program to detect potential deterioration or to assure that all installed safety-related MCCBs would perform satisfactorily in service.

Description. TVA procedure 0-TI-395, “Breaker Testing and Maintenance Program,” required that critical molded case circuit breakers be subject to preventative maintenance (PM) activities. This included, in part, inspection for overheating, mechanical operation, enclosure inspection, overload trip testing, and instantaneous trip testing. The TVA program required that the testing and PM activities be performed every four to six years. UFSAR 8.6.4.1.1 stated, in part, that zero-resistance short circuits at the battery board or any point downstream can be cleared by the breakers operating within their ratings. To ensure this was satisfied by the installed equipment, degradation of breaker performance should have been detectable and acceptably controlled by periodic testing and preventive maintenance. Additionally, UFSAR

8.5.2.11 stated, in part, that the standby AC power system will meet or exceed the requirements of IEEE-308, Criteria for Class 1E Power Systems at Nuclear Generating Stations. This standard recommended that periodic tests be performed at scheduled intervals to detect deterioration of equipment and to demonstrate operability of components that are not exercised during normal operation.

Four MCCBs on 250 VDC Battery Board 1 were selected by the team (breakers 607, 705, 712, 715) as part of the inspection sample. No records of PM activities were found on these four MCCBs. While addressing the extent of this condition, TVA identified a total of 622 safety-related MCCBs in both AC and DC applications for which no testing was being performed.

As a result of the inspector's observations, TVA initiated PER 209095 and scheduled testing of the 622 safety-related MCCBs that were not being tested in accordance with the TVA PM program. As of December 28, 2009, TVA had successfully tested 30 of the 622 breakers. This represented a small sample of the total breaker population. The inspectors plan to review additional PM test results to determine if this performance deficiency is more than minor.

Summary. This issue is unresolved pending further inspection to determine the extent of condition and impact of not implementing a test program to assure that all installed safety-related MCCBs would perform satisfactorily. (URI 05000259, 260, 296/2009008-01, Safety-Related Molded Case Circuit Breakers)

.2.8 4160 VAC Shutdown Board C

a. Inspection Scope

The team selectively reviewed electrical one-line diagrams, loading calculations, short circuit calculations, and electrical protection, to determine the capability of shutdown board C to serve the required power to the associated 4160 VAC loads in accordance with the design and licensing basis. The team also selectively reviewed the schematic diagrams for the shutdown board trip and transfer circuits to confirm that the design basis for shutdown board interlocks and transfer to the diesel generators was satisfied and that no credible single failure vulnerabilities existed in the logic circuits. The team reviewed selected results of preventive maintenance and protective relay surveillances and functional tests to verify that any equipment performance degradation was being adequately addressed. The team reviewed system health reports and associated corrective action history to verify that any degraded conditions were being appropriately addressed. The team also performed a non-intrusive visual inspection of the shutdown board and logic panel to assess the installation configuration, material condition, and potential vulnerability to external hazards to verify that any degraded material conditions were being appropriately addressed.

b. Findings

No findings of significance were identified.

.2.9 Transformer 161 KV/4160 VAC CSST-A

a. Inspection Scope

The team reviewed the electrical one-line diagrams, loading calculations, electrical protection, tap settings, and nameplate data, to determine the adequacy of transformer 161KV/4160 VAC CSST-A to supply required power as a qualified alternate source under design and licensing basis conditions. The team reviewed recent results of transformer preventive maintenance including dissolved gas analysis, temperature trending, and protective relay calibrations to verify that equipment performance degradation was being appropriately addressed. The team reviewed recent system health reports and associated corrective action history to verify that any degraded conditions were being appropriately addressed. The team also interviewed the transformer system engineer and performed a non-intrusive visual inspection of the transformer to assess the installation configuration, material condition, and potential vulnerability of the transformer to external hazards to verify that any degraded material conditions were being appropriately addressed.

b. Findings

No findings of significance were identified.

.2.10 EDG (Electrical)

a. Inspection Scope

The team reviewed related design basis documentation, drawings, TS, and the UFSAR to identify design, maintenance, and operational requirements for the EDGs. EDG start, stop, and shutdown circuits were reviewed to verify that the logic of operation was consistent with the accident analysis and that the energy sources used for control functions would be available and unimpeded during accident conditions. The team reviewed EDG loading calculations including voltage, frequency and loading sequences to verify the capability of the EDGs to perform their intended safety function. The team also reviewed diesel batteries including voltage calculations, maintenance requirements, and operating procedures to verify that the terminal voltage of the battery would remain at or greater than design basis minimum voltage during the worst case design basis accident. Surveillance test procedures were reviewed to verify that applicable test acceptance criteria and test frequency requirements for the EDGs were satisfied. Field walkdowns were performed to assess observable material conditions and verify that system alignment was consistent with current drawings and operating procedures. Selected PERs and work orders were reviewed by the team to verify that any potential component degradation was monitored and appropriately addressed.

b. Findings

No findings of significance were identified.

.2.11 RHR Pump Motors

a. Inspection Scope

The team reviewed the TS, UFSAR, and design basis documentation to identify the RHR pump motors' design basis functions and related accident analysis assumptions. The team reviewed the adequacy, reliability, and availability of the power supply to the RHR pumps in normal and degraded voltage conditions. This review included review of motor specifications, vendor manual requirements, and breaker specifications to verify that they were appropriate. A walk-down of the motors and pumps was conducted to assess the physical condition of the motor and verify that the system configuration was consistent with the design basis assumptions, system operating procedures, and plant drawings.

b. Findings

No findings of significance were identified.

.2.12 EDG Building Ventilation Dampers

a. Inspection Scope

The team reviewed the UFSAR, system description, abnormal operating instructions (AOIs), control logic, drawings and operator actions to verify that the EDG building intake and exhaust ventilation dampers could perform their function of providing cooling to components in the EDG building. A system walkdown was performed to verify physical damper dimensions and that linkages could operate as designed. The team also reviewed selected operating experience issues dealing with EDG building ventilation dampers to verify that issues at other facilities were being appropriately identified and addressed.

b. Findings

Introduction. The inspectors identified a Green non-cited violation (NCV) of TS 5.4.1 for the failure to have an adequate procedure that would ensure tornado depressurization protection of the EDGs. Abnormal Operating Instruction, 0-AOI-100-7, "Severe Weather," did not provide guidance on how to provide pressure equalization of the EDG building for mitigating atmospheric depressurization associated with tornado conditions that could impact the EDG building ventilation system. The significance of this violation was determined using Phase III of the Significance Determination Process (SDP).

Description. The EDGs are part of the standby AC power supply and distribution system. BFN UFSAR Section 8.5.2 states, in part, that the standby AC power system will meet or exceed the requirements of IEEE-308. IEEE-308-1971, "Standard Criteria

for Class 1E Power Systems for Nuclear Power Generating Stations" Section 4.1 states, in part, that Class 1E electric systems shall be designed to assure that a tornado will not cause a loss of electric power to a number of engineered safety features sufficient to jeopardize the safety of the station. BFN UFSAR Section 12.2.1 states, in part, that loading considerations for structures should include tornado depressurization. However, there was not an explicit requirement for the protection of the Class 1E power systems from environmental events to be accomplished by a specific fixed design feature or by procedural guidance to provide protection using system configuration.

The EDG ventilation cooling system intake and exhaust dampers are interlocked with the ventilation fans that take outside air to cool the components inside the EDG room. The intake and exhaust ventilation dampers are normally closed, and open, when the EDG starts or when the ventilation fans are manually started. Manual operation of the ventilation fans is performed locally. When the EDGs are not running, the EDG ventilation system intake and exhaust dampers would have to be manually opened prior to a tornado depressurization event to ensure the EDG building and ventilation system remain intact and operable during and after a tornado. Procedure 0-AOI-100-7 did not provide guidance during a tornado warning to verify that the ventilation dampers in the EDG building are opened. In a tornado event where the EDGs are not running, this condition would result in the inability to mitigate atmospheric depressurization of the EDG building associated with a tornado. This represented a potential common-cause failure of all eight EDG ventilation systems and subsequent potential inoperability of the EDGs. As an immediate corrective action, the licensee added steps to 0-AOI-100-7 procedure to station an operator in the EDG building to perform required manual actions in the event of a tornado warning in the area.

Analysis. The failure to have an adequate procedure to ensure tornado depressurization protection of the EDG buildings is a performance deficiency. This finding is more than minor because it affects the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and operability of the EDGs to perform the intended safety function during a design basis event and the cornerstone attribute of Procedure Quality, i.e. Operating (Post Event) Procedures (AOPs). The inspectors assessed the finding using a Phase I SDP screening which determined a Phase III SDP evaluation was required due to the fact that the finding involved the loss or degradation of equipment specifically designed to mitigate a severe weather initiating event (e.g., tornado doors). The loss of this equipment by itself, during the external initiating event it was intended to mitigate would degrade two or more trains of a multi-train safety system. A Phase III SDP evaluation was performed in accordance with NRC Inspection Manual Chapter 0609 Appendix A by a regional SRA using the NRC SPAR model. The analysis determined that the performance deficiency resulted in a core damage frequency (CDF) risk increase less than 1E-6. Therefore, the finding was characterized as having very low safety significance (Green). The large early release frequency (LERF) result was less than 1E-7 which would not override the CDF risk characterization. The dominant sequence was a tornado which caused damage to the EDG dampers for all EDGs and resulted in a loss of offsite power (LOOP). The risk was mitigated by the low initiating event likelihood for tornadoes and the probability for EDG recovery which was proceduralized.

The inspectors determined that the use of operating experience (OE) information was a significant cause of this performance deficiency. Regulatory Information Summary 2006-03, "Post Tornado Operability of Ventilation and Air Conditioning Systems" as well as an internal licensee OE had raised a similar concern. The licensee was unaware of the vulnerability of the EDG ventilation system to tornado depressurization events until it was brought to their attention by the inspectors. The licensee's failure to use available OE is directly related to the OE component of the cross-cutting area of Problem Identification and Resolution and the aspect of implementing OE through changes to procedures (P.2.(b)).

Enforcement. Technical Specification 5.4.1, "Procedures", requires in part that procedures shall be established, implemented, and maintained covering the activities in Regulatory Guide (RG) 1.33, Rev. 2, "Quality Assurance Program Requirements." Appendix A of RG 1.33 states, in part, that typical safety-related activities such as combating emergencies and other significant events including acts of nature, e.g., tornado, shall be covered by written procedures. Contrary to the above, TVA did not have adequate procedures to ensure tornado depressurization protection of the EDG buildings. This condition has existed since plant initial operation. Because this finding is of very low safety significance (Green) and is entered into the TVA corrective action program as PER 206919, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000259, 260, 296/2009008-01, Violation of Technical Specification 5.4.1 for Failure to Develop Adequate Procedures to Ensure Tornado Depressurization Protection of the Emergency Diesel Generators)

.2.13 Main Steam Line Radiation Monitors

a. Inspection Scope

The team reviewed the TS, UFSAR, and DBDs for the main steam line radiation monitors to identify the component design basis functions and related accident analysis assumptions. The team reviewed the adequacy, reliability, and availability of the power supply to the main steam line radiation monitors. This review included review of vendor manual requirements and breaker specifications. The team also reviewed the calibration settings, calculations and procedures to verify that the main steam line radiation monitors could operate consistent with the design basis assumptions, and system operating procedures. The inspectors reviewed system health, maintenance, and corrective action documents to determine whether the equipment had exhibited adverse performance trends.

b. Findings

No findings of significance were identified.

.2.14 RHR Pump A and C Minimum Flow Shutoff Return Valve to Suppression Pool (MOV FCV-74-7)

a. Inspection Scope

The inspectors reviewed applicable portions of the TS, UFSAR, DBDs for the RHR system to identify the component design basis function and related accident analysis assumptions. The team conducted a visual inspection of MOV FCV-74-7 to verify that any degraded material conditions were being appropriately addressed. In addition, the team verified that the power demand requirements for the valves were captured in electrical load and degraded voltage calculations. The team also verified that the worst case/highest differential pressure (dP) was used to determine the maximum valve opening and/or closing requirements to ensure that the valve would perform its intended safety-related design basis function. A review was conducted of the licensee's testing procedures and results from actual diagnostic valve testing that was performed to verify the MOV was tested in a manner that would detect a malfunctioning valve and verify compliance with GL 89-10 program plan requirements. A review of operating and maintenance procedures was conducted to verify that a malfunctioning MOV could be identified by operators, and that the valves were being properly maintained. The team reviewed historical maintenance work orders and modification packages associated with these valves to verify that they were being properly maintained to ensure reliability and margin for safe operation.

b. Findings

No findings of significance were identified.

.2.15 EECW Throttle Valve to RHR Pump Room Cooler 1C (HOV 67-565)

a. Inspection Scope

The team reviewed the system DBD, related design basis support documentation, drawings, TS, and the UFSAR to identify design, maintenance, and operational requirements for the RHR and core spray (CS) room cooler throttle valves. The team reviewed the cooling specifications, design bases information and supporting calculations to identify system flow requirements. The team reviewed the procedures and results of room cooler and strainer inspections and cleanings, flow balancing and dP testing and trending to verify that degraded conditions were being appropriately addressed. The team reviewed the adequacy of the emergency equipment cooling water (EECW) flow to the room cooler that is controlled by the manually operated flow control valve. Component related PERs, corrective maintenance activities, and system health reports were reviewed to evaluate TVA's capability for detection, monitoring, and correcting potential degradation. A field walkdown was performed with the system engineer to assess observable material conditions and verify that the system configuration was consistent with the design basis assumptions, system operating procedures, and plant drawings.

b. Findings

Introduction. The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, for a failure to provide adequate procedures for flow balancing of the EECW system. The EECW flow balance procedure was inadequate in that it made no provision in the acceptance criteria to limit or evaluate minimum throttle valve seat/disc clearance, and the subsequent potential for increased flow obstruction, resulting from system flow balancing.

Description. Preventative Maintenance Test Procedure 3-PMT-BF-067.042,"EECW Flow Balance," provides the guidance for performing a flow balance of the EECW system. This flow balancing procedure was developed in 1995. The EECW system provided the heat sink for station safety-related heat loads including cooling for the RHR and CS room coolers. The flow balance procedure acceptance criteria addressed maintaining minimum flows to the safety-related equipment. The installed strainers on the EECW system are capable of filtering debris greater than 1/8 inch (.125 inches), potentially allowing debris less than 1/8 inch to pass through and clog downstream throttle valves. A clog in the throttle valves would prevent adequate flow from reaching safety-related heat exchangers unless procedural guidance or limitations prevented throttling valves to disk-to-seat clearances of less then 1/8 inch.

The inspectors evaluated the seat and disc geometry of the installed flow control throttle valves and determined that valves 2-THV-067-0551 and 2THV-067-0594 had disk-to-seat clearances less than the 1/8 inch debris size. These valves on the EECW system provide cooling water to the RHR and CS room coolers respectively. The inspectors concluded that there was a potential for flow obstruction in the throttle valves due to the throttled valve positions resulting in disk-to-seat clearances less than the design size of the EECW strainers. The team noted that the EECW flow balancing procedure contained no provision in the acceptance criteria to limit or evaluate minimum disk-to-seat clearance with respect to content of debris in the system or EECW strainer gap design when changing position of flow control valves. Additionally, there was no reference in the limits and precautions section of the procedure related to the increased potential for flow obstruction due to throttling the flow control valves. The licensee entered this issue into their corrective action program with actions to further evaluate the EECW flow balancing procedure and to correct the throttle positions of the 2-THV-067-0551 and 2THV-067-0594 valves. The inspectors verified and discussed with the licensee existing indications that are available to alert the operator of potential clogging.

Analysis. The failure to provide adequate procedures for flow balancing of the EECW system is a performance deficiency. This finding is more than minor because it affects the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and operability of the RHR and CS pump room coolers to perform the intended safety function due to potential clogging from debris of the throttle valves that supply cooling water during a design basis event. This affected the cornerstone attribute of Procedure Quality, i.e. Maintenance and Testing (Pre-event) Procedures. The team assessed this finding using the SDP and determined that the finding was of very low safety significance (Green) because the inspectors found no documented history of an actual loss of safety

system function. This finding was reviewed for cross-cutting aspects and none were identified.

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented procedures include appropriate qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the EECW system flow balance procedure did not include controls to preclude flow obstruction caused by reduced valve disk-to-seat clearance during valve throttling below 1/8 of and inch. This condition had existed since at least 1995. Because this finding was of very low safety significance and was entered into the TVA corrective action program as PERs 208374 and 208636, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000259, 260, 296/2009008-02, Violation of 10CFR50, Appendix B, Criterion V for Inadequate Procedure for Emergency Equipment Cooling Water System Flow Balancing)

.2.16 Plant Control Air Receiver Relief Valve (0-32-556)

a. Inspection Scope

The team reviewed the system DBD, related design basis support documentation, drawings, TS, and the UFSAR to identify design, maintenance, and operational requirements for the plant control air safety relief valve. Maintenance history and associated PERs were reviewed to verify that potential degradation was being monitored and appropriately addressed. Setpoint procedures were reviewed to verify that appropriate design inputs and vendor tolerances were appropriately incorporated into testing acceptance criteria. The team conducted a field walkdown of the relief valve to verify that the installed configuration was consistent with the design basis and plant drawings and to assess observable material conditions to verify that any degraded conditions were being appropriately addressed.

b. Findings

No findings of significance were identified.

.2.17 Drywell Control Air Check Valves (32-2516, 2521, 2163 and 0336)

a. Inspection Scope:

The team reviewed the DBD, related design basis support documentation, drawings, TS, and the UFSAR to identify design, maintenance, and operational requirements for selected drywell control air system check valves. Maintenance history, as demonstrated by system health reports, preventive and corrective maintenance, and PERs, were reviewed to verify that any degradation was being appropriately monitored and addressed. The team conducted interviews with the control air (CA) system engineer to obtain additional information and verify that the station's implementation and analysis of industry operating experience related to check valves was appropriate.

b. Findings

No findings of significance were identified.

.2.18 Plant Control Air Check Valves (32-2171, 0243)

a. Inspection Scope

The team reviewed the DBD, related design basis support documentation, drawings, TS, and the UFSAR to identify design, maintenance, and operational requirements for selected drywell control air system check valves. Maintenance history, as demonstrated by system health reports, preventive and corrective maintenance, and PERs, were reviewed to verify that any degradation was being monitored and addressed. The team conducted interviews with the CA system engineer to obtain additional information and verify that the station's implementation and analysis of industry operating experience related to check valves was appropriate. The team also conducted a field walkdown of these check valves to verify that the installed configuration was consistent with the design basis and plant drawings and to assess observable material conditions to verify that any degraded conditions were being appropriately addressed.

b. Findings

No findings of significance were identified.

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of five risk-significant and time-critical operator actions. Where possible, margins were determined by the review of the assumed design basis and UFSAR response times and performance times documented by job performance measures (JPMs). For the selected components and operator actions, the team performed an assessment of the Emergency Operating Procedures (EOPs), Abnormal Operating Procedures (AOPs), Annunciator Response Procedures (APPs), and other operations procedures to determine the adequacy of the procedures and availability of equipment required to complete the actions. Operator actions were observed on the plant simulator and during plant walk downs.

The following operator actions were observed on the licensee's operator training simulator:

- Actions to align wetwell vent path after loss of suppression pool cooling per EOP procedure 3-EOI-Appendix 13, "Emergency Venting Primary Containment"
- Actions to align the diesel board for diesel "C" per 0-AOI-57-1A, "Loss of Offsite Power (161 and 500 KV) Station Blackout"
- Actions to inhibit an automatic depressurization system (ADS) during an Anticipated Transient Without Scram (ATWS) per 3-EOI-1, "Reactor Pressure Vessel Control"

- Actions to initiate emergency depressurization (ED) after loss of high pressure coolant injection (HPCI) per EOP procedure 3-EOI-C-1, “Alternate Level Control”
- Actions to lower and control reactor pressure vessel (RPV) level following an ATWS with isolated RPV per 3-EOI-C-5, “Level / Power Control”

b. Findings

No findings of significance were identified.

.4 Review of Industry Operating Experience

a. Inspection Scope

The team reviewed selected operating experience issues that had occurred at domestic and foreign nuclear facilities for applicability at the Browns Ferry Nuclear Plant. The issues that received a detailed review by the team included:

- BL 88-04, Potential Safety-Related Pump Loss
- IN 2006-029, Potential Common Cause Failure of Motor-operated Valves as a result of Stem Nut Wear
- RIS-06-023, Post-tornado Operability of Ventilating and Air-conditioning Systems Housed in Emergency Diesel Generator Rooms
- IN 2006-026, Failure of Magnesium Rotors in Motor-operated Valve Actuators
- IN 2000-21, Detached Check Valve Disk not detected by use of Acoustic and Magnetic Non-intrusive Test Techniques

b. Findings

No findings of significance were identified.

.5 Review of Permanent Plant Modifications

a. Inspection Scope

The team reviewed one risk-significant modification noted below to verify that the design bases, licensing bases, and performance capability of the component had not been degraded due to the modification. The adequacy of design and post-modification testing of the modification was reviewed by performing activities identified in IP 71111.17, “Evaluations of Changes, Test, or Experiments and Permanent Plant Modifications,” Section 02.02.a. Additionally, the team reviewed the modification in accordance with IP 71111.17 to verify that the licensee had appropriately evaluated it for 10 CFR 50.59 applicability.

- DCN 69118, Turbine First Stage Pressure Scram Bypass Setpoint Change

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Other

.1 Power Uprate (IP 71104)

The inspectors reviewed the following modification in accordance with IP 71104, Power Uprate. This is documented in Section 1R21.5 of this report.

- DCN 69118, Turbine First Stage Pressure Scram Bypass Setpoint Change

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 20, 2009, the team presented the inspection results to Mr. Rusty West, Browns Ferry Nuclear Plant Site Vice President, and other members of the licensee staff. The team returned all proprietary information examined to the licensee. No proprietary information is documented in the report.

On December 28, 2009, a telephone exit was conducted to disposition one item related to safety-related MCCBs as a URI (Section 1R21.2.7) to Mr. Jim Randich, Browns Ferry General Manager – Site Operations, and other members of the licensee staff.

4OA7 Licensee-Identified Violations (LIV)

The following violation of very low safety significance (Green) was identified by the licensee. This violation of NRC requirements met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a NCV.

- 10 CFR 50, Appendix B, Criterion III, “Design Control,” requires that that measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. During a design basis loss of coolant accident (LOCA) on Unit 1 (2) concurrent with a LOOP on all three units, given a single active failure affecting a Unit 1/2 EDG or 4160V shutdown board, there could be insufficient power for two RHR pumps to provide suppression pool cooling on Unit 2 (1). Suppression pool cooling on the non-LOCA units will be required to maintain suppression pool temperature within acceptable limits during RCIC and/or HPCI operation during a LOOP. Contrary to this, the licensee had inadequate design basis documentation (calculations) to support parallel EDG operations. This was identified in the licensee’s corrective action program as PER 178142. The inspectors assessed the finding using the SDP and determined that the finding was of very low safety significance (Green)

because of the low initiating event likelihood frequency of LOCA on one unit concurrent with a LOOP on all three units.

ATTACHMENTS: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Binkley, Operator Training
S. Bono, Site Engineering Director
S. Carter, System Engineer, Transformers
J. Davenport, Licensing Engineer
K. Dollar, Electrical Design Engineering
M. Epting, System Engineer, I&C
R. Godwin, Site Licensing and Industry Affairs Manager
G. Harrison, Electrical Design Engineering
K. Harvey, System Engineer, RHR service water
J. Jackson, Civil Design Engineering
E. Kirby, Balance of Plant Engineering
R. Knight, Operator Training
R. Krich, Vice President Nuclear Licensing, TVA Nuclear
J. McCarthy, BFN Director, Safety and Licensing
R. Moxley, Electrical / I&C System Engineer
J. Randich, General Manager – Site Operations

NRC

T. Ross, Senior Resident Inspector, Browns Ferry Nuclear Plant

LIST OF ITEMS OPENED, CLOSED, AND REVIEWED

Opened

05000259, 260, 296/2009008-01	NCV	Violation of Technical Specification 5.4.1 for Failure to Develop Adequate Procedures to Ensure Tornado Depressurization Protection of the Emergency Diesel Generators (Section 1R21.2.12)
05000259, 260, 296/2009008-02	NCV	Violation of 10CFR50, Appendix B, Criterion V for Inadequate Procedure for Emergency Equipment Cooling Water System Flow Balancing (Section 1R21.2.15)
05000259, 260, 296/2009008-01	URI	Safety-Related Molded Case Circuit Breakers (Section 1R21.2.7)

LIST OF DOCUMENTS REVIEWED

Design Basis Documents and System Descriptions

BFN-50-7023, Residual Heat Removal Service Water System, Rev. 14
BFN-50-7030B, Diesel Generator Building Environmental Control System, Rev. 8
BFN-50-7032, Control Air System, Rev. 12
BFN-50-7074, Residual Heat Removal System, Rev. 19
BFN-50-7082, Standby Diesel Generator, Rev. 14
BFN-50-7200, Auxiliary DC Power and Distribution System, Rev. 4
BFN-50-7200C, 250 V DC Power Distribution System, Rev. 7
BFN-50-7200E, 4KV AC Auxiliary Power System, Rev. 12
BFN-50-7200F, Plant Offsite Power Distribution System, Rev. 7

Procedures

0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) Station Blackout, Rev. 73
0-AOI-100-7, Severe Weather, Rev. 30
0-AOI-32-1, Loss of Control and Service Air Compressors, Rev. 0035
0-AOI-32-2, Loss Of Control Air, Rev. 0001
0-AOI-32A-1, Loss Of Drywell Control Air, Rev. 0007
0-OI-30F, Common and DG Building Ventilation, Rev. 28
0-OI-57A, Operating Instruction, Switchyard and 4160 V AC Electrical System, Section 8.3, Control Room Transfer of 4 kV Shutdown Board C Power Supplies; Section 8.7, Local Transfer of 4 kV Shutdown Board C Power Supplies, Rev. 139
0-OI-57D, DC Electrical System, Rev. 129
0-OI-67/ATT-1A,B,C, Valve Lineup Checklist Unit 3, Rev. 0083
0-OI-82, Standby Diesel Generator System, Rev. 0100
0-SI-3.1.15, Surveillance Instruction, Diesel Fuel Oil Transfer Pump Performance, Rev. 7
0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, Rev. 39
0-SR-3.8.1.9(A), Diesel Generator A Emergency Unit 1 Load Acceptance Test, Rev. 4
0-SR-3.8.1.9(C), Diesel Generator C Emergency Unit 1 Load Acceptance Test, Rev. 4
0-TI-54, EECW System Operational Flush, Rev. 0009
0-TI-230V, Vibration Program, Rev. 7
0-TI-346, MR Perf. Ind. Monitoring, Trending, and Reporting, Rev. 0034
0-TI-362, Inservice Testing of Pumps and Valves, Rev.0022
0-TI-395, Breaker Testing and Maintenance Program, Rev. 5
0-TI-443, Condition Monitoring of Check Valves, Rev. 5
0-TI-444, Augmented Inservice Testing Program, Rev. 3
0-TI-545, EECW System flow measurements and adjustments, Rev. 0001
1-ARP-9-3A, Alarm Response Procedure, Rev. 38
1-EOI Appendix 6B, Injection Subsystem Lineup RHR System I LPCI Mode, Rev. 0
1-SI-3.2.1, Inservice Testing and Augmented valve Performance test, Rev. 0011
1-SI-3.6.1.3.5(RHR I), RHR System MOV Operability Loop 1, Rev. 0003
1-SI-3.2.4(RHR I), EECW Check Valve Test On RHR Sys. Div. I, Rev. 0001
1-SI-4.2.B-54, CSCS Loop 1 Discharge Pressure Calibration, Rev. 3

1-SR-3.3.5.1.6(B I), Functional Testing of RHR Loop I Pump and Minimum Flow Valve Logic, Rev. 6
1-SIMI-74B, Residual Heat Removal System Scaling and Setpoint Documents, Rev. 20
1/2-ETU-SMI 1-C.4, Relay Functional Checks on 4kV Shutdown Board C, Rev. 24
2-ARP-9-3A, Alarm Response Procedure, Rev. 39
2-SI-3.2.4(RHR I), EECW Check Valve Test On RHR Sys. Div. I, Rev. 0001
3-AOI-1-1, Relief Valve Stuck Open, Rev. 9.
3-AOI-100-1, Reactor Scram, Rev. 49.
3-AOI-Appendix 5B, Injection Systems Lineup CRD, Rev 1.
3-ARP-9-3A, Alarm Response Procedure, Rev. 40
3-ARP-9-3E, RHR temperature and humidity monitoring, Rev. 0022
3-EOI-1, Reactor Pressure Vessel Control, Rev. 8.
3-EOI-2, Primary Containment Control, Rev. 7.
3-EOI-Appendix 3A, SLC Injection, Rev.1.
3-EOI-Appendix 4, Prevention of Injection, Rev. 4.
3-EOI-Appendix 5A, Injection System Lineup Condensate/Feedwater, Rev. 4.
3-EOI-APP 6B/6C/6D/6E, Injection Systems Lineup RHR Loop 1/11 Core Spray, Rev. 3.
3-EOI-Appendix 8A, Bypassing Group 1 RPV Lo-Lo-Lo Level Isolation Interlocks, Rev. 1.
3-EOI-Appendix 8E, Bypassing Group 6 to RPV Level and High Drywell Pressure Isolation Interlocks, Rev. 1.
3-EOI-Appendix 11A, Alternate RPV Pressure Control Systems MSRVs, Rev 2.
3-EOI-Appendix 13, Emergency Venting Primary Containment, Rev. 3.
3-EOI-Appendix 17A, Rev. 6.
3-EOI-Appendix 17C, RHR System Operation Suppression Chamber Sprays, Rev. 6.
3-EOI-C-1, Alternate Level Control, Rev. 9.
3-EOI-C-2, Emergency RPV Depressurization, Rev. 8.
3-EOI-C-5, Level / Power Control, Rev. 9
3-OI-68, Reactor Recirculation System, Rev. 76.
3-OI-74, Residual Heat Removal System, Rev. 88
3-OI-82, Standby Slow Start at Diesel Engine Control Cabinet, Rev. 89
3-PMT-BF-067.042, Chronological Flow test log, Rev. 0000
3-SI-3.2.4(RHR I), EECW Check Valve Test On RHR Sys. Div. I, Rev. 0001
3-SR-3.8.1.1(3A), Diesel Generator 3A Monthly Operability Test, Rev. 37
ARP-9-3C, Windows 18/31, ADS Logic bus A/B Inhibited, Rev. 25.
BFN-TSD-023-A, Test Scoping Document RHRSW, Rev. 0000
ECI-0-000-MOV001, Maintenance for limitorque MOVs, Rev. 0038
ECI-0-000-MOV009, Using MOVATS Universal Diagnostic System, Rev. 0021
EEB-TI-28, Branch Technical Instruction Setpoint Calculations, Rev. 7
EPI-0-000-MOT002, Oil Lubricated Motors, Rev. 58
EPI-0-254-BAT004, Annual Inspection of 125V DC Diesel Generator Battery Bank, Rev. 14
EPI-0-000-XMR001, Transformer and Gas Detector Relay Oil Sampling, Rev. 4
NETP-107, Medium Voltage Motor Testing and Maintenance Program, Rev. 1
OPDP-1, Conduct of Operations, Rev. 15
SII-1-XX-90-136, Main Steam Line Radiation Monitor Channel Alignment and Functional Test, Rev. 3
SII-1-XX-90-137, Main Steam Line Radiation Monitor Channel Alignment and Functional Test, Rev. 3

Completed Test Procedures

0-SR-3.3.8.1.1(C), Surveillance Procedure, 4 kV Shutdown Board C Degraded Voltage Relay Calibration and Functional Test, Rev. 1, performed on April 3, 2009
0-SR-3.3.8.1.2(C), Surveillance Procedure, 4 kV Shutdown Board C Undervoltage and Time Delay Calibration and Functional Test, Rev. 5, performed on March 24, 2009
0-SR-3.3.8.1.3(C), Surveillance Procedure, 4 kV Shutdown Board C Loss of Power (LOP) Logic System Functional Test, Rev. 8, performed on November 9, 2008
0-SR-3.8.1.1(A), Diesel Generator A Monthly Operability Test, performed on November 15, 2008; February 9, 2009; May 30, 2009; and August 24, 2009
0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 kV Shutdown Boards, Rev. 9, performed on August 24, 2009
0-TI-492, ASME Pump Curve Data Acquisition, performed on October 18, 2006
1-SR-3.3.1.1.3(INBD), Inboard MSIV Limit Switch Calibration and Slow Speed Adjustment, performed on November 14, 2008
1-SI-3.2.12, Verification of Fail-Safe Position for MSIVs, performed on November 27, 2008
1-SI-3.2.1, Inservice Testing and Augmented Inservice Testing Valve Performance, (1-SI-4.5.C.1(3) for 1-CKV-23-510) performed on January 9, 2009; June 29, 2009; and September 19, 2009
1-SI-3.2.1, Inservice Testing and Augmented Inservice Testing Valve Performance, (1-SR-3.6.1.3.5(RHR I) for 1-FCV-74-57 and -590, performed on March 8, 2009; May 14, 2009; and August 6, 2009
1-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test, performed on September 22, 2009
2-SR-3.5.1.6(RHR II-COMP), RHR Loop II Comprehensive Pump Test, performed on May 27, 2009

Drawings

0-0223R0399, Electrical Schematic Diagram, 4kV Shutdown Board, Rev. 2
0-0223R0428, Electrical Schematic Diagram, 4kV Shutdown Board, Rev. 3
0-0223R0432, Electrical Schematic Diagram, 4kV Shutdown Board, Rev. 4
0-15E500-1, Key Diagram of Standby Auxiliary Power System, Rev. 32
0-45B299, Sheet 1, 4000 Volt Motors, Motor Data, Rev.0
0-45E701-1, Battery Board 1, Panels 1-7 Single Line, Rev. 57
0-45E724-3, 4160V Shutdown Board C Single Line, Rev. 31
0-45E765-1, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 56
0-45E765-3, Schematic Diagram, 4160V Shutdown Auxiliary Power, Rev. 24
0-45E765-10, Schematic Diagram, 4160V Shutdown Auxiliary Power, Rev. 8
0-45E765-11, Schematic Diagram, 4160V Shutdown Auxiliary Power, Rev. 39
0-45E767-1, Electrical Schematic Diagram, Diesel Generators, Rev. 21
0-45E771-1, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 10
0-45E771-3, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 33
0-45E767-3, Electrical Schematic Diagram, Diesel Generators, Rev. 15
0-45E767-4, Electrical Schematic Diagram, Diesel Generators, Rev. 9
0-45E765-11, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 39
0-45E767-2, Electrical Schematic Diagram, Diesel Generators, Rev. 13
0-47E-610-67-2, Mechanical Control Diagram EECW, dated July 10, 1969

0-47E840-1, Unit 0 Flow Diagram Fuel Oil System, Rev. 12
0-47E840-2, Unit 0 Flow Diagram Fuel Oil System, Rev. 13
0-47E-858-1, Flow Diagrams RHR Service Water, dated April 8, 1969
0-47E865-8, Mechanical Flow Diagram, Diesel Generator Building, Rev. 12
0-47E-1847-1, Mechanical Control Diagram Contol Air, dated April 17, 1985
0-47W452-5, Units 1-3, Mechanical Residual Heat Removal System, Rev. 8
0-728E998-1-1, Electrical Elementary Diagram, Process Radiation Monitoring System, Rev. 2
0-731E761, Emergency Equipment, Sheet 24, Rev. 12
0-731E761-1, Electrical Schematic Diagram, Emergency Equipment, Rev. 17
0-731E761-2, Electrical Schematic Diagram, Diesel Generators, Rev. 8
0-731E761-11, Elementary Diagram, Emergency Equipment, Rev. 24
0-J8073C1, Diesel Fuel Storage Tank, Rev. 1
14121, EECW Self Cleaning Strainer Drawing, dated March 14, 2009
1-45E620-3, Electrical Schematic Diagram, Annunciator System, Rev. 17
1-45E765-4, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 18
1-45E779-13, Electrical Schematic Diagram, 480V Shutdown Aux Power, Rev. 13
1-47E801-1, Unit 1, Mechanical Flow Diagram Main Steam, Rev. 20
1-47E801-2, Unit 1, Mechanical Flow Diagram Main Steam, Rev. 16
1-47E811-1, Unit 1 Flow Diagram Residual Heat Removal System, Rev. 34
1-47E858-1, Units 1&0, RHR Service Water System, Rev. 63
1-47E859-1, EECW Flow Diagram, dated April 18, 1969
1-47W450-1, Reactor Building Unit 1, Mechanical RHR Service Water Piping, Rev. 9
1-47W451-4, Mechanical EECW, dated August 21, 1970
1-729E814-4, Electrical Elementary Diagram, Process Radiation Monitoring System, Rev. 17
1-730E915-9, Electrical Elementary Diagram, Reactor Protection System, Rev. 20
1-730E915-17, Electrical Elementary Diagram, Reactor Protection System, Rev. 8
2-45E620-3, Electrical Schematic Diagram, Annunicator System, Rev. 34
2-47E610-74-2, Unit 2 Mechanical Control Diagram, Residual Heat Removal System, Rev. 033
2-47E811–1, Unit 2 Flow Diagram Residual Heat Removal System, Rev. 66
2-47W452-6, Unit 2 Mechanical Residual Heat Removal Sys, Rev. 13
2-729E814-4, Electrical Elementary Diagram, Process Radiation Monitoring System, Rev, 22
2-730E915-9, Electrical Elementary Diagram, Reactor Protection System, Rev. 24
2-730E915-17, Electrical Elementary Diagram, Reactor Protection System, Rev. 15
3-15E500-3, Electrical Schematic Diagram, Normal & Standby Aux Power System, Rev. 51
3-45E766-10, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 13
3-45E766-12, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 12
3-45E766-14, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 14
3-45E766-16, Electrical Schematic Diagram, 4160V Shutdown Aux Power, Rev. 13
3-45E767-5, Electrical Schematic Diagram, Diesel Generators, Rev. 20
3-45E767-6, Electrical Schematic Diagram, Diesel Generators, Rev. 15
3-45E767-7, Electrical Schematic Diagram, Diesel Generators, Rev. 19
3-45E767-8, Electrical Schematic Diagram, Diesel Generators, Rev. 13
3-45E768-1, Electrical Schematic Diagram, Diesel Generators, Rev. 21
3-45E771-1, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 19
3-45E771-8, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 19
3-45E771-9, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 27
3-45E771-10, Electrical Schematic Diagram, 480V Diesel Aux Power, Rev. 9
3-47E811-1, Mechanical Flow Diagram, RHR System, Rev. 64

3-47E861-4A, Unit 3 Flow Diagram Diesel Starting Air System Gen 3D, Rev. 9
3-47E861-8, Unit 3 Flow Diagram Cooling System & Lubricating Oil System Standby Diesel Generator 3D, Rev. 12
3-47E865-8, Mechanical Flow Diagram, Diesel Generator Building, Rev. 15
3IN206, Concrete Pumping Station Outline, Sheet 4, Rev. 7 [Section M4-M4 detail]
D-53467, Aloyco Globe Valves, dated June 28, 1978
General Electric vendor drawing 0-0223R0428, Sheet 1, 4 KV Shutdown Board C Elementary Diagram, Rev. 2
General Electric vendor drawing 0-0223R0428, Sheet 2, 4 KV Shutdown Board C Elementary Diagram, Rev. 10
General Electric vendor drawing 0-0223R0428, Sheet 7, 4 KV Shutdown Board C Power and Control Circuits, Breaker 1718, Rev. 3
General Electric vendor drawing 0-0223R0428, Sheet 15, 4 KV Shutdown Board C Power and Control Circuits, Breaker 1624, Rev. 3
W9424202, 2-12" Anchor Darling Flex Wedge Valve, dated March 18, 1994
W9424221, 3" Anchor Darling Stainless Steel Glove Valve, dated May 18, 1994

Calculations

CDQ0999910005, Seismic Verification Walkdown, Attachment 2, 5/31/88 and Attachment 3, 11/30/88
ED-N0245-920598, Setpoint and Scaling Calculation for 0-RLY-245-000A/4, 0-83-245-000A/1, 0-RLY-245-0008/4, and 0-83-245-000B/1, Rev. 0
ED-N2090-890236, Setpoint and Scaling Calculation for the Main Steam Line Radiation Monitors, Rev. 7
ED-N3090-930131, Setpoint and Scaling Calculation for 1/3-R-90-136-, 137, Rev. 6
ED-Q0057-920034, 4.16kV and 480V Busload and Voltage Drop, Rev. 51
ED-Q0057-20020069, Diesel Load Study for Unit 1 and 2, Rev. 14
ED-Q0074-9600009, 1/2/3 RS-0050, -0064 RHR Mini Flow Bypass Setpoint and Scaling Calculation, Rev. 6
ED-Q0999-970004, DC Ground Detection Analysis, Rev. 0
ED-Q2000-870548, Cable and Bus Protection / Breaker Coordination for 4 kV Switchgear and 480 V Load Center, Rev. 31
ED-Q2000-870550, Cable and Bus Protection/Breaker/Fuse Coordination for 250 VDC System, Revs. 24 and 25
ED-Q2999-920229, Mini-Calculation for Protection and Coordination for 4kV Switchgear and 480V Load Centers, Rev. 29
ED-Q3057-920035, Diesel Load Study, Rev. 37
EDN0074920206, 1/2/3-F-74-50 & 1/2/3-F-74-64 Setpoint and Scaling Calculation, Rev. 17
EDN0245920372, Setpoint and Scaling Calculation for Load Tapchangers Controls for Common Station Service Transformers A and B, Rev. 3
EDQ1-999-2002-0075, Thermal Overload Sizing, Rev. 8
EDQ024820020042, 250 VDC Unit Battery Load Study, Voltage Drop, Short Circuit, and Battery Capacity for LOCA/LOOP, Station Blackout, and Appendix R Analysis for Unit / Shutdown Board Batteries, Rev. 24
EDQ2074980125, 2-PS-74-8A, 8B, 19A, 19B, 31A, 31B, 42A, 42B Setpoint and Scaling Calculation, Rev. 4

EDQ0057920034, 4.16 KV and 480 V Busload, Voltage Drop, and Short Circuit Calculation, Rev.51
EDQ005720040026, AC and DC Load Limitations for Units 1, 2, and 3 Operating, Rev. 13
GE-NE-187-06-0890, Analysis to Demonstrate Safe Shutdown with Two Shutdown Boards at Browns Ferry Plant, September 1990 (GE Proprietary)
MD-0018-870164, Diesel Fuel Oil Consumption/Delivery Rate and 7-Day Tank Sizing, Rev. 5
MD-Q0074-870155, Total RHR System Head VS. Flow Rate For Priority 1 Mode Support (SRC MODES 74-03-M-S), Rev. 6
MD-Q0074-880225, Total RHR System Head VS. Flow Rate For Priority 1 Mode Support (SRC MODES 74-01-M-S, 74-02-M-S, 74-04-M-L), Rev. 7
MD-Q0074-960020, Analytical Limits for RHR Minimum Flow Bypass, Rev. 3
MD-Q3074-920435, MOV 3-FCV-74-59, Operator Requirements and Capabilities, Rev. 7
MDN000120060006, Main Steam Line Pressure Drop Hydraulic Model for Extended Power Uprate, Rev. 0
MDQ099920060011, Transient NPSH/Containment Pressure Evaluation of RHR and Core Spray Pumps, Rev. 5
MDQ0999910034, Generic letter 89-10 calculation evaluation, 05/02/2002
MDQ1-074-2002-0068, MOV 1-FCV-074-007 & 1FCV-074-0030, Operator Requirements & Capabilities, Rev. 2
MDQ1-074-2002-0078, MOV 1-FCV-074-0059 & 1-FCV-074-0073, Operator Requirements & Capabilities, Rev. 1
MDQ2-074-910107, MOV 2-FCV-74-07, Operator Requirements and Capabilities, Rev. 7
MDQ2074910107, Operator Requirements and Capabilities, Rev. 0008
MCQ3-074-920433, MOV 3-FCV-74-57, Operator Requirements and Capabilities, Rev. 7
MDQ3074920421, Operator Requirements and Capabilities, Rev. 0006
MDN1-001-2003-0040, Main Steam Pressure Drop and Flow Analysis for BFNP Unit 1 with Modified MSIVs, Rev. 0
MDQ1-074-2002-0068, Operator Requirements and Capabilities, Rev. 0002
NDQ0074880118, Evaluation of LPCI Flow to Reactor Pressure Vessel (RPV) with Failed Open Min-Flow Bypass Valve, Rev. 4
Report 431934, Valve Thrust Calculations, by Powell Valves, 12/22/87
RIMS E31 881109 160, Common Station Service Transformer A Neutral Ground Relay, Rev 0

<u>Corrective Action Documents</u>

042933, Drawing discrepancy drawing shows a 1' pipe instead of 2' pipe, dtd. 01/17/2003

PER 96894, Operating Experience, Evaluation of SC06-01, 10CFR Part 21 Communication
PER 980124, Operating Experience, Evaluation of NRC IN 97-90, Non-conservative Acceptance Criteria in Safety-Related Surveillance Tests
PER 108271, Operating Experience, Stem Nut Wear Information Notice Evaluation, IN 2006-09
PER 116511, RHRSW Inlet Check Valves Found Open
PER 121622, Blown fuse in control circuit, 3/07
PER 124791, EDG 24 Hour Operability Run, Turbo-Charger oil leak
PER 131090, the megger test indicated that a ground condition exists, dtd. 10/01/2007
PER 134271, WO provided incorrect instructions, 11/07
PER 135132, Ineffective corrective action for PER 119954 [drain in JB 4915 not implemented]
PER 149478, Alarm for HI Radiation, 8/08

PER 154539, Unplanned LCO Entry due to EECW strainer drain valve not operating, 10/14/2008
PER 155071, EECW Strainer Not Automatically Rotating, dtd. 10/21/2008
PER 156416, Unsatisfactory relay functional checks on 4 kV Shutdown Board C, 11/5/08
PER 156416, Apparent Cause Evaluation Report, 11/5/08
PER 157895, Spurious alarms for Unit 3 radiation monitors, 11/08
PER 159564, Diesel output frequency fluctuations, 12/08
PER 160106, EECW strainer has experienced numerous problems/failures, 01/02/2009
PER 161042, discrepancies btwn. valve "required position" and their metal tag color, dtd. 01/16/2009
PER 167689, Lighting in U1/2 DGB Elevation 565
PER 171370, Twig found wedged in check valve in EECW supply line, 05/18/2009
PER 172105, Identified the bent stem for 2-FCV-074-0007, RHR system, dtd. 06/08/2009
PER 176227, Electrical isolation between safety-related and quality related circuits, 8/09
PER 177750, EECW strainer High pressure light, dtd. 08/04/2009
PER 178142, DG Paralleling Design Basis

Work Orders

02-013606-000, PM to perform a functional check of the protective relaying associated with CSST A per 0-ETU-RMI-1-161CSSTA, completed June 16, 2005
02-013612-000, PM to perform transformer electrical testing on common station service transformer A, completed June 23, 2005
02-013613-000, Perform inspection of the load tap changer on common station service transformer A, completed June 17, 2005
04-7126-024, Old valve tested and new valve installed, dated February 23, 2005
04-724246-000, Main Steam Line Rad Channel A Detector, 6/05
04-724728-000, Main Steam Line Rad Channel C Detector, 6/05
05-710954-000, Perform PM on breaker and cubicle for breaker 1814, per EPI-0-000-BKR015, completed March 22, 2007
05-711104-000, PM to perform relay calibration tests of the protective relaying associated with common station service transformer A per 0-ETU-RMI 3-161CSSTA, completed June 23, 2005
05-718682-000, Main Steam Line Rad Channel A Detector, 10/05
05-718683-000, Main Steam Line Rad Channel C Detector, 10/05
05-722716-000, Main Steam Line Rad Channel A Detector, 5/07
05-722717-000, Main Steam Line Rad Channel C Detector, 5/07
05-725445-001, Adjust packing, stop leak at pipe plug on 1-SHV -067-0565 , dated January 27, 2006
06-717286-000, Perform PM on breaker and cubicle for breaker 1722, per EPI-0-000-BKR015, completed April 4, 2007
06-717719-000, 125V DC Diesel System Battery B, 8/06
06-723117-000, EECW Strainer Inspection/Evlauation, dated January 26, 2007
06-726651-000, Submergence of medium voltage cable in conduits, completed January 16, 2008
06-727286-000, Main Steam Line Rad Channel C Detector, 9/07
06-727287-000, Main Steam Line Rad Channel A Detector, 9/07
07-711195-000A, PM to perform relay calibration and functional tests on CSST A tap changer relays per 0-ETU-RMI-1-CSSTALTC, completed September 18, 2007

07-711795-000, Inspection and preventive maintenance on 250 V Battery Board 1, breaker 201, per EPI-0-000-BKR003 and EPI-0-000-BKR020, completed June 21, 2007
07-717306-000, 125V DC Diesel System Battery 3A, 8/07
07-717376-000, Main Steam Line Rad Channel A Detector, 8/07
07-717377-000, Main Steam Line Rad Channel C Detector, 8/07
07-719387-000, 125V DC Diesel System Battery B, 1/08
07-720308-000, Perform inspection and preventive maintenance on 250 V Battery Board 1, Breaker 501, per EPI-0-000-BKR003 and –BKR020, completed August 8, 2008
07-722044-000 to investigate and correct low flow condition, dated August 8, 2007
07-725807-000, EECW Strainer Inspection/Evlauation, dated May 21, 2008
07-726178-000, Inspection and PM on 4 kV Shutdown Board C, completed December 3, 2008
08-712781-000, 125V DC Diesel System Battery 3A, 7/08
08-717905-000, PM to perform a functional check of the protective relaying circuitry associated with 4 kV Shutdown Board C, completed November 5, 2008
08-718382-000. Perform PM on breaker and cubicle for breaker 1624, per EPI-0-000-BKR015, completed October 31, 2008
08-718406-000, Perform PM on breaker and cubicle for breaker serving TS1B, per EPI-0-000-BKR015, completed November 1, 2008
08-718407-000, Perform PM on breaker and cubicle for breaker 1718, per EPI-0-000-BKR015, completed November 2, 2008
08-721675-000, Main Steam Line Rad Channel A Detector, 7/09
08-721676-000, Main Steam Line Rad Channel C Detector, 7/09
08-721907-000, EECW Strainer Inspection/Evlauation, dated March 23, 2009
08-722600-000, 125V DC Diesel System Battery B, 5/09
09-711887-000, 125V DC Diesel System Battery 3A, 7/09
09-714822-000, EECW Strainer Inspection/Evlauation, dated October 9, 2009

Design Change Packages

DCN 69118, Adjust Turbine Power/Scram Bypass Setpoint to Comply with Tech Specs, Rev. A

Training Procedures

JPM 155 C, Manually Start C Unit 3 D/G Locally
JPM 355, Rev. 1, 3-EOI Appwndix-13, Emergency Venting Primary Containment. Rev. 1
JPM CDBI-1, Remove ADS fuses in the Aux Instrument Room. *(Based on ARP-9-3C, Windows 18/31, ADS Logic bus A/B Inhibited, Rev 25.)*
OPL171.052, Fuel Pool Cooling and Cleanup System, Rev. 10.
OPL171.042, High Pressure Coolant Injection (HPCI), Rev. 19.
Scenario CDBI-1
Scenario CDBI-2

Operator Work-Arounds

0-082-OWA-2009-0095, Starting Two RHR Pumps on One Unit 1/2 4KV Shutdown Board
0-057-OWA-2009-0113, Implementation of the Safe Shutdown Instructions during an Appendix R Event.

Miscellaneous

3R-090-0136-00-01, Setpoint and Scaling Document, Rev. 1
3R-090-0137-00-01, Setpoint and Scaling Document, Rev. 1
2R-090-0136-00-02, Setpoint and Scaling Document, Rev. 2
2R-090-0137-00-02, Setpoint and Scaling Document, Rev. 2
4299128DOC01, Motor Performance Software, Rev. 2
4299128DOC05, Test Results Summary, Rev. 5
8937R1, RHR Relay Settings, 1/81
Anchor Darling Vendor Man for 3-SHV-67-565 Model
ASME OM CODE-1996 ADDENDA to ASME OM CODE-1995 Code for Operation & Maintenance of Nuclear Power Plants
BFN-VTM-P318-0010, Emergency Diesel Units 1, 2, and 3, Rev. 39
BFN-VTD-A585-0070, Instruction Manual, Atwood & Morrill Free Flow Reverse Current Valves, Rev. 3
BFN-VTD-G080-0735, Instructions, General Electric Gas Detector Relay for Atmoseal Type Transformers, Rev. 0
BFN-VTD-G080-1021, Instruction Manual for General Electric Power/Vac Metal-Clad Switchgear Types 4.16 and 13.8 for Power-Vac Vacuum Circuit Breaker Types VB-4.16, VB-7.2, VB 13.8, VB1-4.16 and VB1-13.8, Rev. 0
BFN-VTD-G080-4020, Instructions for GE Vertical Induction Motors “P” Base Frames, Rev. 2
BFN-VTD-G080-4010, Vendor Technical Manual for GE Vertical Induction Motors, Rev. 5
BFN-VTD-P305-0130, Vendor Man for Powell for hand operated valves, Rev. 0002
DES79050507, TVA Memo, Subject: Nuclear Plant Switchyard Design – Controls for Transformer Cooling, 5/1/79
Doble Engineering Test Results, Common Station Service Transformer A, performed 10/2/2009
Evaluation of Containment Analyses for PER 96894 – GE SC06-01
Letter from TVA to NRC, Browns Ferry (BFN), Watts Bar (WBN), and Bellefonte (BLN) Nuclear Plants – NRC BULLETIN (NRCB) 88-04, “Potential Safety-Related Pump Loss”, dated September 30, 1988
MOT-001, Qualification Maintenance Data Sheets, Rev. 20
R94.033 Design & Seismic Analysis, Rev. 0
Screening Evaluation Work Sheet, SSEL Line No. 9137, 250 V Main Battery 1, 6/30/95
Seismic Verification Walkdown Generic Implementation Procedure, Sections 4.3, 4.5, Rev. 2A
S.P. Kinney AV Series motorized auto self-cleaning strainer Vendor Manual, dated March 19, 2002
Transformer Insulating Oil Analysis, Dissolved Gas History, Common Station Service Transformer A, Sample Dates from 8/8/2006 through 8/20/09
Transformer Insulating Oil Analysis, Dissolved Gas History, Common Station Service Transformer B, Sample Dates from 6/222006 through 9/8/09
Transformer Health Detail, Dissolved Gas Analyses, Common Station Service Transformer A, sample dates 9/15/08, 12/8/08, 3/2/09, 5/26/09, 8/20/09
Transformer Health Detail, Dissolved Gas Analyses, Common Station Service Transformer B, sample dates 7/14/08, 10/9/08, 3/24/09, 6/15/09, 9/8/09
Transformer Health Detail, Oil Quality Analyses, Common Station Service Transformer A, sample dates 2/6/07, 4/25/07, 7/24/07, 10/22/07, 10/23/08
Transformer Health Detail, Oil Quality Analyses, Common Station Service Transformer B, sample dates 11/22/05, 11/21/06, 3/2/07, 10/22/07, 10/23/08

Corrective Action documents initiated due to CDBI activity:

PER 205553, Transformer 1-XFA-099-0010 is humming loudly and appears to be hot
PER 205569, Door 459 to MG set room has dirty air filter
PER 205714, During a NRC CDBI walkdown, a piece of tape was identified inside the seal/shaft area of the pump, which is a C-zone, on the pump seal for the 1B RHR pump
PER 205717, Corrosion between cells and on one of the bolted connections of 250 VDC battery bank
PER 205987, Appears to be no consistency when seismic spacers are installed between battery cells
PER 206032, Bend radius acceptability for inter-cell connectors for 250V Battery bank and insulation split
PER 206896, Ladders hung on storage rack behind 4KV Shutdown Board A
PER 206919, Tornado depressurization protection of the diesel generator building ventilation system
PER 207096, An FME cover over a removed duct segment was being used as a hamper
PER 207239, During walkdown, NRC inspector found door to 1-PLNA-009-0015 open in Unit 1 Aux Equip. Rm.
PER 207295, During a CDBI walkdown, an NRC inspector identified several strips of orange tape covering a connection in the wall behind panel 1-LPNL-925-0062 (located U1 Rx Bldg El. 519' SW Quad)
PER 207297, During a CDBI walkdown, an NRC inspector noted that the approximately 3 inch flex conduit coming from the motor termination box on the 1B RHR pump motor has a bulge at the base as it transitions to the stationary metal conduit.
PER 207305, Battery cart that is unattended and unrestrained located in the Unit 1 250V Battery Room (1C Hallway El. 593' Control Bay)
PER 207308, As part of the CDBI inspection a walkdown of the East side of the plant was conducted on 11/05/09. HH-15 was inspected and debris was found to be interfering with the sump pump operation (causing to run continuously)
PER 207309, Temporary fans in Unit 1 Aux Instrument Room did not comply with 0-TI-471
PER 207899, Indicating lights not illuminated
PER 207913, Empac fields for RHR Room Cooler Supply Valve (all 3 units) 1, 2, 3,-067-565 were not verified in Empac which resulted in delays for finding the vendor information
PER 207959, During the CDBI, it was discovered that a statement in Elec. design calculation EDQ0-057-2004-0026 contradicts with FSAR Section 8.5.2(7)
PER 208134, Conduit interferes with line of sight between B 4KV shutdown board compartment 4 and synchronizing panel
PER 208142, Inadequate evaluation of RIS 2006-23
PER 208285, Non-conservative calculations associated with RHR minimum flow
PER 208286, Temporary power supply in transformer yard has been left in place indefinitely
PER 208374, During the 2009 CDBI inspection, it was found that the inlet throttle valves to the 2A (2-THV-067-0551) and 2B (2-THV-067-0594) CS Room coolers are possibly throttled down so much that the clearance is less than the size of the EECW strainers
PER 208636, During questions asked by the NRC CDBI inspection team, it was discovered that the perforated cones in the EECW strainers had areas where two of the holes met at a seam resulting in a hole larger than 1/8-inch

PER 208853, During the 2009 NRC Component Design Bases Inspection (CDBI), an inspector requested photos of hand-holes 15 and 26 (HH-15 & HH-26). Upon viewing the photos, the inspector inquired about the presence of silicon sealant in the end of one of the conduits entering HH-26. The silicon sealant should be removed from conduits in HH-26.
PER 208854, Adverse trend on Hand Hole 15 for standing water and debris
PER 208780, During the pre-exit meeting for the NRC CDBI, it was noted that a complex operator work around existed for paralleling two diesels (0-082-OWA-2009-0095) and no training was conducted in support of this OWA.
PER 208978, Heat trace indicating light in B RHRSW pump room illuminated
PER 209016, Improvements needed to DG fuel oil transfer pump methods
PER 209095, Safety-related molded case circuit breakers for 250 VDC Battery Board 1 (breakers 110, 607, 705, 712, 715) do not have PMs to test them