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

October 29, 2008

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2008004, 05000260/2008004 AND 05000296/2008004

Dear Mr. Campbell:

On September 30, 2008, 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 results which were discussed, on October 3, 2008, with Mr. Rusty West and other members of your staff.

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

In addition to the routine Reactor Oversight Process baseline inspections for all three units, the inspectors continued to conduct augmented inspections on Unit 1 as mentioned in NRC letters dated May 16, 2007, December 6, 2007 and May 21, 2008. These Unit 1 augmented inspections were conducted to compensate for the lack of valid data for certain Performance Indicators (PI). These additional inspections were only considered to be an interim substitute for the invalid Unit 1 PIs until complete and accurate PI data was developed and declared valid. Since Unit 1 startup on May 22, 2007, the PI's in the Initiating Events and Barrier Integrity cornerstones, and the Safety System Functional Failure PI of the Mitigating Systems cornerstone, have become valid as acknowledged by the Tennessee Valley Authority letters dated January 7, 2008 and July 11, 2008. Consequently, the only PIs that remain invalid, and thereby subject to the augmented baseline inspection, are the Mitigating Systems Performance Index PIs.

Based on the results of this inspection, no findings of significance were identified.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the 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/

Eugene F. Guthrie, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000259/2008004, 05000260/2008004 and 05000296/2008004
w/Attachment: Supplemental Information

cc w/encl.: (See page 3)

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the 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/

Eugene F. Guthrie, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000259/2008004, 05000260/2008004 and 05000296/2008004
w/Attachment: Supplemental Information

cc w/encl.: (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: ☐ Yes ACCESSION NUMBER:_______________________

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRP
SIGNATURE	JBB /RA/	KJK by email	TMR by email	CRS by email	EFG /RA/	LWG /RA/
NAME	JBaptist	KKorth	TRoss	CStancil	EGuthrie	LGarner
DATE	10/29/2008	10/28/2008	10/28/2008	10/29/2008	10/29/2008	10/28/2008
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB6\BROWNS FERRY\REPORTS\2008\BF 08-04\BF 08-04.DOC

cc w/encl:
Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development and Construction
Tennessee Valley Authority
Electronic Mail Distribution

William R. Campbell, Jr.
Chief Nuclear Officer and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Donald E. Jernigan
Senior Vice President
Nuclear Operations
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Thomas Coutu
Vice President
Nuclear Support
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

R. G. (Rusty) West
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

D. Tony Langley
Manager
Licensing and Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

John C. Fornicola
General Manager
Nuclear Assurance
Tennessee Valley Authority
Electronic Mail Distribution

General Counsel
Acting Licensing Manager
Tennessee Valley Authority
Electronic Mail Distribution

Larry E. Nicholson
General Manager
Performance Improvement
Tennessee Valley Authority
Electronic Mail Distribution

Michael A. Purcell
Senior Licensing Manager
Nuclear Power Group
Tennessee Valley Authority
Electronic Mail Distribution

Michael J. Lorek
Vice President
Nuclear Engineering and Projects
Tennessee Valley Authority
Electronic Mail Distribution

Beth A. Wetzel
Manager
Corporate Nuclear Licensing and Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

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

Dr. D. E. Williamson
State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, AL 36130-3017

Kirksey E. Whatley
Director
Office of Radiation Control
Alabama Dept. of Public Health
Electronic Mail Distribution

(cc w/encl: see next page)

cc w/encl: con't
Steven M. Douglas
General Manager
Browns Ferry Site Operations
Electronic Mail Distribution

Letter to William R. Campbell, Jr. from Eugene F. Guthrie dated October 29, 2008

SUBJECT: BROWNS FERRY NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000259/2008004, 05000260/2008004 AND 05000296/2008004

Distribution w/encl.:
E. Brown, NRR
L. Raghavan, NRR
C. Evans, RII
L. Slack, RII
OE Mail
RIDSNRRDIRS
PUBLIC

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-259, 50-260, 50-296
License Nos.:	DPR-33, DPR-52, DPR-68
Report Nos.:	05000259/2008004, 05000260/2008004 and 05000296/2008004
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:	July 1, 2008 through September 30, 2008
Inspectors:	T. Ross, Senior Resident Inspector C. Stancil, Resident Inspector K. Korth, Resident Inspector
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000259/2008-004, 05000260/2008-004 and 05000296/2008-004; 07/01/2008 - 09/30/2008; Browns Ferry Nuclear Plant, Units 1, 2 and 3; routine integrated report.

The report covered a three-month period of routine inspections by the resident inspectors. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, Reactor Oversight Process, Revision 4, dated December 2006.

A. <u>NRC-Identified and Self-Revealing Findings</u>

No findings of significance were identified.

B. <u>Licensee-Identified Violations</u>

A 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 the corrective action program tracking number are described in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at essentially 100 percent rated thermal power (RTP) the entire report period except for two significant downpowers and a shutdown. On August 2, 2008, a planned downpower to 70 percent RTP was conducted on Unit 1 for control rod sequence exchange and to repair secondary plant steam leaks. The unit was returned to 100 percent RTP the next day. On August 7, 2008, an unplanned down power to 50 percent RTP was conducted to maintain river discharge temperatures within environmental limits following a total loss of power to all cooling towers. On August 8, 2008, an unplanned shutdown from 50 percent RTP was conducted following discovery of a Code Class 2 piping leak on the 1A Main Steam Line that could not be isolated at power. The unit was restarted on August 13, 2008, and was restored to 100 percent RTP on August 15, 2008.

Unit 2 operated at essentially 100 percent RTP the entire report period except for two significant downpowers and a shutdown. On August 7, 2008, an unplanned down power to 50 percent RTP was conducted due to elevated river discharge temperatures (see above). Unit power was raised to 75 percent RTP the next day and to 100 percent RTP on August 9, 2008. On September 16, 2008, a planned down power to 22 percent RTP was conducted to repair a hydrogen leak on the Unit 2 main generator. An unplanned shutdown of the unit was conducted the following day to affect repairs to a containment isolation valve and a rod position indicator. The unit was restarted on September 21 and reached 100 percent RTP on September 24, 2008.

Unit 3 operated at essentially 100 percent RTP the entire report period except for three significant downpowers and a shutdown. On July 11, 2008, a planned shutdown was conducted on Unit 3 to repair a steam leak from a manual isolation valve (Main Steam to the 3B Off-Gas pre-heater), which could not be isolated at power. The unit was restarted on July 12 and returned to 100 percent RTP on July 14, 2008. On August 7, 2008, an unplanned down power to 50 percent RTP was conducted due to elevated river discharge temperatures (see above). Unit power was raised to 75 percent RTP the next day and to 100 percent RTP on August 11, 2008. On August 14, 2008, a planned downpower to 90 percent RTP was conducted on Unit 3 to remove a Condenser Cooling Water Pump from service to repair the associated travelling screen. The unit was then returned to 100 percent RTP later in the day. Lastly, on August 14, 2008, an unplanned downpower to 82 percent RTP was conducted to remove the 3C Reactor Feedwater Pump from service to repair a feed water leak on the suction pressure instrument line which could not be isolated with the pump running. Power was restored to 100 percent RTP on August 17, 2008.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

External Flood Protection Measures

a. Inspection Scope

The inspectors performed one external flood protection measures review. The inspectors reviewed plant design features and licensee procedures intended to protect the plant and its safety-related equipment from external flooding events. The inspectors reviewed flood analysis documents including applicable portions of the Updated Final Safety Analysis Report (UFSAR) and BFN-50-C-7101, Design Criteria for Protection from Wind, Tornado Wind, Tornado Depressurization, Tornado Generated Missiles, and External Flooding. The inspectors performed walkdowns of risk-significant areas and susceptible systems and equipment, including the residual heat removal service water (RHRSW) and emergency equipment cooling water (EECW) pump rooms, and Unit 1/2 and Unit 3 emergency diesel generator (EDG) rooms. The inspectors' review included flood-significant features such as sump pump flowrates, sump drains, door seals and the Reactor Building Flood Gate. Plant procedures and calculations for coping with flooding events were also reviewed to verify that licensee actions and maintenance practices were consistent with the plant's design basis assumptions.

The inspectors also reviewed licensee corrective action documents for flood-related issues identified by licensee Problem Evaluation Reports (PERs) written in the previous year to verify the adequacy of the corrective actions. The inspectors reviewed selected completed preventive maintenance procedures and work orders (WOs) for identified level switches, pumps and flood barriers (e.g., Flood Doors) for completeness and frequency.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted three equipment partial alignment walkdowns to evaluate the operability of selected redundant trains or backup systems (as listed below) while the other train or system was inoperable or out of service. The inspectors reviewed the functional systems descriptions, UFSAR, system operating procedures, and Technical Specifications (TS) to determine correct system lineups for the current plant conditions. The inspectors performed walkdowns of the systems to verify that critical components were properly aligned and to identify any discrepancies which could affect operability of the redundant train or backup system. Documents reviewed are listed in the Attachment to this report.

- Unit 1 Core Spray (CS), Division II, per 1-OI-75, CS System and Attachments 1, 2 and 3
- Unit 3 EDG 3C, per 3-OI-82, Standby Diesel Generator System and Attachments 1C, 2C, 3 and 3C
- Unit 2 Reactor Core Isolation Cooling (RCIC) System, per 2-OI-71, RCIC System and Attachments 1, 2 and 3

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed one semi-annual detailed system review. The inspectors completed a detailed alignment verification of the Unit 3 RCIC System using the applicable P&ID flow diagram, 0-47E813-1 along with the applicable operating instruction, 3-OI-71, to verify equipment availability and operability. The inspectors reviewed relevant portions of the UFSAR and TS. This detailed walkdown also verified electrical power alignment, the condition of associated system instrumentation and controls, component labeling, pipe hangers and support installation, and support systems status. Furthermore, the inspectors examined the applicable System Health Reports, WOs, and any PERs that could affect system alignment and operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed licensee procedures, Standard Programs and Processes (SPP)-10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments, and conducted a walkdown of the six fire areas (FA) and fire zones (FZ) listed below. Selected FAs/FZs were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and operational lineup and operational condition of fire protection features or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure SPP-10.9. Furthermore, the inspectors reviewed applicable portions of the Site Fire Hazards Analysis Volumes 1 and 2 and Pre-Fire Plan drawings to verify that the necessary fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place.

- Unit 1/2 Diesel Generator Building (FA-20)
- Unit 3 Reactor Building West, Elevator Vestibule, and Equipment Hatch (FZ 3-1)
- Unit 3 Reactor Building East Side, Elevations 519' and 565' (FZ 3-2)
- RHRSW and Circulating Cooling Water Pump Intake Building (FA -25)
- RHRSW Intake Cable Tunnel (FA -25)
- Common Intake Structure Elevation 550' (FA-25)

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors performed one licensed operator requalification review. On July 7, 2008, the inspectors observed licensed operator requalification simulator examination and training for two crews. Each crew received the same examination scenario contained in OPL177.062, Simulator Exercise Guide, High Pressure Coolant Injection (HPCI) System Logic Failure, Reactor Feed Pump Trip, Reactor Building Closed Cooling Water (RBCCW) System Leak In Drywell, Feedwater Line Rupture In Turbine Building, and Recirculation Line Break.

The inspectors specifically evaluated the following attributes related to each operating crew's performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of Abnormal Operating Instructions (AOIs), and Emergency Operating Instructions (EOIs)
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures (EPIPs)
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

The inspectors attended a post-examination critique to assess the effectiveness of the licensee evaluators, and to verify that licensee-identified issues were comparable to issues identified by the inspector. The inspectors also reviewed aspects of simulator physical fidelity (i.e., the degree of similarity between the simulator and the reference plant control room, such as physical location of panels, equipment, instruments, controls, labels, and related form and function).

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

Quarterly Inspection

a. Inspection Scope

The inspectors reviewed two specific equipment issues listed below for structures, systems and components (SSCs) within the scope of the Maintenance Rule (MR) (10

CFR 50.65) with regard to some or all of the following attributes: work practices; identifying and addressing common cause failures; scoping in accordance with 10 CFR 50.65(b) of the MR; characterizing reliability issues for performance; trending key parameters for condition monitoring; charging unavailability for performance; appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2); system classification in accordance with 10 CFR 50.65(a)(1); and appropriateness and adequacy of (a)(1) goals and corrective actions (i.e., licensee's Ten Point Plan). The inspectors also compared the licensee's performance against site procedure SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; and SPP 3.1, Corrective Action Program. Furthermore, the inspectors examined, as appropriate, applicable work orders, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

- Unit 2 HPCI System Declared (a)(1) Due to Exceeding Performance Criteria for Unavailability
- Unit 3 Residual Heat Removal (RHR) Heat Exchanger Floating Head Leakage Return to (a)(2)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For planned online work and/or emergent work that affected the five combinations of risk significant systems listed below, the inspectors reviewed the licensee maintenance risk assessments, including the actions taken to plan and control work activities to effectively manage and minimize risk. The inspectors also verified that these risk assessments and applicable risk management actions were conducted as required by 10 CFR 50.65(a)(4) and applicable plant procedures such as SPP-7.1, Work Control Process; 0-TI-367, BFN Equipment to Plant Risk Matrix; and BP-336, Risk Determination And Risk Management. Furthermore, the inspectors evaluated the adequacy of the licensee's risk assessments and implementation of risk management actions (RMAs).

- 3C EDG, With Unit 1 HPCI Pump and A3 EECW Pump Out of Service (OOS)
- 2B RHR Heat Exchanger, With D EDG, and C3 EECW Pump OOS
- C1 RHRSW Pump, With 3A EDG and A3 EECW Pump OOS
- Main Bank Battery 3, A RHRSW Subsystem, With 1A RHR Pump and C3 EECW Pump OOS
- 3C EDG, With Unit 3 HPCI Pump and B1 RHRSW Pumps OOS

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the four operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, the inspectors reviewed licensee procedure SPP-3.1, Corrective Action Program, Appendix D, Guidelines for Degraded/Non-conforming Condition Evaluation and Resolution of Degraded/Non-conforming Conditions, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors examined the implementation of compensatory measures to verify that they achieved the intended purpose and that the measures were adequately controlled. The inspectors also reviewed PERs on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- PER 147494, Unit 3 RHR System Piping Over-pressurization
- PER 147283, Unit 1 and 2 C EDG Heat Exchanger Fouling
- PER 148545, Low A RHRSW Header Pressure Due to Failure of A2 RHRSW Pump Discharge Check Valve to Fully Seat
- PER 148300, RHR Room Cooler EECW Flow Measurement Inaccuracy

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the one temporary modification listed below to verify regulatory requirements were met, along with procedures such as 0-TI-405, Plant Modifications and Design Change Control; 0-TI-410, Design Change Control; and SPP-9.5, Temporary Alterations. The inspectors also reviewed the associated 10 CFR 50.59 screening and evaluation and compared each against the UFSAR and TS to verify that the modification did not affect operability or availability of the affected system. Furthermore, the inspectors walked down each modification to ensure that it was installed in accordance with the modification documents and reviewed post-installation and removal testing to verify that the actual impact on permanent systems was adequately verified by the tests.

- TACF 2-08-006-080, Alarm Setpoint Change for TR-80-1, Point 5, Drywell Cooling Unit 2B Inlet Air Temperature

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors performed one permanent plant modification review. The inspectors reviewed the Design Change Notice (DCN) and completed work package WO 05-712911-001 for DCN 63689, Modification to Spent Fuel Pool to Accommodate HI-TRAC Associated with Dry Cask Storage, including related documents and procedures. The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control, and SPP-9.3, Plant Modifications and Engineering Change Control, and observed part of the licensee's activities to implement this design change made while the unit was online. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation to verify that the modifications had not affected system operability/availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the seven post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed SSC operability and functional capability following maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors verified that PMT activities were conducted in accordance with applicable WO instructions, or procedural requirements, including SPP-6.3, Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed any relevant PERs associated with PMTs that were identified and entered into the licensee's corrective action program (CAP).

- WOs 07-726655-003, 07-726655-012 and 07-726655-032, Common: PMT for C1 RHRSW Pump Motor Cable Replacement, Coupling Repair and Motor Replacement in accordance with 2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test.
- WO 08-718803-000, Unit 1/2: PMT for EDG B Field Over-current Alarm in accordance with 1/2-SI-4.9.A.1.d(B), Diesel Generator B 2-Year Inspection and 0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test.

- WO 08-711478-000, Unit 1: PMT for RHR Pump 1D in accordance with 1-OI-74, Residual Heat Removal System.
- WO 08-719647-000, Unit 2: PMT for Replacement of 2-71X-085-45G, East Scram Discharge Volume High Level Trip Relay, in accordance with 2-SR-.3.1.1.8(7A/G), RPS and Tank Not Drained High Water Level in Scram Discharge Tank Functional Test.
- Unit 3: PMT for 3D EDG in accordance with 3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test following 2-year Maintenance Outage.
- WO 07-721939-000, Unit 3: PMT for Standby Liquid Control (SLC) Pump 3B Maintenance in accordance with 3-SI-4.4.A.1, SLC Pump Functional Test.
- WO 08-720185-001, Unit 3: PMT for Replacement of Cell #114 in Main Bank Battery #3 in accordance with 3-SR-3.8.6.2(3), Quarterly Check of 250 Volt Main Bank Number 3 Battery.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 3 Forced Outage Due to Steam Leak in the 3B SJAE Room

a. Inspection Scope

On July 11, 2008, a planned shutdown was conducted on Unit 3 to repair a steam leak from a manual isolation valve (Main Steam to the 3B Off-Gas pre-heater), which could not be isolated at power. The unit was restarted on July 12 and returned to 10 percent RTP on July 14, 2008. During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee’s outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Plant shutdown and cooldown per General Operating Instruction (GOI) 3-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations
- Control of Cold Shutdown (Mode 4) conditions and critical plant parameters
- Plant Oversight Review Committee (PORC) event review and restart meeting in accordance with SPP-10.5, Plant Operations Review Committee
- Reactor startup and power ascension activities per 3-GOI-100-1A, Unit Startup
- Outage risk assessment and management per SPP-7.2, Outage Management
- Control and management of forced outage and emergent work activities per SPP-7.2

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage and attended Corrective Action Review Board (CARB) meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings of significance were identified.

.2 Unit 1 Forced Outage Due To Leakage on Class 2 Piping on 1A Main Steam Line

a. Inspection Scope

On August 8, 2008, an unplanned shutdown was conducted following discovery of a Code Class 2 piping leak on the 1A Main Steam Line that could not be isolated at power. The unit was restarted on August 13, 2008 and was restored to 100 percent RTP on August 15, 2008. During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Reactor Shutdown and cooldown activities per 1-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations
- Control of Cold Shutdown (Mode 4) conditions, and critical plant parameters
- Licensee's conduct of 1-GOI-200-2, Drywell Closeout; and an independent detailed closeout inspection of the Unit 1 drywell by the inspectors
- PORC event review and restart meeting in accordance with SPP-10.5
- Reactor startup and power ascension activities per 1-GOI-100-1A, Unit Startup
- Outage risk assessment and management per SPP-7.2
- Control and management of forced outage and emergent work activities per SPP-7.2

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 1 forced outage and attended CARB meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings of significance were identified.

.3 Unit 2 Forced Outage Due To Hydrogen Leak on Main Generator

a. Inspection Scope

On September 16, 2008, Unit 2 reduced power to 22 percent RTP to repair a hydrogen leak on the main generator. The unit was subsequently shutdown the following day due to an unplanned increase in work scope to affect repairs to a containment isolation valve and a rod position indicator, and subsequent to the shutdown, a failed snubber on a recirculation pump motor. The unit was restarted on September 21 and reached 100 percent RTP on September 24, 2008. During this short forced outage the inspectors examined the conduct of critical outage activities pursuant to TS, applicable procedures, and the licensee's outage risk assessment and outage management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Reactor Shutdown and cooldown activities per 2-GOI-100-12A, Unit Shutdown from Power Operations to Cold Shutdown and Reduction in Power During Power Operations
- Control of Cold Shutdown (Mode 4) conditions, and critical plant parameters
- Licensee's conduct of 2-GOI-200-2, Drywell Closeout; and an independent detailed closeout inspection of the Unit 2 drywell by the inspectors
- PORC event review and restart meeting in accordance with SPP-10.5
- Reactor startup and power ascension activities per 2-GOI-100-1A, Unit Startup
- Outage risk assessment and management per SPP-7.2
- Control and management of forced outage and emergent work activities per SPP-7.2

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 2 forced outage and attended CARB meetings to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data for the following five surveillance tests of risk-significant and/or safety-related systems to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement.

Reactor Coolant System Leak Detection Tests:

- 1-SR-3.4.4.1, Manual Calculation of Unidentified, Identified and Total Leakage

In-Service Tests:

- 2-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure
- 2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test

Routine Surveillance Tests:

- 0-SR-3.7.3.4, Control Bay Habitability Zone Pressurization Test
- MPI-0-000-INS001, Inspection of Flood Protection Devices

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

During the report period, the inspectors observed one Emergency Preparedness (EP) drill on August 6, 2008, to identify any weaknesses and deficiencies in classification, notification, dose assessment and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room and the Technical Support Center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Classification Procedure and other applicable Emergency Plan Implementing Procedures. The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying weaknesses.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Cornerstone: Mitigating Systems

Mitigating Systems Performance Indicator

a. Inspection Scope

The inspectors reviewed the licensee's procedures and methods for compiling and reporting the fifteen PIs listed below, including procedure SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PIs to the NRC. The inspectors reviewed the raw data for the PIs listed below for the third quarter of 2007 through second quarter of 2008 and discussed the methods for compiling and reporting the PIs with cognizant licensing, engineering, and maintenance rule personnel. The inspectors independently reviewed selected plant documents (e.g., plant operating logs, Limiting Condition of Operation Tracking (LCOTR) Logs, PERs, etc.) and maintenance rule unavailability and unreliability records. The inspectors also independently verified inputs and compared the results of the system specific derivation reports against graphical representations and specific values reported to the NRC for the second quarter 2008 PI report to verify that the data was correctly reflected in the report. The inspectors also reviewed the past history of PERs for any that might be relevant to problems with the PI program. The inspectors reviewed Nuclear Energy Institute (NEI) 99-02, Regulatory

Assessment Performance Indicator Guideline, to verify that industry reporting guidelines were applied. Although the Unit 1 Mitigating Systems Performance Index (MSPI) PIs will not be considered valid until the third quarter of 2010, the inspectors conducted the inspection described above for the Unit 1 data as part of the Unit 1 augmented baseline inspection plan described in NRC letter dated May 21, 2008. In addition, for the Unit 1 MSPI PI data, the inspectors ensured that the unavailability and unreliability information in the Unit 1 Basis Document was accurately reflected in the derivation reports.

- Unit 1 Mitigating Systems Performance Index - HPCI
- Unit 2 Mitigating Systems Performance Index - HPCI
- Unit 3 Mitigating Systems Performance Index - HPCI
- Unit 1 Mitigating Systems Performance Index - RCIC
- Unit 2 Mitigating Systems Performance Index - RCIC
- Unit 3 Mitigating Systems Performance Index - RCIC
- Unit 1 Mitigating Systems Performance Index - Emergency AC Power
- Unit 2 Mitigating Systems Performance Index - Emergency AC Power
- Unit 3 Mitigating Systems Performance Index - Emergency AC Power
- Unit 1 Mitigating Systems Performance Index - RHR
- Unit 2 Mitigating Systems Performance Index - RHR
- Unit 3 Mitigating Systems Performance Index - RHR
- Unit 1 Mitigating Systems Performance Index - Cooling Water (RHRSW/EECW)
- Unit 2 Mitigating Systems Performance Index - Cooling Water (RHRSW/EECW)
- Unit 3 Mitigating Systems Performance Index - Cooling Water (RHRSW/EECW)

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of items Entered into the Corrective Action Program

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s CAP. This review was accomplished by reviewing daily PER report summaries and periodically attending daily CARB meetings.

.2 Focused Annual Sample Review - Operator Workarounds

a. Inspection Scope

The inspectors conducted one focused review of existing Operator Workarounds (OWAs) to verify that the licensee was identifying OWAs at an appropriate threshold, entering them into the CAP, establishing adequate compensatory measures, prioritizing resolution of the problems, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. The inspectors examined all active OWAs listed in the LCOTR Log, and reviewed them against the guidance in OPDP-1, Section 4.7.B, Operator Workarounds. The inspectors also discussed these OWAs in detail with on-shift operators to assess their familiarity with the degraded conditions and knowledge of required compensatory actions. Furthermore, the inspectors walked down selected OWAs, and verified the ongoing performance, and/or feasibility of, the required actions. Lastly, for selected OWAs, the inspectors reviewed the applicable PER, including the associated functional evaluation and corrective action plans (both interim and long term).

b. Findings and Observations

No findings of significance were identified. However, the inspectors had the following observations:

- Operations department procedure OPDP-1, Conduct of Operations, has limited guidance for when a PER should be initiated and for escalation of OWA priority based on cumulative impact on a specific watch station or aggregate impact on the plant.
- Several OWAs contained instructions to operate components in an abnormal manner to compensate for degraded equipment, bypassing the normal review and approval process for changing procedures.
- The OWA process did not specify a 50.59 review. NEI 96-07, Guidance for 10CFR 50.59 Evaluations, suggests that compensatory actions taken as a result of a degraded or non-conforming condition receive a 50.59 screening/evaluation.

The licensee entered these observations into their corrective action program as PER 151424.

.3 Focused Annual Sample Review

a. Inspection Scope

The inspectors reviewed the specific corrective actions associated with Unit 2 PER 144253.

The inspectors reviewed the cause evaluation and specific corrective actions associated with PER 144253, High Moisture Content in Unit 2 HPCI Pump Oil. This PER was initiated to evaluate the cause of the high moisture content in the HPCI oil discovered following surveillance testing of the pump. The cause was determined to be leakage past the steam admission valve that resulted in turbine seal leakage and moisture entry into the oil system. A contributing cause was that the insulation at the turbine glands was configured such that it trapped condensation rather than allowing it to dissipate to the room. The licensee took immediate corrective actions that included removing the pump from service, replacing the oil and inspecting individual components that may have been degraded by this condition. The inspectors reviewed the interim corrective actions instituted to prevent recurrence of this condition until the steam admission valve can be repaired. These interim actions included modification to the insulation of the turbine; installation of temporary ventilation to direct any leakage past the turbine seals away from the oil system; increased monitoring of the leakage past the valve; and increased frequency oil sampling and analysis. The long-term actions included repair to the steam admission valve, revision to the maintenance procedures that repair the valve and institution of preventative maintenance on this valve on a regular frequency. In addition, the inspectors verified that the licensee had addressed Extent of Condition issues by inspecting each unit's HPCI and RCIC pump insulation and applying the changes in the preventative maintenance (PM) program to all units.

b. Assessment and Observations

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000296/2008-001-00, Unanticipated Auto-Start of Emergency Diesel Generators

a. Inspection Scope

On May 5, 2008, EDGs 3EC and 3ED auto-started and tied to their respective shutdown boards due to an undervoltage condition on the associated Shutdown Boards. Operations personnel were in the process of returning the Unit 3 4KV Unit Board 3B to the normal supply when the board failed to transfer and was de-energized. The inspectors reviewed the applicable LER that was issued on July 7, 2008, and it's associated PER 144272, which included the root cause determination and corrective action plans. The root cause of this event was determined to be misalignment of the stationary breaker indicating switch mechanism in the normal feeder breaker for 4KV Unit Board 3B which prevented the alternate feeder breaker from closing in. Corrective actions plans were developed that included revising the breaker PM program, as needed, to improve stationary breaker indicating switch reliability.

b. Findings

No significant findings or violations of NRC requirements were identified. This LER is considered closed.

.2 (Closed) LER 05000296/2008-002-00, Main Steam Relief Valve As Found Setpoint Exceeded Technical Specification Lift Pressure

The inspectors reviewed the LER dated July 31, 2008, and the applicable PER 146189, including associated apparent cause determination and corrective action plans.

Following the Unit 3 Cycle 13 refueling outage, the licensee tested 13 Main Steam Relief Valve (MSRV) pilot cartridges that had been in service during the fuel cycle. During the surveillance testing following Cycle 13 operations, seven of the 13 MSRVs lifted at a pressure that was greater than the allowed 3 percent margin above the setpoints prescribed by TS 3.4.3. The cause of the MSRV as-found setpoints being above their TS limits was determined to be corrosion bonding between the pilot valve seat and disc, which was a recognized industry problem. In addition to these seven MSRVs, one MSRV pilot valve failed to lift at all due to excessive seat leakage. The failure of these MSRVs to lift within the allowed setpoint limits constituted a condition prohibited by TS 3.4.3. However, subsequent Pressure Transient Analysis by the licensee concluded that the as-found condition of the MSRVs from Cycle 13 would have been sufficient to fulfill the over pressure relief safety function during design basis over-pressure transient events.

b. Findings

This LER is considered closed. Since the setpoint drift problems were found during surveillance testing this LER was dispositioned as an NCV in Section 4OA7 of this report.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No significant findings were identified.

.2 (Open) NRC Temporary Instruction (TI) 2515/176, Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing

a. Inspection Scope

The objective of this TI was to gather information to assess the adequacy of nuclear power plant EDG endurance and margin testing as prescribed by plant-specific TS. The inspectors interfaced with the appropriate station staff to obtain the information specified in Attachment 1 of the TI, Worksheet. The TI applies to all operating nuclear power reactor licensees that use EDGs as the onsite standby power supply. The inspectors verified the accuracy of the information by review of TS, EDG Design Basis Event loading calculations, EDG endurance run test procedures, test data from the last three endurance tests performed on each EDG, EDG ratings, and EDG operating history. The information gathered will be forwarded to the Office of Nuclear Reactor Regulation, Division of Engineering, Electrical Engineering Branch (NRR/DE/EEEB) for further review to assess the adequacy and consistency of EDG testing at nuclear stations.

b. Findings

The TI is presently scheduled to be open until August 31, 2009, pending completion of the NRR/DE/EEEB review.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 3, 2008, the senior resident inspector presented the inspection results to Mr. Rusty West and other members of the staff. No proprietary information was included in this inspection report.

4OA7 Licensee-Identified Violations

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

- The licensee identified a violation of TS 3.4.3 which required that twelve of thirteen MSRVs lift at a setpoint within plus or minus 3 percent of a specified value. Contrary to this, during surveillance testing following the Unit 3 Cycle 13 refueling outage, the licensee discovered that eight MSRVs did not meet the TS allowed pressure band as described in the licensee's PER 146189. This finding was determined to be of very low safety significance because the as-found lift setpoint conditions of the Unit 3 MSRVs were analyzed and demonstrated to meet the design basis criteria for an over-pressurization event.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Berry, Systems Engineering Manager
S. Bono, Engineering Manager
T. Brumfield, Site Nuclear Assurance Manager
P. Chadwell, Field Maintenance Superintendent
T. Chan, Corporate Engineering
S. Douglas, General Manager of Site Operations
A. Elms, Assistant General Manager Operations
J. Emens, Site Licensing Supervisor
D. Feldman, Interim Operations Manager
E. Frevold, Design Engineering Manager
L. Hughes, Operations Superintendent
C. Johnson, System Engineer - EDG
D. Langley, Site Licensing Manager
E. Leonard, System Engineer
S. Lovvorn, Electrical Design Engineer
J. Mitchell, Site Security Manager
R. Rogers, Maintenance & Modifications Manager
P. Sawyer, Radiation Protection Manager
E. Scillian, Operations Training Manager
R. Stowe, Nuclear Operations Support Superintendent
J. Underwood, Chemistry Manager
L. Vandiver, Probabilistic Risk Assessment Engineer
R. West, Site Vice President
J. Woodward, Equipment Reliability Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

TI 2515/176	TI	Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing (Section 4OA5)

Closed

05000296/2008-001	LER	Unanticipated Auto-Start of Emergency Diesel Generators (Section 4OA3.1)
05000296/2008-002	LER	Main Steam Relief Valve As Found Setpoint Exceeded Technical Specification Lift Pressure (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Documents
FSAR Section 2.4, Hydrology, Water Quality, and Marine Biology, BFN-19
FSAR Appendix 2.4A, Maximum Possible Flood, BFN-22
FSAR Appendix 10.9, RHR Service Water System, BFN-22
FSAR Section 12.2, Principle Structures and Foundations, BFN-22
TRM 3.3.6, Flood Protection Instrumentation, Rev. 0
BFN-50-C-7101, Protection from Wind, Tornado Wind, Tornado Depressurization, Tornado Generated Missiles, and External Flooding, Rev. 3
DCN 63691, Modification to Equipment Access Lock/Roadway, Rev. A
DCN 63959, Replace portions of 4KV RHRSW Pump Cables routed underground in wet locations, Rev. A
DCN 68803, Replace faulted section of cable to A2 RHRSW pump and rework sump pump in Man-Hole 15, Rev. A

Procedures
0-AOI-100-3, Flood Above Elevation 558', Rev. 32
0-TI-171, RHRSW Sump Pump Flow Rate Test, Rev. 6
0-SI-4.2.H-1, Reservoir Level Monitoring Calibration and Functional Test, Rev. 14
1-ARP-9-22A, Annunciator Response, Rev. 4
CCI-0-LS-23-087, RHR Service Water Pump Compartment Level Switches, Rev. 4
EPI-0-000-SWZ006, Calibration and Inspection of Station Drainage and Intake Sump Pump Level Switches, Rev. 20
MPI-0-000-INS001, Inspection of Flood Protection Devices, Rev. 10
MPI-0-260-DRS001, Inspection and Maintenance of Doors, Rev. 34

Calculations
MD-Q0023-870149, RHRSW Pump Compartment Sump and Sump Pump Capacity, Rev. 9
MD-Q0999-920112, Prevention of Backflooding, Rev. 3

Drawings
0-47E851-1, Units 1-3 Flow Diagram Drainage, Rev. 28
0-47E851-4, Units 1-3 Flow Diagram Drainage, Rev. 13
0-47E851-6, Unit 0 Flow Diagram Drainage, Rev. 2
0-47W585-1, Standby Diesel Generator Building Units 1 & 2 Mechanical Drains & Embedded Piping, Rev. 2
3-47W587-1, Standby Diesel Generator Building Unit 3 Mechanical Drains & Embedded Piping, Rev. 3
0-44N236-4, Reactor Building Equipment Access Lock Flood Gate Machinery Arrangement, Rev.
0-45E777-20, Wiring Diagram, 480V Unit Auxiliary Power Schematic Diagram. Rev. 4

PERs
133899, RHRSW Pump Room Doors
134338, Handhole-15 (East plant yard) was partially flooded
134346, Missing documentation for inspection of water tight doors

134453, Calculation on maximum possible flooding event was revised in 1998 and not incorporated into licensing basis
135220, River height specified for shutdown in Flooding AOI, may not provide enough time for shutdown
136149, Calibration and functional check for the EQ access lock waterseal level switch needs to be added to EPI-0-077-SWZ002
137115, DCN for routing discharge piping from CO2 Relief Valve to outside the Unit 1/2 Diesel Generator Building did not address impact on external flooding analysis
139398, Potential for flooding the OG building
143051, Operating experience review for Underground Cable Ground Faults
146998, Errors Found in TVA Generic Probable Maximum Flood Calculation
147337, Dallas Bay contribution to PMF calculation was omitted

Work Orders
07-720996-000, Door Inspection PM
07-727161-000, Inspect Flood Gate
07-712417-001, A2 RHRSW Pump Motor Cable Replacement
08-711479-000, Inspect, Clean and Test 480v RVNT Board 1B, Compartment 12A, Feeder For RB Equipment Access Lock Floodgate Machinery

Section 1R04: Equipment Alignment

1-OI-75, Core Spray System, Rev. 9, and Attachments 1, 2 and 3.
2-OI-71, Reactor Core Isolation Cooling (RCIC) System, Rev. 57 and Attachments 1, 2, and 3
3-OI-71, Reactor Core Isolation Cooling (RCIC) System, Rev. 42 and Attachments 1, 2, 3 and 4
3-OI-82, Standby Diesel Generator System, Rev 83 and Attachments 1C, 2C, and 3C.
1-47E814-1, Unit 1 Core Spray System Flow Diagram, Rev. 23
2-47E813-1, Flow Diagram Reactor Core Isolation Cooling System, Rev. 47
3-47E813-1, Flow Diagram Reactor Core Isolation Cooling System, Rev. 45
3-47E861-3, Flow and Control Diagram Diesel Starting Air System Diesel Gen. 3C, Rev. 20
3-47E861-7, Flow Diagram Cooling System and Lubricating Oil System Standby Diesel Generator 3C, Rev. 14
FSAR Section 4.7, Reactor Core Isolation Cooling System, BFN-21
FSAR Section 7.3, Primary Containment Isolation System, BFN-22
FSAR Section 7.18, Backup Control System, BFN-18
Technical Specifications and Bases 3.5.3, RCIC System, Amendment 249
Technical Specifications and Bases 3.3.3.2, Backup Control System, Amendment 244
Technical Specifications and Bases 3.3.5.2, RCIC System Instrumentation, Amendment 213
Technical Specifications and Bases 3.3.6.1, Primary Containment Isolation Instrumentation, Amendment 213
Technical Requirements Manual and Bases 3.3.3.4, ECCS and RCIC Trip System Bus Power Monitors, Rev. 0
Design Criteria BFN-50-7071, Reactor Core Isolation Cooling System, Rev. 15
System Health Report Unit 3 System 71, RCIC, FY2008 – P2
WO 08-720332-000, BFN-3-FI -071-0001B: Flow Indicator Indicates ~ 6000 Lbs/Hr with RCIC In Standby
WO 08-720221-000, RCIC Turbine Oil Level is Low
WO 08-720160-000, Cut Back Insulation Away From Unit 3 RCIC Turbine Inboard And Outboard Steam Seals

WO 08-719325-000, Small Oil Leak on a Union for Temperature Element 3-TE -071-0047
WO 08-716277-000, Perform Switch Adjustments for 3-ZS-71-10A and 10B
WO 08-713472-000, Perform Troubleshooting and Corrective Maintenance to Return ICS 'Trip' Alarm to Service
PER 145567, Unit 3 RCIC Steam Inlet Line Condensate Inboard Drain Valve, failed closed
PER 145357, U3 RCIC declared inoperable due to blown diaphragm on 3-LCV-71-6A
PER 145211, During U3C13 outage, 3-FCV-071-0006A had multiple diaphragm failures when the valve was stroked
PER 144879, Backup Control Panel RCIC Bias Speed Setting Potentiometer found removed
PER 140425, During Backup Control Panel Testing, no indicating lights were lit for 3-FCV-43-14
PER 135479, Heat related damage found to 6 conductors from the 3-LCV-71-5 to JBOX
PER 131431, U3 RCIC, Condensate Pump for Barometric Condenser had Ground
CDE 685, U3 RCIC Functional Failure due to blown diaphragm on 3-LCV-71-6A
CDE 622, U3 RCIC Condensate Pump for Barometric Condenser Grounded

Section 1R05: Fire Protection

Fire Protection Report, Volume 1, Fire Hazards Analysis for Units1/2/3, Rev. 1
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No DG12-583, Rev. 8
Fire Protection Report, Volume 2, Section IV.2, Pre-plan No DG12-565, Rev. 8
Fire Protection Report, Volume 2, Section IV.7, Pre-plan No RX3-565, Rev. 7
Fire Protection Report, Volume 2, Section IV.7, Pre-plan No RX3-519, Rev. 7
Fire Protection Report, Volume 2, Section IV.7, Pre-plan No RX3-519NE, Rev. 7
Fire Protection Report, Volume 2, Section IV.7, Pre-plan No RX3-519SE, Rev. 7
Fire Protection Report, Volume 2, Section IV.13, Pre-plan No IS-550, Revision 8
Fire Protection Report, Volume 2, Section IV.13, Pre-plan No IS-565
Fire Protection Report, Volume 2, Section IV.14, Pre-plan No ISCT-GRD
Fire Protection Impairment # 08-1599, Theromlag Removed for access to J-Boxes in order to pull cable for RHRSW pumps
FP-0-000-INS001(D), Inspection of Portable and Wheel Type Fire Extinguisher Stations, Rev. 14
0-SI-4.11.A2(2), Supervised Fire System Circuit Operabilty Test, conducted July 24, 2008
0-SI-4.11.A.1(4), Local Fire Control Panel 0-LPNL-025-0538 Intake Pumping Station And Cable Tunnel Detection Operability Test, completed April 15, 2008
1-SI-4.11.A.1, Annual Smoke Detector Functional Test, completed July 25, 2008

Section 1R12: Maintenance Effectiveness

PER 81236, 2A RHR HX Leakage
RHRSW/EECW Cable (a)(1) 10 Point Plan
Maintenance Rule Expert Panel Meeting Minutes dated September 11, 2008
PER 144253, High Moisture in Unit 2 HPCI Oil
CDE 663, 2-FCV-73-16 unplanned unavailability
CDE 690, Unit 2 HPCI Exceeds Unavailability Performance Criteria

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

0-TI-367, BFN Equipment to Plant Risk Matrix, Rev. 10
BP-336, Risk Determination and Risk Management, Rev. 7
SPP-7.1, On Line Work Management, Rev. 10

Calculation ND-N0999-000009, Risk Significance for On-Line Maintenance, Rev. 3
EWR 08-MEB-999-07, PRA Evaluation for Components Out-of-Service
EWR 08-MEB-999-095, PRA Evaluation for Components Out-of-Service

<u>Section 1R15: Operability Evaluations</u>

Functional Evaluation (FE) 42709, Fouling of U0 C1 and C2 Diesel Generator Heat Exchangers
PER 147283 Fouling of U1/2 C1 and C2 DG Coolers
Memorandum, DG Operability Question due to Clams, Shults to Feldman, dated 6/26/08
Technical Specifications 3.7.1, RHRSW and UHS, Amendment 214
Technical Specifications 3.7.2, EECW and UHS, Amendment 215
FSAR Section 10.10, Emergency Equipment Cooling Water System, BFN-22
BFN Mechanical Calculation MD-Q0067-910008, Flow Requirements of EECW Fed Components, Rev. 13
BFN Mechanical Calculation MD-Q0067-880201, Diesel Generator Heat Exchangers in Series Analysis, Rev. 3
BFN Mechanical Calculation MD-Q0082-000016, Diesel Generator Jacket Water, Rev. 1
BFN-50-7067, Emergency Equipment Cooling Water, R17
BFN-50-7023, Emergency Residual Heat Removal Service Water System, R12
BFN-50-7082, Standby Diesel Generator, R13
WO 07-721077-000, U0 C1 and C2 DG Heat Exchanger Eddy Current Testing

Functional Evaluation 42711, U3 RHR Loop II Potential Over Pressurization Due to Check Valve Leakage
3-SR-3.6.1.3.5(RHR II), RHR System MOV Operability Loop II, Rev. 9
0-TI-362, Inservice Testing of Pumps and Valves, Rev. 19
Drawing 3-47E811-1, RHR Flow Diagram, Rev. 63
Unit 3 Technical Specifications 3.5.1, 3.5.2, 3.6.1.3.11
FSAR Section 4.8, Residual Heat Removal System
FSAR Section 6.4, Emergency Core Cooling - Description
FSAR Section 6.5, Emergency Core Cooling - Safety Evaluation
FSAR Section 7.4, Emergency Core Cooling Control and Instrumentation

Functional Evaluation (FE) 42761, ‘A’ RHRSW Header Pressure
PER 148545, Failure of A2 RHRSW Pump Discharge Check Valve to Fully Seat
0-OI-23, Residual Heat Removal Service Water System, Revision 85
2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test, Revision 99
2-ARP-9-3E, Panel 9-3 2-XA-55-3E, Window 31, RHRSW HDR PRESS LOW 2-PA-23-4, Revision 21
WO 04-712616-035, X-ray, Disassemble, and Inspect 0-CKV-023-0506
Drawing 1-47E858-1, Flow Diagram RHR Service Water System, Revision 56
Calculation MD-Q0023-880334, Evaluation of Pipe/Check Valve Failure at the Interface of the RSW and RHRSW Systems, Revision 1
Technical Specifications and Bases 3.7.1 Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)
FSAR Section 10.9, RHR Service Water System, BFN-22
General Design Criteria BFN-50-7023, Residual Heat Removal Service Water System, Revision 12

Section 1R18: Plant Modifications

Temporary Modification

TACF 2-08-006-080, Revise Setpoint on 2-TR-80-1, Point 5, Drywell Cooling Unit 2B Inlet Air Temperature
WO 07-718659-000, Troubleshoot and Repair Unit 2 Drywell Atmosphere Temperature High Alarm
2-SIMI-80B, Primary Coolant System Scaling and Setpoint Documents, Rev. 13
2-ARP-9-3B, Alarm response Procedure, Rev. 21
CCI-0-XR-00-236, Yokogawa HR 2500 Series Recorder Calibration, Rev. 7
0-OI-55, Annunciator System, Rev. 42
OPDP-4, Annunciator Disablement, Rev. 2
Drawing 2-47E610-80-1, Mechanical Control Diagram, Primary Containment Cooling Temperature Monitoring System, Rev. 2
2-080-OWA-2008-0085, Verification of DW Temperatures with Alarm Locked in
2-SR-2, Instrument Checks and Observations, Rev. 60
2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, Rev. 23
Unit 2 Technical Specification 3.6.1.4, Drywell Air Temperature
Unit 2 Technical Specification Basis B3.6.1.4, Drywell Air Temperature
FSAR Section 5.2, Primary Containment System
FSAR Section 14.6, Analysis of Design Basis Accidents - Uprated
FSAR Table 14.6-3, Summary of Power Uprate Input Parameters Used for All Containment Analyses

Permanent Modification

WO 05-712911-001, Implement DCN 63689 Stage 2
DCN 63689, Modification to Spent Fuel Pool to Accommodate HI-TRAC Associated with Dry Cask Storage, Rev A
PIC-69420, Change drawings to reflect removal of swing bolt lugs and cut-outs on leveling plates, Rev. A
0-TI-405, Plant Modifications and Design Change Control, Rev. 0
SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 16
FSAR Section 10.3, Spent Fuel Storage
FSAR Section 10.4, Tools and Servicing Equipment
FSAR Section 10.5, Fuel Pool Cooling and Cleanup System
BFN-50-7079, Fuel Handling and Storage System, Rev. 9
Drawing 48N959, Unit 1 Miscellaneous Steel, Spent Fuel Storage Pool Liner Plan, Rev. 3
PER 149129, Dry Cask Fuel Pool Mod Interference
SPP-6.5, Foreign Material Exclusion, Rev. 13
RCI-33, Underwater Diving Operations, Rev. 4
TVA Safety Manual Procedure 814, Underwater Diving, Rev. 3

Section 1R19: Post-Maintenance Testing

WO 08-711478-000, Perform Bridge and Megger, DC Step Voltage Testing and Borescope Inspection of 1D RHR Pump
WO 08-714009-000, Perform Thermography, Motor Current Signature Analysis and Strobe Measurement of Shaft Speed for 1D RHR Pump

WO 08-713996-000, Perform Inspection and Bridge and Megger Testing for RHR Pump 1D Cooling Fan Motor
1-OI-74, Residual Heat Removal System, Rev. 58
EPI-0-000-TST001, Bridge, Megger and High Potential Testing of Electrical Equipment, Rev 54
EPI-0-000-INS001, RHR/Core Spray Pump Motor Borescope Inspection, Rev 2
WO 08-711478-000, Perform Bridge and Megger, DC Step Voltage Testing and Borescope Inspection of 1D RHR Pump
WO 08-714009-000, Perform Thermography, Motor Current Signature Analysis and Strobe Measurement of Shaft Speed for 1D RHR Pump
WO 08-713996-000, Perform Inspection and Bridge and Megger Testing for RHR Pump 1D Cooling Fan Motor
1-OI-74, Residual Heat Removal System, Rev. 58
EPI-0-000-TST001, Bridge, Megger and High Potential Testing of Electrical Equipment, Rev 54
EPI-0-000-INS001, RHR/Core Spray Pump Motor Borescope Inspection, Rev 2
Technical Specifications 3.8.1, AC Sources - Operating
3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test
Work Order 08-718803-000, Troubleshoot B Diesel Generator Field Over-current Alarm
1/2-SI-4.9.A.1.d(B), Diesel Generator B 2-Year Inspection, Rev. 44
0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, Rev. 35
1/2-ARP-9-23B, Annunciator Response, Rev.12
0-47E767-3, Wiring Diagram, Diesel Generators, Schematic Diagram, Rev. 15
0-47E767-3, Wiring Diagram, Diesel Generators A - D, Schematic Diagram, Rev. 9
PER 148258, Received Field Overcurrent Alarm on B Diesel During 2-Year Inspection
WO 08-719647-000, SDV High Level Relay Replacement
WO 08-719647-001, Conduct inspection of SDV Level Switch and Associated Relays in Aux Instrument Room
2-SR-3.3.1.1.8(7A/G), RPS and Tank Not Drained High Water Level in Scram Discharge Tank Functional Test, Rev. 7
ECI-0-000-RLY004, Replacement of HFA Relay Components and/or Calibration of HFA Relays, Rev. 34
WO 07-726655-003, Implement DCN 63969 Stage 3 for C1 RHRSW Pump Motor
WO 07-726655-012, Uncouple/recouple C1 RHRSW Pump in Support of Cable Replacement
WO 07-726655-020, Install and Remove Ground Cart in 4160V Shutdown Board B, Compartment 10
WO 07-726655-027, Perform Bridge/Megger Testing for After Replacement of Cables 0ES100-I for RHRSW Pump Motor C1
WO 07-726655-032, Perform Actions to Remove and Reinstall RHRSW C1 to Allow Motor/Pump Coupling Inspection
WO 07-726655-033, Fabricate New Adjusting Plate for C1 RHRSW Pump
0-SI-3.1.10, RHRSW System Pump Baseline Data Evaluation, Rev. 16
2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test, Rev. 99
ECI-0-000-MOT001, Removal and Reinstallation of AC and DC Motors, Rev. 49
WO 07-721939-000, Breaker Maintenance, Bridge/Megger, and Bearing Lubrication, of SLC Pump Motor 3B
3-SI-4.4.A.1, Standby Liquid Control Pump Functional Test, Rev. 35
EPI-0-000-MOT001, Motor Bearing Lubrication, Rev. 59
EPI-0-000-BKR003, General Electric Type AK-15/25 Circuit Breakers and Switch Gear Maintenance, Rev. 76

EPI-0-000-BKR020, Testing and Troubleshooting of 250 VDC and 480 VAC Power Circuit Breakers and Trip Devices, Rev. 35
EPI-0-000-TST001, Bridge, Megger and High Potential Testing of Electrical Equipment
WO 07-720185-000, Place 250 Volt Main Battery 3 on Equalize Charge per ECI-0-248-BAT001
WO 07-720185-001, Replace Battery Cell #114
3-SR-3.8.6.2(3), Quarterly Check of 250 Volt Main Bank Number 3 Battery (conducted on 8/27/08)
3-SR-3.8.6.2(3), Quarterly Check of 250 Volt Main Bank Number 3 Battery (conducted on 8/22/08)
ECI-0-248-BAT001, Equalize Charging the 250 Volt Main Bank Batteries
ECI-0-248-BAT005, 250 Volt Main Bank 1, 2, and 3 Battery Cell Replacement

Section 1R22: Surveillance Testing

2-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure
0-SR-3.7.3.4, Control Bay Habitability Zone Pressurization Test, Revision 10
BFN-50-7030A, Browns Ferry Nuclear Plant Control Bay and Reactor Building Board Rooms Environmental Control Systems, Revision 12
Drawing 0-47E200-12-1, Control Bay Habitability Zone (CBHZ) Boundary, Revision 1
Control Bay Habitability Zone Breaching Permits: 2062, 2078, 2084, 2162, 2177, 2206, 2208, 2217, 2219, 2221, 2222, 2225, and 2226
0-SI-3.1.10, RHRSW System Pump Baseline Data Evaluation, Rev. 16
2-SI-4.5.C.1(3), RHRSW Pump and Header Operability and Flow Test, Rev. 99
MPI-0-000-INS001, Inspection of Flood Protection Devices, Rev. 10
DCN 63691, Modification to Equipment Access Lock/Roadway, Rev. A
WO 07-727161-000, Inspect Flood Gate
WO 08-713109-000, Inspect Flood Gate
1-SR-3.4.4.1, Manual Calculation of Unidentified, Identified and Total Leakage, Rev. 0
TS 3.4.4, RCS Operational Leakage
TRM 3.3.10, Reactor Coolant Leakage Detection
WO 08-721120-000, Unit 1 DW Equipment Drain Sump Flow Integrator Inoperable
PER 152127, Drywell Equipment Drain Leakage Calculation Overly Conservative

Section 1EP6: Drill Evaluation

EPIP-1, Emergency Classification Procedure, Rev. 43
EPIP-6, Activation and Operation of the Technical Support Center, Rev. 28
Radiological Emergency Plan, Rev. 88

Section 4OA1: Performance Indicator Verification

Procedures, Manuals, and Guidance Documents
NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 5
SPP-3.4, Performance Indicator for NRC Reactor Oversight Process for Compiling and Reporting PIs to the NRC, Rev. 7
Unit 1 Mitigating System Performance Index (MSPI) Basis Document, Revision 2
Unit 2 Mitigating System Performance Index (MSPI) Basis Document, Revision 1
Unit 3 Mitigating System Performance Index (MSPI) Basis Document, Revision 1

Records and Data

Unit 1, 2 and 3 Second Quarter PI Summary Sheet
Unit 1, 2 and 3 MSPI Derivation Report for Unreliability Index for each applicable system (Emergency AC Power, High Pressure Injection, Heat Removal System, Residual Heat Removal and Cooling Water Systems) for the second quarter 2008.
Unit 1, 2 and 3 MSPI Derivation Report Unavailability Index for each applicable system for the second quarter 2008
Maintenance Rule Spreadsheets for each applicable system
LCO Tracking Log from July 1, 2007 to June 30, 2008
Cause Determination Evaluation (CDE) 547, 480v RMOV Board 2E Normal Feeder Breaker
CDE 553, 2D RHR HEX
CDE 569, 2C RHR Pump Room Cooler
CDE 601, Loop I LPCI Injection Valve
CDE 602, 2D RMOV Board Impact on 2-FCV-74-53
CDE 612, 2D RHR Pump Room Cooler
CDE 613, RHR Pump D Room Cooler
CDE 621, 2B RHR HEX
CDE 666, 2-FCV-74-75
CDE 702, 1D RHR Suppression Pool Suction Valve

Corrective Action Program Documents

PER 127680, U2 and U3 MSPI data not matching basis documents
PER 129556, RHR Pump Demands reported in EPIX higher than expected
PER 136362, U1 and U2 RCIC pump has the pump driver listed as a motor instead of a turbine for MSPI report
PER 139177, Generic review of SQN PER 135288, MSPI Failure classification for Diesels
PER 141881, Duplicate devices exist in the CDE database for the RHRSW pumps
PER 142759, Disk stem separation of RHRSW Outlet Valves reported as MSPI failures
PER 144313, Diesel generator output breakers not setup correctly in CDE to support MSPI
PER 148995, Unit 1 MSPI Data Reporting Errors
PER 106084, Unit 1 Basis Document and CDE Initial Input
PER 127680, Basis Documents Not Matching Submittal
PER 129566, RHR Pump Demands
PER 139177, MSPI Failure Classification
PER 141881, Unit 3 Duplicate Devices for RHRSW Pumps
PER 142759, EPIX Report 385 Possible Revision
PER 106084, Unit 1 Basis Document and CDE Initial Input
PER 147593, RHR MSPI Data Entry Late, Internal to Licensee
PER 148995, Unit 1 MSPI Data Reporting Errors

Section 4OA2: Identification and Resolution of Problems

OPDP-1, Conduct of Operations, Revision 10
PER 148919, OWA Concept Does Not Contain Procedure Change Barriers
PER 151156, Low CCW Bearing Water Flow Alarms
PER 150240, Inactive Operator Workaround Carried as Active
Plan of the Day (POD) Report, Active Operator Workarounds, dated 8/18/08

Operations Department Level Performance Indicators, Operator Workarounds (OWAs) – Outage and Non-Outage for July 2008
Self-Assessment BFN-OPS-07-SS22, BFN Operator Work Arounds
EMPAC Select Focus Area Report for WO Related OWAs, dated 8/27/08

Section 4OA3: Event Follow-up

LER 05000296/2008-001, Unanticipated Auto-Start of Emergency Diesel Generators
PER 144272, 3B 4kv Unit Board De-energized when Attempting Transfer
PER 140121, CASA Logic Failed to Pick Up During 3D DG LAT
LER 05000296/2008-002-00, Main Steam Relief Valve As Found Setpoint Exceeded TS
PER 146189, MSRV As-Found Setpoint Results
BFE-13 ASME and ATWS Overpressurization Analysis With As-Tested MSRV Setpoint Data

Section 4OA5.2: TI 2515/176

Licensing Documents

Browns Ferry Nuclear Plant Unit 1 Technical Specifications and Bases
Browns Ferry Nuclear Plant Unit 2 Technical Specifications and Bases
Browns Ferry Nuclear Plant Unit 3 Technical Specifications and Bases
Browns Ferry Nuclear Plant Updated Final Safety Analysis Report

Procedures

0-SR-3.8.1.7(A), Diesel Generator A 24 Hour Run
0-SR-3.8.1.7(B), Diesel Generator B 24 Hour Run
0-SR-3.8.1.7(C), Diesel Generator C 24 Hour Run
0-SR-3.8.1.7(D), Diesel Generator D 24 Hour Run
3-SR-3.8.1.7(3A), Diesel Generator 3A 24 Hour Run
3-SR-3.8.1.7(3B), Diesel Generator 3B 24 Hour Run
3-SR-3.8.1.7(3C), Diesel Generator 3C24 Hour Run
3-SR-3.8.1.7(3D), Diesel Generator 3D 24 Hour Run

TVAN Calculations

Diesel Load Study for Unit 1 and 2 (EDQ005720020069), Rev. 12
Diesel Load Study for Unit 3 (EDQ3057920035), Rev. 33

Completed Surveillance Tests

0-SR-3.8.1.7(A), Diesel Generator A 24 Hour Run (08/30/03, 08/26/05, and 07/28/07)
0-SR-3.8.1.7(B), Diesel Generator B 24 Hour Run (08/29/03, 02/10/06, and 02/09/08)
0-SR-3.8.1.7(C), Diesel Generator C 24 Hour Run (11/10/01, 04/23/04, and 07/14/06)
0-SR-3.8.1.7(D), Diesel Generator D 24 Hour Run (01/02/04, 01/27/06, and 12/30/07)
3-SR-3.8.1.7(3A), Diesel Generator 3A 24 Hour Run (06/10/05, 05/13/07, and 11/23/07)
3-SR-3.8.1.7(3B), Diesel Generator 3B 24 Hour Run (08/09/03, 07/08/05, and 09/02/07)
3-SR-3.8.1.7(3C), Diesel Generator 3C 24 Hour Run (08/22/03, 09/16/05, and 03/01/08)
3-SR-3.8.1.7(3D), Diesel Generator 3D 24 Hour Run (08/23/03, 10/14/05, and 09/15/07)

Corrective Action Program Documents
PER 153890, Conflict in SR-3.8.1.7(A) for Specified Maximum Loading Requirements
PER 153896, Temporary Overcurrent Relay Exceeds EDG Load Limit
PER 153480, Numerous EDG Load Study Documentation Deficiencies
PER 152807, Minimum KVAR limit Not Maintained During 3D EDG 24 Hour Load Run

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
AOI	Abnormal Operating Instructions
CAP	corrective action program
CARB	Corrective Action Review Board
CFR	*Code of Federal Regulations*
CS	core spray
DCN	design change notice
EECW	emergency equipment cooling water
EDG	emergency diesel generator
EP	emergency preparedness
EPIP	Emergency Plan Implementing Procedure
EOI	Emergency Operating Instruction
FA	fire area
FZ	fire zone
GOI	General Operating Instruction
HPCI	high pressure coolant injection
LCOTR	limiting condition of operation tracking
LER	licensee event report
MR	maintenance rule
MSPI	mitigating systems performance index
MSRV	main steam relief valve
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OOS	out of service
OWA	operator workaround
PAR	protective action recommendation
PARS	publically available records
RBCCW	reactor building closed cooling water
PER	problem evaluation report
PI	performance indicator
PM	preventative maintenance
PMT	post maintenance testing
PORC	Plant Oversight Review Committee
RCIC	reactor core isolation cooling
RHR	residual heat removal
RHRSW	residual heat removal service water
RMA	risk management action
RTP	rated thermal power
SLC	standby liquid control
SPP	Standard Programs and Processes
SSC	structure, system, and/or component
TI	Temporary Instruction
TS	Technical Specification(s)
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
WO	work order