UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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August 15, 2005

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2005007

Dear Mr. Singer:

On July 16, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed a quarterly inspection period associated with recovery activities at your Browns Ferry 1 reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on August 1, 2005, with Mr. Jon Rupert and other members of your staff.

We previously informed you, in a letter dated December 29, 2004, of our plans for the transition of four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) to be monitored under the ROP baseline inspection program. Consequently, as of January 2005, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections. They will no longer be documented in the Unit 1 recovery quarterly integrated reports such as this one, but will be documented in the Unit 2 and 3 Integrated Quarterly Reports. Inspection Report 05000259, 260, 296 / 2005003, issued July 29, 2005, is the most recent Unit 2 and Integrated Quarterly Report which contains Unit 1 ROP inspection of this type.

This inspection examined activities conducted under your Unit 1 license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license and also with fulfillment of Unit 1 Regulatory Framework Commitments. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of this inspection, no violations or findings of significance were identified. Our inspection results indicated that your staff was successful in their coordination of the refueling cavity fill milestone. Overall, we have found your staff's recovery activities to be well controlled during this inspection period.

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/

Stephen J. Cahill, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket No. 50-259
License No. DPR-33

Enclosure: Inspection Report 05000259/2005007
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2005007
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	April 17 - July 16, 2005
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 E. Christnot, Resident Inspector C. Patterson, Senior Resident Inspector, Farley (Section E8.2) L. Mellen, Senior Reactor Inspector (Section E8.1) C. Peabody, Reactor Inspector (Section E8.1)
Approved by:	Stephen J. Cahill, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2005007

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 recovery project. This report covered a 3-month period of resident inspector inspection. In addition, NRC staff inspectors from the regional office conducted inspections of Unit 1 Recovery Special Programs in the areas of open inspection items. The inspection program for the Unit 1 Restart Program is described in NRC Inspection Manual Chapter 2509. Information regarding the Browns Ferry Unit 1 Recovery and NRC Inspections can be found at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/bf1-recovery.html. Per the Partial Cornerstone Transition letter from the NRC to TVA dated December 29, 2004, four Reactor Oversight Process (ROP) Cornerstones (Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection) are monitored under the ROP baseline inspection program as of January 2005. Consequently, inspections for these cornerstones are integrated with Unit 2 and 3 ROP baseline inspections and are no longer documented in the Unit 1 recovery quarterly integrated reports such as this one, but in the Unit 2 and 3 Integrated Quarterly Reports.

Inspection Results - Engineering

- The inspector's review of modification design packages associated with six planned design changes concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The DCNs adequately addressed the changes needed to restore Unit 1 to current requirements. (Section E1.1)

- Modification installation activities associated with seven permanent plant design changes were observed and found to be performed in accordance with the documented requirements. (Section E1.1)

- Activities associated with five temporary alterations which affected the Reactor Protection System, Reactor Water Cleanup System, and Primary/Secondary Containment Isolation System did not cause any significant impact on the operability of equipment required to support operations of Units 2 and 3. (Section E1.2)

- The licensee's System Return to Service activities continued to be performed in accordance with procedural requirements. Any system deficiencies were identified and appropriately addressed by the licensee's corrective action program. (Section E1.3)

- Implementation of restart testing activities was acceptable. Some test deficiencies were identified during performance of testing but were appropriately documented under the licensee's corrective action program. Licensee processes were effective at identifying problems before components were placed in service. (Section E1.4)

- Electrical cable installation activities were performed in accordance with documented requirements. (Section E1.5)

- Based on review of six Engineering Work Requests the inspectors concluded that the Unit 1 engineering organization had provided adequate guidance and technical information to support activities associated with reactor vessel fill and removal of the reactor cavity gates. (Section E1.6)

Inspection Results - Maintenance

- The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which do not require design changes to support the Unit 1 Restart. (Section M1.1)

Inspection Results - Operations

- The inspectors determined that activities associated with the filling of the RPV and reactor cavity were conducted in a safe and controlled manner. (Section O8.1.1)

- The reactor cavity gates were removed in a safe and controlled manner and the gates were placed in the proper storage racks within the spent fuel storage pool. The ongoing gate removal activities did not affect maintaining normal fuel pool level nor impact the ability to adequately cool irradiated fuel in the Unit 1 fuel storage pool. (Section O8.1.2)

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REPORT DETAILS

Summary of Plant Status

Unit 1 has been shut down since March 19, 1985, and has remained in a long-term lay-up condition with the reactor defueled. The licensee initiated Unit 1 recovery activities to return the unit to operational condition following the TVA Board of Directors decision on May 16, 2002. During the current inspection period, reinstallation of plant equipment and structures continued. Recovery activities include ongoing replacement of piping in the reactor coolant, reactor water cleanup, and feedwater systems; reinstallation of balance-of-plant piping and turbine auxiliary components; installation of small and large bore pipe supports; and installation of new electrical cables, conduits, and conduit supports. Limited system return to service (SRTS) activities continued during this reporting period. The licensee completed reactor vessel and reactor cavity fill, removed the reactor cavity gates, and initiated inspection of reactor vessel internals during the reporting period.

II. Engineering

E1 Conduct of Engineering

E1.1 Permanent Plant Modifications (71111.17, 37550, 37551)

a. Inspection Scope

The inspectors reviewed planned Design Change Notice (DCN) packages to the Unit 1 120-Volt (V) alternating current (AC) System, Feedwater System, Reactor Core Isolation Cooling (RCIC) System, Main Steam System, and various instrumentation located in the Unit 1 Auxiliary Instrument Room. In addition, the inspectors reviewed the planned modifications of the Unit 3 Emergency Core Cooling System (ECCS) Accident Signal Logic. The DCN was developed by the Unit 1 engineering organization and the recommended changes to the Unit 3 logic were due to the differences in the Unit 3 and the Unit 1 and Unit 2 ECCS Accident Signal Logics. The modifications to the Unit 1 and Unit 2 ECCS Accident Signal Logics were previously reviewed and documented in Inspection Reports 05000259/2004-09 and 2005-06. The inspectors reviewed criteria in licensee procedures Standard Program and Process (SPP)-9.3, Plant Modifications and Engineering Change Control; SPP-7.1, Work Control Process; SPP-8.3, Post-Modification Testing; and SPP-8.1, Conduct of Testing, to verify that risk-significant plant modifications were developed, reviewed, and approved per the licensee's procedure requirements.

The inspectors reviewed and observed ongoing permanent plant modification activities for the Core Spray System, Feedwater System, plant annunciators, relocation of control room hand switches, Vacuum Priming, 480-V AC, and Raw Cooling Water (RCW). The inspectors evaluated the adequacy of the modifications and observed field work to verify that the design basis, licensing basis, and Technical Specification (TS) requirements for the systems had not been degraded as a result of the modifications.

b. Observations and Findings

b.1 DCN Package Review

DCN 51017

The inspectors reviewed the Unit 3 permanent plant modification DCN 51017, ECCS Accident Signal Logic, Systems 68, 74, 75, and 82. The intent of this DCN was to implement the modifications recommended for the Unit 3 ECCS, the Residual Heat Removal (RHR) initiation logic, and the Core Spray initiation logic. The recommended changes were due to the differences in the Unit 3 ECCS Accident Signal Logic and the Unit 1 and Unit 2 ECCS Accident Signal Logics. The differences were the results of the implementation of DCN 51016, Unit 1 ECCS Accident Signal Logic, and DCN 51018, Unit 2 ECCS Accident Signal Logic. DCNs 51016 and 51018 were previously reviewed and documented in Inspection Reports 05000259/2004-09 and 2005-06. Among the scheduled modifications included in this DCN were the sparing of Divisional cables between control room panel 3-9-3 and panel 3-9-32, between panel 3-9-32 and panel 3-9-33, and between control room panel 3-9-3 and panel 3-9-33; remove and install HGA and HFA relays, indicating lights, and hand switches in panel 3-9-32; removal Agastat relays and deletion of internal wiring associated with RHR redundant pump start logic and RHR Low Pressure Coolant Injection (LPCI); and deletion, installation, and modification of internal wiring in panels 3-9-32 and 3-9-33 associated with the new inhibit hand switches and relays.

The DCN also provides for disconnection of linkages and abandonment of actuators associated with RHR DIV I testable check valve, 3-FCV-74-54; deletion of DIV I cables in DIV II logic panel 3-9-33; deletion, installation, and modification of internal wiring, hand switches, and indicating lights on Control Room panel 3-9-3; disconnection of linkages and abandonment of actuators associated with RHR DIV II testable check valve, 3-FCV-74-68; and deletion of DIV II cables in DIV I logic panel 3-9-32.

DCN 51081

The inspectors reviewed the Unit 1 permanent plant modification DCN 51081, Modifications to Unit 1 Auxiliary Instrument Room. The intent of this DCN was to implement various instrumentation modifications associated with System 1, Main Steam System; System 63, Standby Liquid Control (SLC) System; and System 64, Primary Containment Isolation System (PCIS) Logic. Stage 1 to this DCN will modify cables to provide functional redundancy and separation; add various control switches, relays and indicating lights; add a time delay relay for each train of the Automatic Depressurization System (ADS) to bypass High Drywell pressure initiation after low low (level I) reactor pressure vessel (RPV) water level signal occurs. Stage 2 will modify various SLC system components in Panel 1-9-19. Stage 3 will modify various PCIS components in Panels 1-9-3, 1-9-15, 1-9-17, 1-9-18, 1-9-29, 1-9-32, 1-9-32, 1-9-42, 1-9-43, 1-9-81, 1-9-82, 1-9-87, 1-9-88, and 1-25-32. Stage 4 will modify various PCIS components in Panel 1-9-36A. Stage 5 will modify various PCIS components in Panel 1-9-36B.

DCN 51085

The inspectors reviewed the Unit 1 permanent plant modification DCN 51085, 120-V AC - Control Bay, Systems 252, 253, and 256. The intent of this DCN was to implement modifications recommended for 120V AC systems in the Control Bay as follows: System 252, Unit Preferred 120V AC; System 253, 120V AC Instrumentation and Control Power Supply; and System 256, ECCS Inverters. Among the scheduled modifications associated with this DCN were the following:

- System 252, Unit Preferred 120V AC. DCN 51085 provides for the disconnection and removal of the Unit Preferred Motor Generator (MG) set. Specifically, this DCN will disconnect all AC and direct current (DC) cables, conduits, and junction boxes connected to the MG set; install new solid state controlled 35-kVA inverter in the space formally occupied by the MG set; pull back, delete cable, conduit, supports and junction boxes, and abandon cable, as necessary, to install the new inverter; install and connect new AC and DC cables, conduits, and junction boxes.

- System 253, 120V AC Instrumentation and Control Power Supply. DCN 51085 provides for the changing of 36 circuit breakers from General Electric (GE) Type TEF to GE Type TED and with trip set points at 15 amps or 20 amps as applicable; and changing of selected existing breaker set points and fuses for fuse and breaker coordination.

- System 256, ECCS Inverters. DCN 51085 provides for removal of old inverters, disconnection and removal of all interconnecting cables, installation of new solid state controlled 5-kVA inverters, 1-INVT-256-01 and 1-INVT-256-02; termination of the new cables at the inverters; and removal of abandoned cables, conduits, and junction boxes to make room for the new inverters at the DIV I ECCS Analogue Trip Units (ATUs) and the DIV II ECCS ATUs.

Also included in this DCN were work activities involving the +/- 24 Volt DC Neutron Monitoring Battery Chargers, 1-CHGD-283-A1-1, -A2-1, -B1-1, and -B2-1, the Reactor Protection System regulating transformers, and the power supplies to the ECCS ATU panels.

DCN 51066

The inspectors reviewed the Unit 1 permanent plant modification DCN 51066, Reactor Feedwater - Inside Drywell, System 3. The intent of this DCN was to implement the mechanical modifications recommended for the reactor feedwater system inside the drywell. Scheduled modifications associated with this DCN include installation of large bore piping supports; snubber testing; and installation of drywell platform steel at Elevation 584' 9 1/2".

DCN 51069

The inspectors reviewed the Unit 1 permanent plant modification DCN 51069, Main Steam - Inside the Drywell, System 1. The intent of this DCN was to implement the mechanical modifications recommended for the main steam system inside the drywell. Scheduled modifications associated with this DCN included installation of large bore piping supports for Main Steam and High Pressure Coolant Injection Systems; snubber testing for Main Steam and High Pressure Coolant Injection Systems; and modification of miscellaneous steel support framing.

DCN 51220

The inspectors reviewed the Unit 1 permanent plant modification DCN 51220, Unit 1 Electrical - System 71, RCIC. The intent of this DCN was to implement various electrical modifications associated with RCIC System to reroute cables, de-terminate/re-terminate various cables associated with motor-operated-valves (MOVs) being replaced, replace EQ components and internal wiring in various panels and modify RCIC logic for auto restart upon low RPV water level. This DCN also eliminates the electronic overspeed trip function and replaces obsolete GE time delay relays.

b.2 Implementation of Permanent Plant Modifications

DCN 51200

The inspectors reviewed the Unit 1 permanent plant modification DCN 51200, Reactor Building Mechanical, Core Spray - System 75. The intent of this DCN was to replace various components in the Unit 1 Core Spray System, including piping sections, valves, valve actuators, and relief valves. In addition, Stage 1 of this DCN included the replacement of the ECCS suction strainers with new strainers with a strainer size and design which is intended to provide acceptable head loss under all plant conditions. The new strainers were designed by GE and similar to those previously installed on Units 2 and 3 under DCNs T40210A and T40211A. The inspectors reviewed the completed DCN package and a video of the underwater placement and torquing of the hold down bolting for the new replacement strainers.

DCNs 51163 and 51231

The inspectors reviewed the Unit 1 permanent plant modification DCNs 51163 and 51231, Reactor Feedwater - System 3. The intent of these DCNs was to replace the four feedwater containment isolation check valves as part of the licensee's Cobalt Reduction Program. The valves to be replaced included the two inboard isolation check valves, 1-CHK-003-0558 and 1-CHK-003-0572, and the two outboard check valves, 1-CHK-003-0554 and 1-CHK-003-0568. The inspectors observed selected ongoing work activities for the inboard check valves located in the Unit 1 drywell. In addition, the inspectors reviewed Work Orders (WOs) 03-0044727-074 and 03-0044727-075 which dealt with the outboard check valves located in the steam tunnel. The inspectors also observed licensee controls to ensure that secondary containment integrity was maintained throughout the ongoing work activities. In addition, the inspectors reviewed Engineering Work Request (EWR) 04-CEB-003142 which had been issued to clarify structural requirements for implementation of this work activity. The EWR provided the engineering basis for allowing the containment breech with a known measured value for secondary containment leakage rate without further analysis or the need for installation of temporary restraints. The inspectors noted that WOs 03-0044727-074 and 03-0044727-075 included precautionary measures to ensure that requirements of the EWR were observed during the work progression and that system engineering perform leakage testing prior to removal of the check valves.

<u>DCN 51100</u>

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51100, Modifications to Control Room Panel 1-9-22, issued to address identified Human Engineering Deficiencies (HEDs). Activities observed by the inspectors included relocation of selected control switches for System 2, main condensate storage and transfer; control switches for System 24, raw cooling water; control switches for System 32, control air; and control switches for System 67, EECW.

<u>DCN 51107</u>

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51107, Annunciator Upgrade, System 55, issued to change the analog alarm system to a digital system. Activities observed by the inspectors included replacement of the alarm window 1-XA-55-20A for panel 1-9-20, which affected the alarms for systems such as emergency equipment cooling water, control air, raw cooling water, traveling screens for the condenser circulating water, and raw service water; replacement of the alarm window 1-XA-55-5B for panel 1-9-5, which affected the alarms for systems such as reactor protective system, main steam, drywell, rod worth minimizer, and standby liquid control; and replacement of the alarm window 1-XA-55-3D for panel 1-9-3, which affected the alarms for systems such as residual heat removal, core spray, reactor core isolation cooling, and reactor water cleanup.

<u>DCN 51216</u>

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51216, 480-V Distribution - Reactor Building, System 57-4. Activities observed by the inspectors included installation of load shed cables to 480-V reactor motor-operated valve (RMOV) 1C, replacement of normal feeder cable to 480-V RMOV 1B, replacement of alternate feeder cable to 480-V RMOV 1A, and installation of load shed cables to 480-V RMOV 1B.

<u>DCN 51090</u>

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51090, 480-V Distribution - Control Bay, System 57-4. Activities observed by the inspectors included the removal and installation of new breakers, new fuses, and new thermal overloads in various circuit breakers and cubicles in RMOV Boards 1A and 1B. Among the activities associated with Board 1A were the following: Cubicle 17C, main steam line drain inboard isolation valve; cubicle 18E, raw service water storage tank isolation control valve; breaker 12C, EECW north head sectional valve; and cubicle 18C, reactor protective system motor-generator set 1A. Among the activities associated with Board 1B were the following: Breaker 18E2, main turbine bearing lift pumps; and breaker R10F, primary containment atmospheric monitor system analyzer sample return pump.

c. Conclusions

The inspector's review of modification design packages associated with six DCNs concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The DCNs adequately addressed the changes needed to restore Unit 1 to current requirements.

Modification activities associated with seven ongoing permanent plant modifications were performed in accordance with the documented requirements.

E1.2 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed licensee procedure SPP-9.5, Temporary Alterations. The inspectors also reviewed the following temporary alterations: Temporary Alteration Configuration Form (TACF) 1-84-029-069, 1B Reactor Water Clean-Up (RWCU) Pump Motor; TACF 1-85-032-099, Unit 1 Reactor Protective System (RPS); TACF 1-88-002-69, RWCU Non-Regenerative Heat Exchanger; TACF 1-2002-002-069, RWCU System Piping; and TACF 1-05-002-64D, Primary/Secondary Containment Isolation Control Logic Circuitry. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design bases documentation and reviewed selected completed work activities of the system to verify that installation was consistent with the modification documents and the TACF. In addition, special emphasis was placed on the potential impact of these temporary modifications on operability of equipment required to support operations of Units 2 and 3.

b. Observations and Findings

The inspectors reviewed and observed selected removal activities for the temporary alterations involved with the primary/secondary containment isolation logic circuitry, RPS System, and RWCU System. The temporary alterations reviewed and observed were as follows:

- TACF 1-84-029-069, 1B RWCU Pump Motor, System 69, was initiated to disable Relay 49X, the thermal protective relay for the 1B RWCU pump motor. The relay was disabled by lifting a lead in the 1B 480-V AC Shutdown Board, compartment 8B. The relay was actuated by thermotectors located internal to the motor. The original motor was replaced with a motor that did not contain thermotectors. To prevent the inadvertent operation of the protective relay, and thereby inhibit operation of the pump motor, the lead was lifted. The absence of thermotectors on the replacement motor decreased the possibility of preventing motor failure due to temperature. The RWCU pump is not safety-related and a failure would not result in damage to equipment important to safety. DCN 51194, RWCU - Reactor Building, Piping and Pumps, was issued for Unit 1 restart and eliminated the need for this TACF. The inspectors reviewed the associated DCN and TACF removal activities.

- TACF 1-85-032-099, Unit 1 Reactor Protective System (RPS), System 99, was initiated in 1985 to disable the full scram function of the RPS. The initial TACF used jumpers equipped with alligator clips to disable the function. More recently, Stage 2 of DCN 51206, Control Rod Drive Mechanical - Reactor Building, was initiated to perform modifications involving the scram discharge volume instrumentation and piping. A total of 14 WOs was issued to implement Stage 2. Post-Issuance Change (PIC) 60321 to the DCN stated, in part, the following: TACF 1-85-032-099 must remain in place until all modifications in Stage 2 are completed to prevent an inadvertent initiation of a Full Scram. WO 04-717791-00 was issued to remove all alligator clips and replace with jumpers equipped with termination lugs which were landed under terminal screws. This change in scope of the temporary alteration was performed in order to reduce the use of alligator clips and make the disabling of the function more reliable. The inspectors reviewed the associated DCN and inspected the newer temporary alteration configuration.

- TACF 1-88-002-69, RWCU Non-Regenerative Heat Exchanger, System 69, was initiated to disable the heat exchanger outlet high temperature indicating switch, 1-TSI- 69-11. When the switch failed, and with Unit 1 in a long term layup, a decision was made not to replace or repair the switch. The switch was electrically removed from the circuit by the use of jumpers. DCN 51235, RWCU-Reactor Building, Mechanical/Electrical, was issued for the Unit 1 restart and replaced the heat exchanger outlet high temperature indicating switch. The inspectors reviewed the DCN associated with closure of this temporary alteration.

- TACF 1-2002-002-069, RWCU System Piping, was initiated to install piping connections for chemical decontamination cleaning of the system piping prior to removal and replacement. The cleaning was successful and the radiation levels were significantly reduced for As Low As Reasonably Achievable (ALARA) considerations. The temporary alteration was installed in 2002 and was removed since it was no longer needed. The system piping was removed and replaced with new piping. This TACF was closed. The inspectors reviewed and observed portions of the removal of the TACF and activities associated with returning the system to normal status.

- TACF 1-2005-002-64D, Primary/Secondary Containment Isolation Control Logic Circuitry, System 64D, was initiated to de-energize control power to selected Unit 1 inboard and outboard Group 6 isolation valves. This activity involved lifting electrical leads in Panels 1-9-42 and 1-9-43. The intent of the TACF was to minimize the number of secondary containment isolations that would occur during DCN implementation on the valves and the control circuits. Among the valves affected were the following: System 84, valve 1-FCV-84-20, Drywell Vent to Standby Gas Treatment; System 76, valves 1-FCV-76-17, Drywell and Torus Nitrogen Makeup Inlet, 1-FCV-76-18, Drywell Nitrogen Makeup Inboard Isolation, and 1-FCV-76- 24, Nitrogen Purge Supply Inlet; and System 64, 1-FCV-64-19, Torus Air Purge, 1-FCV-64-29, Drywell Vent, and 1-FCV-64-32, Torus Vent. The PCIS control logic circuitry remained functional to initiate secondary containment isolation for high Refuel Floor or Reactor Zone radiation conditions. The inspectors observed selected activities associated with installation of this temporary alteration.

c. Conclusions

The inspectors determined that activities associated with the five temporary alterations which affected the RPS, RWCU, and Primary/Secondary Containment Isolation System did not cause any significant impact on the operability of equipment required to support operations of Units 2 and 3. No violations or deviations were identified.

E1.3 System Return to Service Activities (37550)

a. Inspection Scope

The inspector reviewed and observed portions of the licensee's SRTS activities. The SRTS activities were performed in accordance with Technical Instruction 1-TI-437, System Return to Service Turnover Process for Unit 1 Restart.

The inspectors focused on portions of two major systems Fuel Pool Cooling (FPC), System 78, and RWCU, System 69. These systems were originally intended to be functional prior to removal of the gates between the fuel pool and the reactor cavity. However, the RWCU System did not complete the System Pre-Operability Checklist (SPOC) I process during the reporting period. System 78, Fuel Pool Cooling (FPC), and System 53, Demineralizer Backwash Air, were returned to Operations in a fully operable status. Operations assumed total control of both systems.

b. Observations and Findings

SRTS activities continued during the reporting period. The SRTS process consisted of three parts: System Plant Acceptance Evaluation (SPAE), which consists of verification of design changes, engineering programs analysis, drawings, calculations, corrective action items, and licensing issues; SPOC I, which consists of the completion of items required for system testing; and SPOC II, which consists of the completion of system testing and the completion of items that affect operational readiness. The SRTS activities reviewed and observed by the inspectors included system testing on System 78, Spent Fuel Pool Cooling, System 53, Demineralizer Backwash Air, and System 69, RWCU. During this report period, the SPAE and SPOC II processes were completed for Demineralizer Backwash Air and Spent Fuel Pool Cooling. The inspector reviewed and observed portions of the licensee's SRTS activities for the following:

- System 53, Demineralizer Backwash Air, completion of the SPAE process and completion of the SPOC Phase II process

- System 78, Spent Fuel Pool Cooling, completion of the SPAE process and completion of the SPOC Phase II process

- System 69, RWCU, SPAE activities and SPOC activities

- System 79, Fuel Handling, SPOC II and SPAE activities

- System 70, Reactor Building Closed Cooling Water, SPOC I and SPAE activities

Activities observed included periodic meetings to discuss the SRTS status of various systems, which included the status of the SPOC I checklists, the status of the SPAE process, and the status of the SPOC II checklists. The activities also included observation of licensee walkdowns of portions of plant systems and review of PERs initiated during the SRTS process. Any PERs were adequately addressed by the Unit 1 Restart corrective action program.

c. Conclusions

SRTS activities continued to be performed in accordance with procedural requirements. Any system deficiencies were identified and appropriately addressed by the licencee's corrective action program.

E1.4 System Restart Testing Program Activities (37551)

a. Inspection Scope

The inspectors reviewed and observed the on-going activities associated with the Restart Test Program (RTP). The RTP item reviewed and observed consisted of Post-Modification Test Instructions (PMTI), Operating Instructions (OI), Surveillance Instructions (SI), and WOs. The inspectors also reviewed selected corrective action documents initiated during the RTP process to document test deficiencies or other problems.

b. Observations and Findings

b.1 Observation of Testing Activities

The following PMTIs were developed and approved to test portions of the associated DCNs. The areas involved in the tests were the control room, fuel pool cooling, RWCU valve and heat exchanger rooms, and the refueling floor. Pre-test briefings were held, assignments were made, and communications were established prior to performance of testing. The inspectors observed and reviewed portions of the following testing:

- 1-PMTI-BF- 51090-S57-64+S79 (Stages 57 through 64 and Stage 79), tested portions of DCN 51090, 480 VAC Control Bay - Electrical, System 57-4, Stage 61 and Stage 62. These sections of the PMTI performed functional tests of stage 61 and stage 62 of DCN 51090. Stage 61 involved the load shed time delay relay for the reset coil and the inhibit control circuit for the Unit 2 drywell blower fan 2B-1. The time delay relay for fan 2B-1 was verified to be at 40 seconds. Stage 62 involved the load shed time delay relay for the reset coil and the inhibit control circuit for the Unit 2 drywell blower fan 2B-2. The time delay relay for fan 2B-2 was verified to be 50 seconds. Test Deficiencies (TD) were identified and adequately resolved.

- 1-PMTI-51100-STG06, Stage 6 and Stage 7, which are included in the Control Room Design Review (CRDR) program, tested portions of DCN 51100. The two tests verified that equipment in the Demin Water System which had been moved from Panel 1-9-20 to Panel 1-9-22 were operable. Equipment moved included hand switch 0-HS-2-155A, Demin Water Transfer Pump A; hand switch 0-HS-2-159, Demin Water Head Tank Inlet Valve; level indicator 0-LI-2-153A, Demin Water Storage Tank Level; and hand switch 0-HS-2-154A, Demin Water Transfer Pump B. No TDs were identified.

- 1-PMTI-51203-STG02, tested portions of DCN 51203, Fuel Pool Cooling Reactor Building - Mechanical, System 78, Stage 2. This stage removed the interlock between valve 1-FCV-78-7, Fuel Pool Header Drain to Main Condenser, and alarm 1-FIS-78-5, Fuel Pool Gate Leakage. Prior to this modification the valve could not be operated from local control panel 1-25-16 unless 1-FIS-78-5 was in the alarm condition. This required that a temporary alteration under a TACF be installed during refueling operations in order to operate the valve with the alarm not present. Testing in accordance with this PMTI verified that the valve operated as required from panel from Panel 1-25-16. No TDs were identified.

- 1-PMTI-023-052, Stage 25, which is included in the CRDR program, tested portions of DCN 51904. The test verified that hand switch 0-HS-23-88-A/1, RHRSW Pump B3, after being re-located on Panel 1-9-3 was operable. The test verified that the start, stop, and motor trip-out contacts associated with the hand switch operated as required. The test simulated a motor trip-out condition by depressing the local trip push button on the 4160-V supply breaker while the pump was running. No TDs were identified.

- 1-PMTI-002-011, Stage 5, which is included in the CRDR program, tested portions of DCN 51100. The test verified that equipment related to Condensate Storage Tanks 4 and 5, part of the Condensate Storage and Supply System, moved from Panel 1-9-20 to Panel 1-9-22 were operable. Equipment moved included hand switch 0-HS-2-269A, Condensate Storage Tank 4 Discharge Isolation Valve; hand switch 0-HS-2-177, Condensate Storage Head Tank Isolation Valve; level indicator 0-LI-2-271, Condensate Storage Tank 5 Level; and hand switch 0-HS-2-273A, Condensate Storage Tank 5 Discharge Isolation Valve. No TD's were identified.

- Operating Instruction 1-OI-69, Reactor Water Cleanup System, Section 6.5, Filter Demineralizer Manual Backwash and Precoat, in conjunction with WO 05-713937-00, were used to perform a valve test of the various air-operated control valves for the RWCU demineralizer to verify their as-found condition. Several deficiencies were identified and adequately resolved.

- Surveillance Instruction 1-SI-3.3.3, ASME Section XI System Pressure Test of Fuel Pool Cooling System (ASME Section III Class 3), was performed to satisfy the ten year hydrostatic requirements for portions of the System 78, FPC system. The test was specified by the ASME Section XI Code, 1995 Edition, 1996 Addenda for Class 3 components and Code Case –498-4. This surveillance instruction was listed in procedure 0-TI-364, ASME Section XI System Pressure Tests, which implemented the ASME pressure testing program of TRM 3.4.3 and TSR 3.4.3.1. The scope of the test was to inspect piping and components for leaks in accordance with N-VT-4 after the FPC system had been in service for at least four hours. The dryer/separator pool and the reactor cavity well were required to be filled. The system was aligned for the test using procedure 1-OI-78, Fuel Pool Cooling System Operating Instruction, with no deficiencies identified.

b.2 <u>Review of Test Deficiencies</u>

The test directors identified and documented TDs during the performance of ongoing testing. For each example reviewed the TDs were addressed by issuing applicable WOs, punchlist items, and/or PERs as applicable. The inspectors reviewed the following TDs:

- 1-PMTI-BF-51090-S57-64+S79, during the performance of the test for Stage 61 a TD was identified at Step 6.8.12, Verify the Fan Unit 2B-1 breaker closes. The power supply breaker for the drywell blower failed to close as required by the PMTI. It was determined that control power for the breaker was not available. Power was restored to the breaker and the test was continued.

- 1-PMTI-BF- 51090-S57-64+S79, during the performance of the test for Stage 61 a TD was identified at Step 6.8.17, Verify, after a 40-second delay, that Fan Unit 2B-1 breaker closes by ensuring the red indicating lamp is illuminated. The red indicating lamp failed to illuminate as required by the PMTI step. WO 04-720414-17 was initiated to troubleshoot the problem and a design wiring error was discovered. PIC 63940 was issued to correct the design error.

- 1-PMTI-BF-51090-S57-64+S79, during the performance of the test for Stage 62 a TD was identified in Step 6.9.17, Verify, after a 50-second delay, that Fan Unit 2B-2 breaker closes by ensuring the red indicating lamp is illuminated. The red indicating lamp failed to illuminate as required by the PMTI step. WO 04-720414-18 was initiated to troubleshoot the problem and a design wiring error was discovered. PIC 63941 was issued to correct the design error. As a result of the design wiring error, PER 79750 was issued to document the design wiring errors for the TD in the Stage 61 and Stage 62 testing.

- During performance of Operating Instruction 1-OI-69, Reactor Water Cleanup System, several deficiencies were identified involving valves not functioning, valves functioning with indicating lights not operating, valves not meeting the ten-second open/close requirements, and valves not capable of being adjusted to meet the open/close requirements. Various WOs were initiated to correct these deficiencies.

b.3 <u>Review of Corrective Action Documents</u>

The inspector also reviewed selected PERs initiated during the RTP process. The following PERs were reviewed:

- PER 79750 was initiated to document the TD's identified during the performance of 1-PMTI-BF-51090-S57-64+S79, Stage 61 and Stage 62. The TDs involved design wiring errors. Unit 2 Drywell Blower 2B-1 failed to restart as required by the PMTI, after the 40-second time delay relay timed out. PIC 63940 was issued to change the applicable blower design documents. Blower 2B-2 failed to restart after the 50-second time delay relay timed out. PIC 63941 was issued to change the applicable blower design documents.

- PER 83861 was initiated to document that while performing leak rate tests on Core Spray valves 1-SHV-75-54A and 1-SHV-75-54B the label for the 54A valve was incorrectly located on the 54B valve, and the 54B label was located on the 54A valve. The labels were removed and the error corrected.

- PER 84205 was initiated to document that a failure mode existed in the Programable Logic Controls (PLCs) used in the digital alarm system for the spent fuel cooling system. This failure mode was determined to be the same as a previously identified failure mode in the analog system used in Units 2 and 3. The operators were aware of this failure mode.

b.4 Baseline Test Requirements Document Review

The inspectors reviewed 1-BFN-BTRD-079.002, System 79, Fuel Handling and Storage System, Revision 2. This Baseline Test Requirements Document (BTRD) was initially issued to identify that the functional tests for System 79 were consistent with the system modes required to support safe shutdown as identified in the Safe Shutdown Analysis. The BTRD was revised and Revision 2 was issued to document test exceptions in that System Mode 79-01, provide safe fuel handling using the refuel bridge; and System Mode 79-02, provide interlocks to the Control Rod Drive (CRD), System 85, during fuel movement, could not be tested under current plant conditions. Restart test surveillance requirements procedure 1-SR-3.9.1.1, Refueling Interlocks Functional Test, could not be performed until proper interfaces with several other systems are functional and the systems are SPOC I status. The systems that must be functional are the Reactor Protection System (RPS), Reactor Manual Controls System (RMCS), Rod Position Indicating System (RPIS), sub-systems of the CRD System, applicable Control Room Annunciators. Also required are the applicable Control Room switches and controls for the systems per applicable Control Room Design Review modifications. This revision will allow System 79 to be placed in the SPOC II status with those documented testing exceptions. These two modes of System 79 will be tested when all testing prerequisites are satisfied.

c. Conclusions

Implementation of restart testing activities was acceptable. Some test deficiencies were identified during performance of testing, but were appropriately documented under the licensee’s corrective action program. Since their deficiencies were detected by licensee processes developed explicitly for that purpose and before components were put in service, the deficiencies did not constitute violation of regulatory requirements.

E1.5 Special Program Activities - Cable Installation and Cable Separation (37551)

a. Inspection Scope

The inspectors continued to observe and/or review the licensee’s activities associated with the installation of electrical cables. The installation activities were controlled by modifications WOs and licensee procedures. Among the procedures were the following: Modification and Addition Instruction (MAI) 1.3, General Requirements for Modifications; MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts Units 1, 2, and 3; MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts Units 1, 2, and 3; and MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control.

b. Observations and Findings

The licensee continued to perform limited cable installation activities during this report period. These were mostly power distribution cables and load shed cables. The majority of these activities were part of DCN 51216, Electrical 480-V Distribution System - Reactor Building. Activities observed or reviewed included the following:

- WO 03-001001-050, replace normal feeder cable for 480-V RMOV Board 1A
- WO 03-001001-052, replace normal feeder cable for 480-V RMOV Board 1B
- WO 03-001001-070, replace load shed cables for Load Shed Panel 0-PNL-25-44A-11
- WO 03-001001-071, replace load shed cables for Load Shed Panel 0-PNL-25-44B-11

During the above cable installation activities, the inspectors observed that the three-phase two-bolt hole lugs on the ends of the replacement cable for the normal feeder to the 480-V RMOV Board 1A, WO 03-001001-050, did not match up with the two-bolt hole bus terminal lugs. This DCN had increased the size of the cable to 300 million circular mils (MCMs) which resulted in a larger terminal lug size. However, MAI 3.3, Section 6.3 E, allowed for the installation of bus terminal bolt lug extensions for bolting in the new cable. The extensions were installed and the cable was terminated. The inspector concluded that this was an acceptable method to address the difference in lug sizes.

c. Conclusions

Electrical cable installation activities were performed in accordance with documented requirements.

E1.6 Review of Engineering Work Requests to Support RPV Fill and Removal of Reactor Cavity Gates (37551)

a. Inspection Scope

The inspectors reviewed EWRs associated with the RPV fill to 603 inches above RPV zero, flood up of the reactor cavity to fuel pool level, and preparations for removal of the spent fuel pool to reactor cavity gates. This review was performed to ensure that these planned Unit 1 recovery activities would not degrade the ability of the fuel storage pool to maintain the irradiated fuel in the pool covered and adequately cooled.

b. Observations and Findings

The licensee's engineering organization issued EWRs to provide guidance and technical information from the various engineering branches to support the license's planned activities in this area. The information provided in the EWRs was not considered as a design change, functional evaluation, 10 CFR 50.59 review, condition adverse to quality, 10 CFR 72.48 review, or change to the USFAR. The EWRs were written to obtain guidance and technical information from engineering such as system boundaries for valve closures, temporary system hook ups, supports for installed piping with attachments, spring pipe hangers, temporary supports for installed piping, snubbers, and permanent supports for installed piping. The following six EWRs were reviewed:

EWR 04MEB068199

This EWR from the Mechanical Engineering Branch provided system piping boundaries for RPV fill and cavity flood up with the fuel pool gates still installed. It also contained a summary of relevant engineering documents associated with the reactor fill activities. This EWR consisted of five attachments:

- Attachment A, Reactor Fill Mechanical Scope, listed 33 specific items with the applicable design output documents and associated modification. Examples of items listed included replacement of reactor recirculation piping in the drywell, DCN 51045; replacement of recirculation inlet, outlet, and jet pump nozzle safe ends, DCN 51045; replacement of RHR piping within the fill boundary, DCN 51151; replacement of Core Spray System piping and safe ends within the fill boundary, DCN 51152; replacement of feed water valve 1-FCV-3-188A and operator, DCN 51163; and replacement of main steam valve 1-FCV-1-55 and operator, DCN 51143.

- Attachment B, Fill Boundary Sense Line Modifications, listed the reactor vessel nozzle connections, root valves, drywell penetrations, instrument/component, design output documents, and applicable DCN.

- Attachment C, Pipe Supports, Part 1, listed a total of 265 supports by plant system, support number, calculation number, DCN, and support drawing number. Part 2 of Attachment C listed a total of 22 pipe supports with integral attachments by location, support number, plant system, and DCN.

- Attachment D, Reactor Fill Boundary Drawing, gave a visual boundary representation, which showed the piping, valves, and other components affected by the RPV fill.

- Attachment E, Discussion of the Technical Requirements and Licensing Basis for Reactor Fill Activity, was a narrative of the requirements for Unit 1 under the present conditions.

The inspectors noted that this EWR specified that piping was evaluated to meet dead weight condition only and no seismic analysis was supported by this configuration; System Engineering was required to establish an appropriate monitoring program for leak detection; and System Engineering was required to coordinate with Operations to establish an effective contingency plan for leakage and ensure availability of an appropriate draindown path and storage or waste processing of the boundary volume.

EWR 05MEB068015

This EWR, from the Mechanical Engineering Branch, provided clarification of the requirements for the installation of integral piping attachments associated with the supports identified in Part 2, Attachment C, of EWR 04MEB068199. A total of 22 pipe supports were listed with integral piping attachments in systems such as RHR, Core Spray, Reactor Recirculation, RWCU, and Main Feedwater.

EWR 05MEB068031

This EWR, from the Mechanical Engineering Branch, provided for a scope change in the vessel fill boundary to include the RPV flange low pressure seal leakoff piping and vessel nozzle N14. The EWR listed three supports that were required for RPV fill. The EWR referenced DCN 51255, which installed and modified the supports on the one-inch low pressure seal leakoff piping from the RPV head.

EWR 05MEB068010

This EWR, from the Mechanical Engineering Branch, provided engineering analysis and an evaluation for pulling the fuel pool to reactor cavity gates following RPV fill and cavity flood up. The EWR, Attachment A, contained the part of the response which represented the scope associated with Instrumentation and Control, Mechanical, and Electrical designs to support the removal of the fuel pool gates. This attachment listed applicable DCNs, and the individual status, required for gate removal. Some of the DCNs listed were required to be in the Return-to-Operation (RTO) status while others were listed as requiring only specific stages to be in the RTO status. Other DCNs were listed as requiring only partial closures. Among the DCNs listed and their required status were the following: DCNs 51046, RWCU System - Drywell, and 51194, RWCU System - Reactor Building, were to be in the RTO status, with all work complete and signed off; DCN 51351, Fuel Pool Supports - Reactor Building, was also to be in the RTO status; DCN 51107, Unit 1 Annunciator Upgrade, Stages 14, 15, and 17, were to be in the RTO status; DCN 51642, which was issued to replace fuses in various electrical systems in the reactor building, Stages 1, 3, 5, 23, and 24, were to be in the RTO status; DCN 51196, Reactor Building - Mechanical, RCIC System 71, modify valve 1-FCV-71-39, RCIC pump discharge, and replace valve 1-FCV-71-40, RCIC pump discharge, with partial closures through WO 03-001991-06 and WO 03-001991-02 respectively; and DCN 51231, remove wiring and cables from panel 25-5-1 and install new panel 25-426, using partial closures. In addition, this ERW required that System 69, RWCU, shall be SPOC I complete; System 78, Fuel Pool Cooling, shall be

SPOC II complete; Core Spray, Loop II, will be available for level control; and weep holes and vents in electrical boxes shall be installed to support work completion.

EWR 05CEB068053

This EWR, from the Civil Engineering Branch, provided a list of the supports required by engineering to be installed on applicable systems prior to pulling Unit 1 fuel pool to reactor cavity gates. The supports were for the piping within the boundary identified in EWR 05MEB068010. Among the systems listed were Reactor Recirculation, System 68; RWCU, System 69; Reactor Core Isolation Cooling, System 71; High Pressure Coolant Injection, System 73; Residual Heat Removal, System 74; and Core Spray, System 75. The EWR stated that the requirements of EWR 04MEB068199 and EWR 05MEB068015 must be met as well as the additional requirements in this EWR. The EWR also stated the following: Snubbers would not be installed and that the snubbers shall be temporarily replaced by rigid fabricated struts; all spring hangers must be unpinned/unblocked and set within the cold setting range; no fuel movement is permitted in Unit 1 spent fuel pool after gates are removed; and Reactor Water Clean Up, System 69, shall be SPOC I, complete with DCN 51194, Reactor Building - Mechanical, Reactor Water Cleanup, System 69 and DCN 51046 RWCU, System 69, Electrical and Mechanical, in the RTO status. The EWR further stated: There is no licensing basis for seismic requirements for the current Unit 1 condition; however, as stated in the UFSAR, Section 10.5.5, the fuel storage pool must maintain the ability to (1) maintain irradiated fuel submerged in water, (2) re-establish normal fuel pool level, or (3) safely remove fuel from the plant; and engineering must ensure that should a Design Basis Earthquake (DBE) occur in this current Unit 1 condition, the irradiated fuel in the pool will remain covered and be adequately cooled.

EWR 05CEB068066

This EWR, from the Civil Engineering Branch, was issued to prioritize the installation of the pipe supports required by engineering to be installed prior to the Unit 1 fuel pool gate pull, as identified in EWR 05CEB068053. The EWR listed five temporary rigid supports to replace five snubbers and listed three supports to be installed prior to the gate pull. The EWR revised Item 5 in EWR 05CEB068053 as follows: all spring hangers must be unpinned/unblocked as noted on Attachment “B;” after this process, all spring hangers shall be confirmed to have a minimum of one-inch travel; and all temporary supports for spring hangers must be removed prior to gate pull.

c. Conclusions

Based on review of six Engineering Work Requests the inspectors concluded that the Unit 1 engineering organization had provided adequate guidance and technical information to support activities associated with reactor vessel fill and removal of the reactor cavity gates.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) Review of Generic Letter 82-33, Instrumentation to Follow the Course of an Accident - Regulatory Guide (RG) 1.97

The inspectors reviewed GL 82-33, Instrumentation to Follow the Course of an Accident - RG 1.97. The purpose of the inspection was to verify that Browns Ferry Unit 1 has an instrumentation system for assessing plant conditions during and following the course of an accident that meets the criteria specified in RG 1.97. The inspection was carried out using the methods specified in Temporary Instruction 2515/87, Inspection of Licensee's Implementation of Multi-plant Action A-17: Instrumentation for Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (RG 1.97). The Licensee committed to complete the requirements of GL 82-33 in a letter dated April 30, 1984. Previous NRC reviews of GL 82-33 are documented in Inspection Report Nos. 259,260,296/90-032, /93-201, and /95-039. The inspectors reviewed the following RG 1.97 Category I instrumentation: reactor water level, reactor coolant system pressure, drywell pressure, primary containment isolation valve positions, reactor coolant system circulating radiation levels, drywell hydrogen and oxygen concentrations, and primary containment radiation level. Category I instrumentation was inspected for equipment qualification, redundancy, power sources, quality assurance program, display and recording, range, equipment identification, installed interfaces, and service, testing, and calibration. Inspectors reviewed the following RG 1.97 Category 2 instrumentation: suppression pool water temperature, drywell atmosphere temperature, safety relief valve positions, high pressure coolant injection flow, core spray flow, residual heat removal system flow, and residual heat removal system heat exchanger outlet temperature. Category 2 instrumentation was inspected for equipment qualification, quality assurance program, display and recording, range, equipment identification, installed interfaces, and service, testing, and calibration. Inspectors conducted the review of Categories 1 and 2 instrumentation through a combination of documentation review; and walkdowns of the control boards, instrument rooms, and the reactor building and concluded that planned modifications on Unit 1 instrumentation for assessing plant conditions during and following the course of an accident were adequate to meet the criteria specified in RG 1.97. The inspectors noted that additional review in this area will occur associated with a license amendment involving a proposed upgrade to the neutron monitoring system. The inspectors concluded that planned modifications on Unit 1 instrumentation for assessing plant conditions during and following the course of an accident were adequate to meet the criteria specified in RG 1.97. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.2 (Closed) TMI Action Item II.F.2.4, Instrumentation for Detection of Inadequate Core Cooling (Generic Letter 84-23)

The inspectors reviewed TMI Action Item II.F.2.4, Instrumentation for Detection of Inadequate Core Cooling, to determine the status of the licensee's efforts in this area. TVA letters dated April 8, 1985, July 15, 1985, October 15, 1985, and March 12, 1986, provided the licensee's response to GL 84-23 and stated that vessel water level instrumentation would be modified to improve accuracy and reliability under transient and accident conditions and decrease the need for operator diagnosis. Review and approval of the licensee's proposed modifications were documented in a letter dated November 18, 1986. In addition, closure of this item prior to restart of Unit 2 was documented in NRC Inspection Report 50-259,260,296/89-35 based on review of the implemented modifications. For Unit 3 the modifications were tracked as a restart commitment. The inspectors reviewed DCNs 51066 and 51163 and associated drawings 1-47E600-2519 and 1-47E600-2520, which provided the details associated with planned upgrades to reactor vessel water level instrumentation. This describes the licensee's modifications to reduce vertical drops in the instrumentation reference legs in the drywell on Unit 1. These DCNs are discussed in more detail in Section E1.1 of this report. The inspectors determined that the licensee's planned upgrades were intended to bring Unit 1 instrumentation up-to-date, remain comparable to Units 2 and 3, and comply with NUREG 0737 requirements. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.3 (Closed) TMI Action Item II.K.3.13, HPCI/RCIC Initiation Levels

The inspectors reviewed TMI Action Item II.K.3.13, HPCI/RCIC Initiation Levels, to determine the status of the licensee's efforts in this area. This action item involves two separate issues. The first portion of this item deals with the separation of HPCI and RCIC system reactor water level initiation setpoints. In relation to this matter, the licensee endorsed the BWR Owners Group evaluation which stated that the separation of HPCI and RCIC initiation levels would be of negligible safety benefit. The staff endorsed this position in a Safety Evaluation dated March 16, 1983.

The second portion of this item deals with the re-start of RCIC on a low reactor level condition following an automatic shut-off on high water level. In order for the RCIC system to be able to automatically re-start in the above described manner, a modification to the RCIC control logic is planned in accordance with DCN 51220. More specifically, this design change will alter the valve which closes on a high level condition. Prior to the modification, the turbine trip valve, 1-FCV-71-9, would close on a high water level. Following the modification, the steam supply valve, 1-FCV-71-8 will receive the close signal on a high level condition. Other RCIC trip signals will continue to close the turbine trip valve and require manual action to reset the system for injection. The inspectors reviewed DCN 51220, which provided the details associated with planned upgrades to the RCIC System on Unit 1. This DCN is discussed further in Section E1.1 of this report. The inspectors determined that the licensee's planned upgrades were intended to bring Unit 1 RCIC controls up-to-date, remain comparable to Units 2 and 3, and comply with NUREG 0737 requirements. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.4 (Closed) TMI Action Item II.K.3.18, ADS Actuation Modifications

The inspectors reviewed TMI Action Item II.K.3.18, ADS Actuation Modifications, to determine the status of the licensee's efforts in this area. The licensee reported their intentions for compliance in a letter dated March 5, 1987. In that letter TVA committed to implement the second of two acceptable options discussed in a letter from NRR dated June 3, 1983. The option selected by TVA was to modify the ADS logic to allow for the bypassing of the high drywell pressure permissive after a sustained low water level and the addition of a manual inhibit switch. NRC review of this issue on Unit 2 was documented in Inspection Report 259,260,296/91-10. NRC review of this issue on Unit 3 was documented in Inspection Report 259,260,296/95-60. The inspectors reviewed DCN 51081, which provided the details associated with planned upgrades to the ADS logic on Unit 1. This DCN is discussed further in Section E1.1 of this report. The inspectors determined that the licensee's planned upgrades were intended to bring Unit 1 instrumentation up-to-date, remain comparable to Units 2 and 3, and comply with NUREG 0737 requirements. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.5 (Closed) TMI Action Item II.K.3.28, Qualification of ADS Accumulators

The inspectors reviewed TMI Action Item II.K.3.28, Qualification of ADS Accumulators, to determine the status of the licensee's efforts in this area. The licensee reported their intentions for compliance in letters dated July 12, 1984, and July 11, 1985. The licensee's planned modifications for the ADS accumulators were reviewed by NRR and approved by the NRC by letter dated July 24, 1985. NRC review of this issue on Unit 2 was documented in Inspection Report 259,260,296/91-10. NRC review of this issue on Unit 3 was documented in Inspection Report 259,260,296/95-56. The inspectors reviewed DCN 51205, which provided the details associated with planned upgrades for the nitrogen supply in the drywell on Unit 1. This DCN is discussed further in Inspection Report 259/04-06. The inspectors determined that the licensee's planned upgrades were intended to bring Unit 1 ADS accumulator design up-to-date, remain comparable to Units 2 and 3, and comply with NUREG 0737 requirements. The inspectors determined that no further actions were required for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to the Unit 2 and 3 solutions with the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

III. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Program

b. Inspection Scope

The inspectors continued to observe and/or review the licensee's activities involved with the maintenance program. Maintenance work activities were controlled by approved plant procedures and WOs. The inspectors also reviewed selected corrective action documents initiated during the maintenance activities to document problems.

c. Observations and Findings

The Unit 1 recovery maintenance organization routinely performs activities which do not involve design changes. Major recovery activities which involve design changes are normally performed by the Unit 1 modifications organization. Major recovery project activities performed by Maintenance during the reporting period were limited to installation of the three replacement condensate pumps and motors. These pumps and motors are non-safety related and were replaced for the extended power uprate. Activities observed and reviewed by the inspectors included the following:

- WO 04-723643-00, Perform Local Leak Rate Testing (LLRT) of selected valves in the Reactor Feedwater System using procedure TI-106, General Leak Rate Test Procedure, Revision 10
- WO 03-020012-47, Demolish selected fire protection piping and hangers in the quad rooms, elevation 593, to prepare for DCN 51180
- WO 02-012943-00, Repair and test the north fuel preparation machine
- WO 04-712785-00, Perform loop calibration of main control room pressure indicator PI-33-003A/1, Service Air System

The inspectors also reviewed PERs issued by the licensee documenting conditions adverse to quality observed during the maintenance process. Among the PERs reviewed were the following:

- PER 85973, documented that a washer and screw, supplied by GE, were missing from the north fuel preparation machine. Substitutes were installed and will be replaced when the correct parts arrive. This did not prevent the preparation machine from being tested.

- PER 85735, documented that a type EC trip device in breaker circuit BKR 624, a type AK breaker, was identified as leaking oil.

- PER 85565, documented that a type EC trip device in breaker circuit BKR 604, a type AK breaker, failed the short time trip criteria of between 0.12 to 0.34 second when it tripped at 0.68 seconds.

- PER 85399, was issued to document the extent of condition of the problems identified with the type EC trip devices located in the type AK breakers, such as leaking oil and not meeting trip time criteria.

c. Conclusions

The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which do not require design changes to support the Unit 1 Restart.

I. Operations

O1.1 Conduct of Operations

O8.1 Miscellaneous Operations Issues

O8.1.1 Filling of the Unit 1 Reactor Pressure Vessel/Cavity and Vessel Disassembly

a. Inspection Scope

The inspectors reviewed and observed the Unit 1 recovery activities associated with the filling of the Unit 1 RPV and reactor cavity with demineralized water, RPV disassembly, and preparations for the removal of the spent fuel pool gates. The RPV fill was considered to be a major milestone for the Unit 1 recovery effort. The overall filling and disassembly activities were performed in accordance with approved plant procedures, such as periodic operating instructions (POIs), technical instructions (TIs), and general operating instructions (GOIs).

b. Observations and Findings

The inspectors observed and reviewed licensee activities associated with filling the RVP and reactor cavity and disassembly of the reactor vessel. The Unit 1 RPV was filled using hoses from the Demineralized Water System (DWS) to various one-inch piping connections located on the Reactor Recirculating System. The RPV disassembly included removal of the vessel head, the steam dryer, and the steam separator. Specific activities observed included the following:

Reactor Vessel Fill Readiness

The inspectors reviewed licensee readiness for the RPV fill. Preparations for filling the RPV were controlled by temporary instruction 1-TI-498, Unit 1 Reactor Vessel Fill Readiness. This instruction provided precautions, prerequisites, and managerial concurrence that the Unit 1 RPV and Reactor Cavity were ready for safe and efficient filling. The inspectors noted that this instruction also provided a systematic method to ensure that all open items required to support the fill were completed or appropriately addressed prior to the fill. The instruction was formatted in a manner that required signatures by department managers, supervisors, and various other lead personnel ensuring that a particular work activity was completed. Work activities observed or reviewed included ensuring that temporary level indication was installed; ensuring that temporary cameras, with a monitor in the control room, were installed at each temporary level instrument and on the refueling floor; and verifying that motor operated valves within the vessel fill boundary were manually closed by the valve operator.

RPV and Cavity Fill

Actual RPV and reactor cavity filling was controlled by temporary instruction 1-POI-75-1, Vessel/Cavity Fill. This instruction provided for filling the RPV and reactor cavity using temporary hoses from the DWS as the preferred method; filling the RPV and cavity using the condensate transfer system into the Core Spray System Loop II as an alternate method; and a method for reducing the vessel/cavity level as needed. The instruction was written assuming that the reactor vessel is defueled, primary containment integrity is not required, and electrical power is not available to any flow path motor operated valves during the initial vessel fill.

1-POI-75-1 also provided directions on lowering and raising the RPV and cavity level. The procedure stated that when the RWCU System was functional, the RWCU would be used to control vessel/cavity level. The instruction further stated that with the RWCU available the temporary hoses would be removed and this procedure would be closed out. However, EWR 05MEB068092, was issued by the Mechanical Engineering Branch and superceded EWR 05MEB068010. This EWR provided for a scope change in that EWR 05MEB068010, which provided the original engineering analysis for RPV/cavity fill, had required that the portions of RWCU be in a RTO status. The licensee decided that the RWCU would not be required for the pulling of the gates between the reactor cavity and SFP because adequate water quality and clarity was maintained by the temporary demineralizer. The inspectors reviewed EWR 05MEB06092 and determined that the licensee's decision was reasonable. EWR 05MEB068010 is further discussed in Section E1.6.

The inspectors observed filling of the RPV to 603 inches above RPV zero reference (inner bottom of the RPV) on April 7, 2005. The level was subsequently raised to the vessel flange on April 29, 2005, following the installation of the four main steam line nozzle plugs on April 28, 2005. The inspectors noted that leak checks of the main steam line plugs were performed prior to raising the level above the vessel flange.

Reactor Cavity Level Control

Requirements for monitoring the cavity with the fuel gates installed were provided in temporary instruction 1-POI-78-1, Monitoring Reactor Cavity Level With Fuel Gates Installed. This instruction required that a temporary level scale, in graduated increments from minus 8 inches to plus 8 inches, be mounted on the vessel cavity wall; a temporary camera be installed on the refueling floor to enable remote monitoring of the temporary level scale in the control room; and the cavity level be maintained between plus 4 inches and minus 4 inches on the temporary level scale. The instruction also required that the cavity level be raised and lowered, when necessary, in accordance with 1-POI-75-1, Vessel/Cavity Fill. The instruction stated that upon removal of the fuel pool gates, the SFP storage and the RPV cavity level would be controlled in accordance with applicable permanent plant instructions. The inspectors observed the raising of the water level above the vessel flange to match the level of the fuel pool on May 1, 2005, in preparation for the removal of the fuel pool gates. In addition, the inspectors observed installation of a temporary demineralizer on the refueling floor to facilitate the removal of dissolved cobalt in the RPV water.

RPV Disassembly

In addition, a new permanent Unit 1 instruction 1-GOI-100-3A, Refueling Operations (RX Vessel Disassembly and Floodup), was developed based on existing Unit 2 instruction 2-GOI-100-3A, Revision 29. The instruction provided the precautions, limitations, prerequisites, and procedural steps for safe and efficient vessel disassembly, reactor cavity level control, and the coordination of both mechanical and electrical surveillance instructions and requirements. The instruction also provided requirements for the removal of the gates between the Unit 1 reactor cavity and SFP. The inspectors observed or reviewed licensee activities associated with removal of the drywell head, RPV head, steam dryer and separator. The RPV head and steam dryer were removed on April 24, 2005, in preparation for a special test of the dryer. This special test, referred to as a “rock” test, was performed by the vendor, and was completed on April 27, 2005. This testing was performed after the dryer was instrumented and placed back into the reactor vessel and was intended to verify that the steam dryer had an adequate degree of fit inside the RPV. The test was performed and the results were acceptable. Following the test, the steam dryer was returned to the equipment pit where the instrumentation was removed. The steam separator was then removed and placed in the equipment pit, after the level was raised to match the level of the fuel pool, on May 2, 2005. These activities were also performed in order to make preparations for the In-Vessel Visual Inspection (IVVI).

c. Conclusions

Licensee activities associated with the filling of the RPV and reactor cavity were conducted in a safe and controlled manner.

O8.1.2 Removal of Gates Between Reactor Cavity and Fuel Pool

a. Inspection Scope

The inspectors reviewed and observed the Unit 1 recovery activities for the removal of the gates between the Unit 1 reactor cavity and the spent fuel storage pool. The removal of the fuel pool gates was considered to be a major milestone for the Unit 1 recovery effort. The overall removal activities were performed in accordance with approved plant procedures, such as TIs, EWRs and GOI instructions.

b. Observations and Findings

The inspectors observed and reviewed licensee activities associated with removal of the gates between the reactor cavity and spent storage pool. Readiness for removal of reactor cavity gates was controlled by TI 1-TI-507, Unit 1 Reactor Pressure Vessel Floodup/Gates Pulled. This instruction provided precautions, prerequisites, and managerial concurrence that the gates were ready for safe and efficient removal. The instruction also provided a systematic method to ensure that all open items required to support the gate removal were completed or appropriately dispositioned prior to the removal. The instruction was formatted in a manner that required signatures by department managers, supervisors, and various other lead personnel ensuring that a particular work activity was completed. Work activities observed or reviewed included: ensuring that system engineering had reviewed all items coded to gates pulled (coded as MB) for closure or were otherwise appropriately addressed; ensuring that design engineering has reviewed all items coded to MB for closure or were otherwise appropriately addressed; ensuring that operations had reviewed all applicable systems with items coded to MB for closure or were otherwise appropriately addressed; ensuring that the requirements for pulling gates, as identified in the applicable EWRs, were met or appropriately addressed; and verifying that motor-operated valves within the gate removal boundary were manually closed by the valve operator. The inspectors noted that licensee activities associated with removal of the gates were also controlled by permanent instruction 1-GOI-100-3A. The gates were removed using the Reactor Building overhead crane on July 8, 2005. The water levels in the reactor cavity and spent fuel storage pool were verified as being the same prior to the removal.

The inspectors concluded that the reactor cavity gates were removed in a safe and controlled manner and that the gates were placed in the proper storage racks within the spent fuel storage pool.

c. Conclusions

The reactor cavity gates were removed in a safe and controlled manner and the gates were placed in the proper storage racks within the spent fuel storage pool. The ongoing gate removal activities did not affect maintaining normal fuel pool level nor impact the ability to adequately cool irradiated fuel in the Unit 1 fuel storage pool.

V. Management Meetings

X1 Exit Meeting Summary

On August 1, 2005, the resident inspectors presented the inspection results to Mr. Jon Rupert and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Baron, Nuclear Assurance Manager, Unit 1
M. Bennett, QC Manager, Unit 1
D. Burrell, Electrical Engineer, Unit 1
P. Byron, Licensing Engineer
J. Corey, Radiological and Chemistry Control Manager, Unit 1
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
R. Cutsinger, Civil/Structural Engineering Manager, Unit 1
B. Hargrove, Radcon Manager, Unit 1
E. Hollins, Maintenance and Modifications Manager, Unit 1
R. Jackson, Bechtel
B. Ditzler, TVA Welding Engineering Supervisor, Unit 1
R. Jones, Plant Recovery Manager, Unit 1
S. Kane, Licensing Engineer
D. Kehoe, Nuclear Assurance, Unit 1
J. Lewis, ISI Program Engineer, Unit 1
G. Lupardus, Civil Design Engineer, Unit 1
J. McCarthy, Licensing Supervisor, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Rupert, Vice President, Unit 1 Restart
J. Schlessel, Maintenance Manager, Unit 1
J. Symonds, Modifications Manager, Unit 1
E. Thomas, Bechtel
D. Tinley, NDE Level III & Unit 1 ISI Project Manager
J. Valente, Engineering Manager, Unit 1

INSPECTION PROCEDURES USED

IP 37550	Onsite Engineering
IP 37551	Engineering
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 92701	Follow-up

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

82-33	GL	Instrumentation to Follow the Course of an Accident - Regulatory Guide 1.97. (Section E8.1)
II.F.2.4	TMI	Instrumentation for Detection of Inadequate Core Cooling. (Section E8.2)
II.K.3.13	TMI	HPCI/RCIC Initiation Levels. (Section E8.3)
II.K.3.18	TMI	ADS Actuation Modifications. (Section E8.4)
II.K.3.28	TMI	Qualification of ADS Accumulators. (Section E8.5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section E1.1: Plant Modifications

Procedures and Standards

SPP-9.3, Plant Modifications and Engineering Change Control, Revision 9
MAI-4.2B, Piping, Revision 20
G-94, Piping Installation, Modification, and Maintenance, Revision 2

DCNs

DCN 51017, Unit 3 ECCS Accident Signal Logic
DCN 51066, Reactor Feedwater
DCN 51069, Main Steam
DCN 51081, Modifications to Unit 1 Auxiliary Instrument Room
DCN 51085, 120-Volt AC
DCN 51090, 480-V Distribution
DCN 51100, Modifications to Control Room Panel 1-9-22
DCN 51107, Annunciator Upgrade
DCN 51163, Reactor Feedwater
DCN 51200, Core Spray
DCN 51216, 480-V Distribution
DCN 51220, RCIC System
DCN 51231, Reactor Feedwater

Section E1.2: Temporary Modifications

Procedures, Guidance Documents, and Manuals

0-TI-405, Plant Modifications and Design Change Control, Revision 0
0-TI-410, Design Change Control, Revision 1
SPP-9.5, Temporary Alterations, Revision 6

Other Documents

TACF 1-84-029-069, 1B RWCU Pump Motor
TACF 1-85-032-099, Unit 1 Reactor Protective System
TACF 1-88-002-69, RWCU Non-Regenerative Heat Exchanger
TACF 1-2002-002-069, RWCU System Piping
TACF 1-2005-002-64D, Primary/Secondary Containment Isolation Control Logic Circuitry

Section E1.3: System Return to Service Activities

Procedures, Guidance Documents, and Manuals

Technical Instruction 1-TI-437, System Return to Service (SRTS) Turnover Process for Unit 1 Restart, Revision 0
0-TI-404, Unit One Separation and Recovery, Revision 4
1-TI-474, Cleanliness Verification Program, Revision 0
0-TI-373, Plant Lay-up and Equipment Preservation, Revision 4
MSI-1-000-PRO001, Cleanliness of Unit 1 Fluid Systems, Revision 1

Section E1.4: Restart Test Program

Procedures and Standards

Technical Instruction 1-TI-469, Baseline Test Requirements, Revision 1
Operating Instruction, 1-OI-69, Reactor Water Cleanup System, Revision 27
Surveillance Instruction 1-SI-3.3.3, ASME Section XI System Pressure Test of Fuel Pool Cooling System, Revision 0
Post Modification Test Instruction (PMTI) 1-PMTI-BF-51090-S57-64+S79, Functional Testing of 480-VAC Reactor MOV Bards and 480-VAC Shutdown Boards - Control Bay, System 57-4, Revision 0

PMTIs

1-PMTI-BF- 51090-S57-64+S79 (Stages 57 Thru 64 and Stage 79), Revision 0
1-PMTI-51100-STG06, Stage 6 and Stage 7, Revision 0
1-PMTI-51203-STG02, Revision 0
1-PMTI-023-052, Stage 25, Revision 0
1-PMTI-002-011, Stage 5, Revision 0

Problem Evaluation Reports (PERs)

79750, test deficiencies during performance of 1-PMTI-BF-51090-S57-64+S79
83861, label deficiencies identified during leak rate test on 1-SHV-75-54A and 54B
84205, failure mode identified in fuel pool cooling logic controls

Other Documents

BTRD 1-BFN-BTRD-079.002, System 79, Fuel Handling and Storage System, Revision 2

Section E1.5: Special Program Activities - Cable Installation and Cable Separation

Procedures and Standards

MAI 1.3, General Requirements for Modifications, Revision 21
MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts, Revision 41
MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts, Revision 45
MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control, Revision 6

Work Orders

03-001001-050, replace normal feeder cable for 480-V RMOV Board 1A
03-001001-052, replace normal feeder cable for 480-V RMOV Board 1B
03-001001-070, replace load shed cables for Load Shed Panel 0-PNL-25-44A-11
03-001001-071, replace load shed cables for Load Shed Panel 0-PNL-25- 44B-11

Section E1.6: Review of Engineering Work Requests to Support RPV Fill and Removal of Reactor Cavity Gates

EWRs

EWR 05CEB068053
EWR 05CEB068066
EWR 04MEB068199
EWR 05MEB068010
EWR 05MEB068015
EWR 05MEB068031

Section M1: Conduct of Maintenance

Procedures and Standards

SPP-10.2, Clearance Program, Revision 6
TI-106, General Leak Rate Test Procedure, Revision 10

Work Orders

04-723643-00, Perform LLRT of selected valves in the Reactor Feedwater System using Procedure TI-106, General Leak Rate Test Procedure
03-020012-47, Demolish selected fire protection piping and hangers in the quad rooms, elevation 593, to prepare for DCN 51180
02-012943-00, Repair and test the north fuel preparation machine
04-712785-00, Perform loop calibration of main control room pressure indicator PI-33-003A/1, Service Air System

Problem Evaluation Reports (PERs)

85399, extent of condition of the problems identified with the type EC trip devices
85565, type EC trip device in breaker circuit BKR 604 failed the short time trip criteria
85735, oil leak in type EC trip device in breaker circuit BKR 624, a type AK breaker
85973, washer and screw were missing from the north fuel preparation machine

Section O8.1.1: Filling of the Unit 1 Reactor Pressure Vessel/Cavity and Vessel Disassembly

1-TI-498, Unit 1 Reactor Vessel Fill Readiness, Revision 1
1-POI-75-1, Vessel/Cavity Fill, Revision 1
1-GOI-100-3A, Refueling Operations (RX Vessel Disassembly and Floodup), Revision 5

Section O8.1.2: Removal of Gates Between Reactor Cavity and Fuel Pool

1-TI-507, Unit 1 Reactor Pressure Vessel Floodup/Gates Pulled, Revision 1
1-GOI-100-3A, Refueling Operations (RX Vessel Disassembly and Floodup), Revision 5