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

May 5, 2003

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

SUBJECT: WATTS BAR NRC INTEGRATED INSPECTION REPORT 50-390/03-02 AND 50-391/03-02

Dear Mr. Scalice:

On April 5, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Watts Bar Nuclear Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings which were discussed on April 7, 2003, with Mr. W. Lagergren and other members of your staff.

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

The report documents one Unit 2 NRC-identified finding that was determined to be a severity level IV violation of NRC requirements consistent with Section IV.B of the NRC Enforcement Policy. The report also documents one NRC-identified finding and one self-revealing finding for Unit 1. Both Unit 1 findings were determined to be of very low safety significance (Green), one of which was determined not to be a violation of NRC requirements. Because of the very low safety significance and because it is entered in your corrective action program, the NRC is treating the other Unit 1 finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, one licensee-identified Green NCV is listed in Section 4OA7 of this report. If you contest any non-cited violation in the enclosed report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington D.C. 20555-0001; and the NRC Resident Inspector at the Watts Bar facility.

In accordance with 10 CFR 2.790 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 Website 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 Nos. 50-390, 50-391
License No. NPF-90 and Construction
Permit No. CPPR-92

Enclosure: NRC Inspection Report 50-390/03-02, 50-391/03-02
w/Attachment: Supplemental Information

cc w/encl:(see page 3)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-390, 50-391
License Nos:	NPF-90 and Construction Permit CPPR-92
Report No:	50-390/03-02, 50-391/03-02
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	1260 Nuclear Plant Road Spring City TN 37381
Dates:	December 22, 2002, through April 5, 2003
Inspectors:	T. Morrissey, Acting Senior Resident Inspector J. Reece, Resident Inspector R. Carrion, Project Engineer, Region II (Sections 1R05, 1R06, and 1R22) L. Miller, Senior Operations Inspector, Region II (Section 1R11.2) W. Bearden, Senior Resident Inspector, Browns Ferry Unit 1 Recovery (Sections 1R02 and 1R17) R. Chou, Reactor Inspector, Region II (Sections 1R02 and 1R17) R. Maxey, Reactor Inspector, Region II (Sections 1R02 and 1R17)
Approved by:	Stephen J. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000390/2003-002, 05000391/2003-002; Tennessee Valley Authority; 12/22/2002-04/05/03, Watts Bar, Units 1 & 2. Evaluations of Changes, Tests or Experiments, Surveillance Testing, Other Activities, and Event Followup

The report covered approximately a three-month period of inspection by resident inspectors and an announced inspection by a regional project engineer, a regional senior operations inspector, and regional reactor inspectors. The inspection identified one green non-cited violation (NCV), one severity level IV NCV, and one self-revealing green finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, Significance Determination Process (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, Reactor Oversight Process, Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A licensee welding procedure was determined to be inadequate because it defined the root pass of a weld as up to two layers, while applicable codes and standards defined a pass as a single layer. This allowed the licensee to perform Liquid Penetrant (PT) examinations on the second layer of welds instead of the first layer or root of the weld. This practice could mask defects existing in the root pass or root of the weld.

  An inspector-identified, non-cited violation of 10 CFR 50, Appendix B, Criteria IX, Control of Special Processes, was identified. This finding is greater than minor because it affected the objective of the Initiating Events cornerstone. Failure to perform a PT on the root pass of certain welds could allow weld defects in the root pass to remain undetected. Undetected defects could develop into cracks or other problems later and impact component or system safety. The issue was determined to be of very low safety significance based upon no actual failure of welds. (Section 1R02)

- Green. A self-revealing finding was identified for inadequate preventive maintenance (PM) work instructions for an electrical connection associated with the potential device on the C-phase main transformer. Problems with the PM caused a ground fault which resulted in a reactor trip.

  This self-revealing finding is greater than minor because it resulted in a perturbation in plant stability by causing a reactor trip. The finding was of very low safety significance because, although it caused a reactor trip, it did not increase the likelihood of a primary or secondary system loss of coolant accident (LOCA) initiator, did not contribute to a combination of a reactor trip and loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. The finding was not a violation of regulatory requirements because it occurred on nonsafety-related secondary plant equipment. (Section 4OA3)

Other Activities

- The inspectors identified that the applicant had initiated an unapproved reduction in equipment preservation to the Unit 2 lay-up process. The applicant had elected to cease performing preventive maintenance on many components.

  An inspector-identified, non-cited violation of 10 CFR 50, Appendix B, Criteria XIII, Identification and Control of Materials, Parts, and Components, was identified. This finding satisfied a traditional enforcement criterion of failure to receive NRC approval for a change in licensee activity In accordance with the NRC Enforcement Policy, the finding was characterized as a Severity Level IV NCV involving a failure to receive prior NRC approval for a change in licensee activity (Section 4OA5).

B. Licensee-Identified Violations

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 corrective action tracking number is listed in Section 4OA7.

Report Details

Summary of Plant Status

Unit 1 operated at or near 100 percent power for the entire inspection period except for one reactor trip. On March 10, Unit 1 automatically tripped due to a short on the capacitance tap connector on the C-phase main transformer. The cause of the trip was determined, and the unit was restarted on March 15. Unit 2 remained in a suspended construction status.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On January 22, 2003, the inspectors reviewed licensee actions taken for actual freezing weather conditions to limit the risk of freeze-related initiating events and to adequately protect mitigating systems from its effects. The inspectors walked down selected components associated with the refueling water storage tank level instruments and condensate/feedwater piping to evaluate implementation of plant freeze protection. In addition, the material condition of selected freeze-protected components' insulation was inspected for damage. Corrective actions to items identified in relevant problem evaluation reports (PERs), work orders (WOs), and a self-assessment of freeze protection practices and procedures were assessed for effectiveness and timeliness. On March 19, 2003, the inspectors reviewed licensee actions for a tornado watch to verify that the actions were in accordance with procedures. The inspectors toured the affected plant grounds inside the protected area to identify any loose debris which could become missiles during a tornado event. Specific documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of licensee evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility or procedures may be made and tests conducted without prior NRC approval. The inspectors reviewed evaluations for six changes and additional information, such as calculations, supporting analyses, the Updated Final Safety Analysis Report (UFSAR), and drawings to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The six evaluations reviewed are listed in the attachment.

The inspectors also reviewed samples of changes such as design changes and procedure changes for which the licensee had determined that evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The 13 "screened out" changes reviewed are listed in the attachment.

The inspectors also reviewed one recent Nuclear Assurance assessment and three self-assessments of the 10 CFR 50.59 process and reviewed two PERs associated with the 10 CFR 50.59 process to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

b. Findings

Introduction: A Green inspector-identified non-cited violation (NCV) was identified during review of a procedure change for failure to adequately define the term "root" or "root pass" in G-29A-S02, G-29 General Welding Procedure Specification. This procedure addressed measures required in 10 CFR 50, Appendix B, Criteria IX, Control of Special Processes, which, in part, requires that measures shall be established to assure that special processes, including welding, are controlled and accomplished by using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Description: On February 5, 2003, the inspectors identified that the licensee failed to correctly control or define "root" or "root pass" in G-29A-S02, G-29 General Welding Procedure Specification (GWPS) for American Society of Mechanical Engineers (ASME) and American National Standard Institute (ANSI), GWPS 1.M.1.2, Revision 2, Addendum 1, Revision 0, page 1 of 1. The G-29 General Welding Procedure Specification defines "root" or "root pass" to be a minimum of the first two layers of the weld. It also states that these definitions are not necessarily standard American Welding Society (AWS) A3.0 definitions. On May 28, 2002, the licensee revised this procedure to add another definition for using ASME Code Case N-416-1, Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items, on Class 3 components. The added definition states that for implementation of the additional surface examinations required by the NRC for use of Code Case N-416-1 on Class 3 components, the term "root pass" is to be considered up to the first two layers of the weld.

Footnote 6 of 10 CFR 50.55a states, in part, that ASME Code cases that have been determined suitable for use by the Commission staff are listed in, among others, NRC Regulatory Guide (RG) 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1. RG 1.147 allows the use of Code Case N-416-1 provided that additional surface examinations are performed on the root (pass) layer of butt and socket welds. ASME (1992 edition) Section IX in QW-102, Terms and Definitions refers to QW-492 and AWS A3.0-80, Terms and Definitions. QW-492 (1992 edition), Page 169, defines "pass - a single progression of a welding or surfacing operation along a joint, weld deposit, or substrate." The term "pass" on page 19 of AWS A3.0-80 refers to "weld pass." AWS A3.0-80, page 30, defines weld pass as "a single progression of a welding or surfacing operation along a joint, weld deposit, or substrate. The result of a pass is a weld bead, layer, or spray deposit." AWS Handbook "Welding Inspection" (1968 edition) which superseded Inspection Handbook B1.1-45, prepared by the AWS Committee on Methods of Inspection, page 154, states that the first layer, or root pass, is the most important one, from the point of view of final soundness. Both ASME and AWS define "weld pass" or "pass" as a single progression of a welding. AWS Welding Inspection Handbook clearly defines the first layer as the root pass. Therefore, the root pass is the first layer and a single layer of the weld.

The licensee's G-29 procedure definition of weld root and root pass contradict the definitions by ASME and AWS. The licensee defines a "root" or "root pass" as a minimum of two layers or up to two layers which do not conform with ASME and AWS Codes. This would allow the licensee to apply a Liquid Penetrant Test (PT) on the second layer of the weld when the PT is required by Code Case N-416-1 to be performed on the root pass or root of the weld. When a PT is performed on the second layer instead of the first layer, defects existing in the first layer might be masked and not detected. The licensee indicated that they have previously applied this procedure in the field.

<u>Analysis</u>: The inspectors determined that this finding affected the objective of the Initiating Events Cornerstone to limit the likelihood of those events that upset plant stability. Failure to perform adequate PT on the root weld of Class 3 components could result in undetected defects which could impact equipment performance or cause piping failures which could result in flood hazards. The issue was evaluated using the Significance Determination Process (SDP). No actual defects or failures of Class 3 components were identified as a result of the performance deficiency and the liklihood of an undetected first layer defect is very low. Therefore, the finding did not contribute to the likelihood of a LOCA initiator, nor the likelihood of both a reactor trip and mitigating system availability, nor the likelihood of a fire or flood, and, therefore, is Green.

<u>Enforcement</u>: 10 CFR 50, Appendix B, Criteria IX, Control of Special Processes, requires, in part, that measures shall be established to assure that special processes, including welding, are controlled and accomplished by using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Applicable codes and standards in the Watts Bar licensing basis include ASME Section IX, QW-102, QW-492, and AWS A3.0-80, which define pass as a single progression of a welding or surfacing operation along a joint, weld deposit, or substrate. Contrary to the above, the licensee's measures established in G-29, General Welding Procedure Specification, incorrectly defined the root of the weld or root pass as a minimum of two layers which resulted in a condition which weld defects could be masked when a PT is required to be performed on the root pass. Because the failure to adequately control the procedure specification in this case is of very low safety significance and no failed welds were actually found and because the licensee documented this issue in their corrective action program under PER 03-003075-000, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-390/2003-02-01, Failure to Adequately Define the Root of the Weld or Root Pass in G-29 General Welding Procedure Specification.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted three partial equipment alignment walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with the other train or system inoperable or out of service. The inspectors reviewed the functional system 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. Specific documents reviewed are listed in the attachment.

- “A” train safety injection (SI), containment spray (CS) and residual heat removal (RHR) spray systems during “B” train SI and 713 elevation room cooler outage
- 1A, 1B, 2B, diesel generator (DG) systems during 2A DG outage
- “B” train SI and centrifugal charging pump (CCP) systems during “A” train SI system outage

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of eight areas important to reactor safety, listed below, to verify the licensee’s implementation of fire protection requirements as described in the Fire Protection Program, Standard Programs and Processes (SPP)-10.0, Control of Fire Protection Impairments, SPP-10.10, Control of Transient Combustibles, SPP-10.11, Control of Ignition Sources (Hot Work). The inspectors evaluated, as appropriate, conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment, and features; and (3) the fire barriers used to prevent fire damage or fire propagation. Other documentation reviewed is listed in the attachment.

- Cable spreading room
- A train DG rooms
- Motor-driven auxiliary feedwater (AFW) pumps and Component Cooling System (CCS) area
- Turbine-driven auxiliary feedwater (TDAFW) pump room
- B train DG rooms
- B train RHR pump rooms
- Intake pump structure (IPS)
- B train SI pump room

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the UFSAR, Section 2.4, including related figures and drawings; Design Criteria WB-DC-40-29, Flood Protection Provisions, Revision 8; and interviewed cognizant licensee personnel knowledgeable about site flood protection measures and plant drainage plans to identify those design features important to flood protection and to identify those areas that can be affected by flooding, design flood levels, and protection features for areas containing safety-related equipment, such as level switches, and sumps.

The inspectors reviewed licensee instructions for cross-tying systems in the event of severe flooding and evaluated the availability of the identified spool piece to be used between the essential raw cooling water (ERCW) system and the CCS. The inspectors also reviewed the licensee's corrective action program for documents with respect to flood-related items identified in PERs written in 2002 to verify the adequacy of the corrective actions. The inspectors reviewed selected completed preventive maintenance procedures and WOs, listed in the attachment, for identified level switches, pumps, and 1E manholes for completeness and frequency.

The inspectors walked down selected areas, listed below, which contain risk-important equipment located below design flood levels to evaluate the adequacy of flood barriers, doors, floor drains, sump level switches, and sump pumps to protect the equipment, as well as their material condition. In addition, the inspectors observed maintenance personnel inspecting and performing preventive maintenance on the sump pump of Manhole #2.

- Lower level of the IPS to observe its external flooding protection features.
- Auxiliary building (elevations 676', 692', and 713'), the control building (elevation 708'), and the turbine building (elevation 685') to observe their internal flooding protection features.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Quarterly Inspection

a. Inspection Scope

On March 18, 2003, the inspectors observed operators in the plant's simulator during licensed operator requalification training associated with 3-OT-SRT0030B, Loss of ERCW/250VDC/Turbine Trip/SGTR, to verify that operator performance was adequate and that training was being conducted in accordance with Procedures TRN-1, Administering Training, and TRN-11.4, Continuing Training for Licensed Personnel. In addition, the inspectors verified that the training program included risk-significant operator actions, emergency plan implementation, and lessons learned from previous plant experiences.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection (Partial)

a. Inspection Scope

The inspectors reviewed and evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations. The inspectors also reviewed a sample of simulator performance test records (transient tests, malfunction tests, steady state test, and procedure tests), simulator modification request records, and the process for ensuring continued assurance of simulator fidelity to ensure compliance with 10 CFR 55.46, Simulation Facilities. Specific documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors sampled portions of structures, systems and components (SSCs), listed below, as a result of performance-based problems, to assess the effectiveness of maintenance efforts that apply to scoped SSCs and to verify that the licensee was following the requirements of Technical Instruction (TI)-119, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting, 10 CFR 50.65, and SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting 10 CFR 50.65. Reviews focused, as appropriate, on: (1) maintenance rule scoping in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) or (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). Specific documents reviewed are listed in the attachment.

- PER 03-001248-000, Failure of B train main control room (MCR) chiller on January 16, 2003
- WO 02-09260-000, Solid State Protection System (SSPS) board failure associated with performance of 1-SI-99-10-B on October 25, 2002

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors evaluated, as appropriate, for the selected six SSCs listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved. The inspectors verified that the licensee was complying with the requirements of 10 CFR 50.65 (a)(4); SPP-7.0, Work Control and Outage Management; SPP-7.1, Work Control Process; and TI-124, Equipment to Plant Risk Matrix.

- WO 03-002365-000, Repair rod control urgent failure annunciator
- WO 02-013401-000, DG 1A-A air compressor repair/replace
- WO 02-000813-000, 2A-A DG outage
- WO 03-004558-000, Troubleshoot and repair MCR Chiller B
- WO 03-005692-000, Repair 2A-A DG air valve 2-PCV-82-232A
- WO 03-005659-000, Repair 1A-A RHR pump vent valve 1-VTV-074-0010

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

For the nonroutine plant event described below, the inspectors reviewed operator logs, plant computer data, completed procedures, and interviewed plant personnel to determine what occurred and how the operators responded. In addition, the inspectors verified that the operator response was in accordance with plant procedures. Additional documents reviewed are listed in the attachment.

- On March 10, an automatic reactor trip from 100 percent power due to a fault associated with the C-phase main transformer

b. Findings

No findings of significance were identified. Section 4OA3 contains additional information on this event.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed five selected operability evaluations affecting risk-significant mitigating systems and barrier integrity, listed below, to assess as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) whether the compensatory measures, if involved, were in place, would work as intended, and were appropriately controlled; (5) where continued operability was considered unjustified, the impact on TS Limiting Conditions for Operation (LCOs) and the risk significance in accordance with the SDP. The inspectors verified that the operability evaluations were performed in accordance with SPP-3.1, Corrective Action Program, and SPP-10.6, Engineering Evaluations for Operability Determinations.

- PER 03-001248-000, Investigate MCR chiller trip on low suction pressure
- PER 03-003755-000, MCR chiller degraded condition
- PER 03-003677-000, 1B containment spray system maintenance resulted in loss of containment integrity
- PER 03-003466-000, Evaluate ERCW traveling screen corrosion and degradation
- PER 03-005311-000, Gas found is emergency core cooling systems (ECCS) hot leg injection piping.

b. Findings

No findings of significance were identified. Further information regarding PER 03-005311-000 is documented in Section 1R22. A licensee-identified violation associated with PER 03-003677-000 is documented in Section 4OA7.

1R17 Permanent Plant Modifications

a. Inspection Scope

As part of the biennial review, the inspectors evaluated design change notice (DCN) packages and engineering design change (EDC) packages for eight modifications, in the Initiating Events and Mitigating Systems cornerstone areas, to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The modifications and the associated attributes reviewed are as follows:

DCN D-50565-A, Change 6.9 kV Shutdown Board Degraded Voltage Relay Time Delay (Mitigating Systems)
- Control signals appropriate under accident conditions
- Heat removal

- Failure modes bounded by the existing analysis
- Response time sufficient to serve functional requirements

DCN 50895-A, Add Clean Out Ports in Piping Upstream of the Auxiliary Feedwater Isolation Valve for the ERCW System (Mitigating Systems)
- Material compatibility
- Design basis document changes
- Foreign material prevention
- Welding records
- Quality control inspection records
- Post modification testing

EDC 50951-A, Reduce Ice Weight Requirements (Mitigating Systems)
- Supporting license basis and safety evaluation documentation
- License design basis documents updated
- Supporting vendor thermohydraulic analysis
- Technical Specifications revision
- FSAR revision

DCN 50965, Clarify Rad Monitor Setpoint Scaling Document (Mitigating Systems)
- Functional test results
- Design basis document changes
- Updating of procedures reflect new setpoint

DCN 51072, Replace Analog Rod Position Indication System Electronics with Upgraded System (Initiating Events)

- Energy needs and Heat Removal
- Seismic qualification
- Operations procedures and training
- Failure modes bounded by the existing analysis

DCN 51125, Modify MDAFP Start Circuit to Improve Seal-in Function (Mitigating Systems)
- Functional requirements in accordance with design bases
- Heat removal
- Control signals appropriate under accident conditions
- Failure modes bounded by the existing analysis

EDC E51131A, Rev. A, Evaluation on Adding Alternate Purification Media to Mixed Bed Demineralizer to Complement Zinc Injection (Mitigating Systems)

- Materials compatibility
- Replacement of resin with same volume
- Enhancement of reduction of radiation levels
- Improvement of efficiency to remove corrosion products

EDC E51334-A, Rev. A, Document Revision for 10% Tube Plugging Margin and Associated UA Value for the Containment Spray Heat Exchanger (Mitigating Systems)
- Calculations revised
- System description revised
- Heat load increase
- Drawing changes
- Containment pressure integrity analysis

For selected modification packages, the inspectors observed the as-built configuration. Documents reviewed included procedures, engineering calculations, modifications design and implementation packages, WOs, site drawings, corrective action documents, applicable sections of the living UFSAR, supporting analyses, TS, and design basis information.

Five procurement engineering group (PEG) packages were also reviewed by the inspectors. These packages were reviewed to verify that proper engineering evaluation and/or dedication of any components/materials procured to commercial specifications had occurred. The five PEG packages reviewed are listed in the attachment.

The inspectors also reviewed the results of two recent self-assessments covering the modifications process to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed six post-maintenance test (PMT) procedures and/or test activities, as appropriate, for selected risk-significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function. The inspectors verified that these activities were performed in accordance with SPP-8.0, Testing Programs; SPP-6.3, Pre-/Post-Maintenance Testing; and SPP-7.1, Work Control Process. Additional documents reviewed are listed in the attachment.

- 1B-B SI pump component outage
- 2A-A DG component outage
- WO 03-001238-000, MCR B chiller did not trip on low suction pressure
- 1A-A SI pump component outage
- 1B CSP component outage
- WO 03-006161-000, Replace 1-ITS-82-A2 on 1A2 DG engine

b. Findings

No findings of significance were identified.

1R20 Unit 1 Forced Outage Due to Automatic Reactor Trip

a. Inspection Scope

On March 10, an automatic reactor trip from 100 percent power occurred due to a fault associated with the C-phase main transformer. The inspectors reviewed the forced outage plans to confirm that the licensee had appropriately considered risk in developing and implementing the plans. During the outage the inspectors observed or reviewed portions of the outage and preparations for restart. The inspectors verified that these activities were conducted in accordance with licensee procedures. Additional documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified. Section 4OA3 contains additional information on this event.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data for the six selected risk-significant SSCs, listed below, to assess, as appropriate, whether the SSCs met the requirements of the TS; the UFSAR; SPP-8.0, Testing Programs; SPP-8.2, Surveillance Test Program; and SPP-9.1, ASME Section XI. The inspectors also determined whether the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. Supporting documents reviewed are listed in the attachment.

- WO 02-013826-000, 0-FOR-70-2, Component cooling system pump 2B-B quarterly performance test
- WO 02-013971-000, 1-SI-63-901-B, Safety injection pump 1B-B quarterly performance test (Inservice Test)
- WO 02-015604-000, 1-SI-74-901-A, Residual heat removal 1A-A quarterly performance test

- WO 02-015828-000, 0-SI-82-12-B, Monthly diesel generator start and load test DG 2B-B
- WO 02-017633-000, 1-SI-63-901-A, Safety injection pump 1A-A quarterly performance test
- WO 03-005266-000, Partial performance of 1-SI-63-10-A, ECCS pumps and discharge pipes venting - Train A, as required during performance of ultrasonic examination (UT) of associated piping

b. Findings and Observations

The NRC has continued to evaluate the ECCS gas issue described in URI 50-390/02-04-02 and documented in NRC Integrated Inspection Report 50-390/02-04 and 50-391/02-04. The URI is associated with procedure 1-SI-63-10-A, ECCS pumps and discharge pipes venting - Train A, which is performed to satisfy TS Surveillance Requirement (SR) 3.5.2.3 that requires the licensee to "verify ECCS piping is full of water" every 31 days. The procedure also contains an administrative program that involves an assessment by the licensee to determine if gas may have accumulated in the area of various vent valves which have problems with accessibility. The assessment allows the licensee to waive the associated procedure appendixes and not perform the venting during normal plant operation. The inspectors had previously questioned the validity of the administrative program, as the licensee did not have as-found data from venting or UT of the associated piping performed early in a refueling outage or during a forced outage. During a forced outage on March 13, 2003, the licensee performed UT of the associated ECCS piping with the following results:

- Small gas pocket (3 inch by 4 inch) found at 1-VTV-63-639, HOT LEG 1 RHR INJ LINE VENT
- Gas pocket (4 inch pipe less than half full, 2.2 cubic feet of gas) found at 1-VTV-63-689 & 657, HOT LEG 1 & 3 SIP 1A-A INJ LINE VENT
- Gas pocket (4 inch pipe half full, 1.2 cubic feet of gas) found at 1-VTV-63-649, HOT LEG 2 & 4 SAFETY INJ LINE VENT
- No gas was found at vent valves associated with RHR hot leg 3 injection or boron injection to cold legs 2 & 3

The licensee subsequently vented the gas using the applicable appendixes of 1-SI-63-10-A and initiated PER 03-005311-000 to evaluate the above conditions. A licensee engineering evaluation determined that the gas found in the piping would not have impacted system operability. This event will be incorporated into URI 50-390/02-04-02 and will be evaluated by the NRC.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary plant modification against the requirements of SPP-9.5, Temporary Alterations, and SPP-9.4, 10 CFR 50.59 Evaluation of Changes, Test, and Experiments, and verified that the modification did not affect system operability or availability as described by the TS and UFSAR. In addition, the inspectors verified that the installation of the temporary modification was in accordance with the work package, that adequate configuration control was in place, that procedures and drawings were updated, and that post-installation tests verified operability of the affected systems.

- TACF 1-03-01-62, Install ultrasonic flow meter and video equipment to support repairs on 1-LPF-62-139/140 (WO 03-000052-000)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On February 20, 2003, the inspectors observed a licensee-evaluated emergency preparedness drill to verify that the emergency response organization was properly classifying the event in accordance with Emergency Plan Implementing Procedure (EPIP)-1, Emergency Plan Classification Flowchart, and making accurate and timely notifications and protective action recommendations in accordance with EPIP-2, Notification of Unusual Event; EPIP-3, Alert; EPIP-4, Site Area Emergency; EPIP-5, General Emergency; and the Radiological Emergency Plan (REP). In addition, the inspectors verified that licensee evaluators were identifying deficiencies and properly evaluating performance against the performance indicator criteria in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verifications

Licensee records were reviewed to determine whether the submitted performance indicator (PI) statistics were calculated in accordance with the guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2.

.1 Initiating Events Cornerstone

a. Inspection Scope

The inspectors verified the accuracy of the PI for the number of unplanned transients per 7000 critical hours which were reported to the NRC. The inspectors reviewed data applicable for the period of January 1, 2002, through December 31, 2002. The inspectors reviewed control room logs and monthly operating reports to determine the number of reactor critical hours. The inspectors also independently calculated the reported values to verify their accuracy.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors verified the accuracy of the PI for safety system unavailability associated with the auxiliary feedwater system and the emergency AC power system. The inspectors reviewed data applicable for the period of January 1, 2002, through December 31, 2002. The inspectors reviewed control room logs, DG inoperability logs, and operator aid computer history files to determine the number of unavailability hours for safety system. The inspectors also independently calculated the reported values to verify their accuracy.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 Automatic Turbine/Reactor Trip Due to a Ground on the C-phase Main Transformer Potential Device Due to Inadequate Preventative Maintenance

a. Inspection Scope

The inspectors reviewed the licensee’s event critique and PER 03-004747-000, which documented this event in the corrective action program, to verify that the cause of the reactor trip event of March 10, 2003, was identified and that corrective actions were reasonable. The turbine trip/reactor trip was caused by a ground on a connection associated with a C-phase main transformer potential device. This ground caused the main generator backup relay (121GB) to actuate which resulted in the opening of the main generator breakers and a subsequent turbine/reactor trip. The inspectors reviewed plant parameters and verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required.

b. Findings

Introduction: A Green self-revealing finding for inadequate preventative maintenance (PM) was identified.

Description: The licensee determined that the root cause of the event was inadequate PM work instructions associated with the main transformer potential device. The PM contained inadequate guidance on replacing the insulating grease and the tightening of the connectors. In addition, the PM specified the use of an incorrect insulating grease. Improper completion of these tasks had a direct result on the ability of the connection circuit to create a ground fault.

Risk Analysis: The inspectors determined this finding was more than minor because it resulted in an upset in plant stability by causing a reactor trip. While the finding resulted in an actual trip, the inspectors determined that the finding did not contribute to the likelihood of a primary or secondary system LOCA initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. Thus, the finding was screened as Green (very low safety significance). This issue is in the licensee's corrective action program as PER 03-004747-000.

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because it occurred on nonsafety-related secondary plant equipment. Therefore, it is identified as Finding 50-390/2003-02-03, Inadequate Preventative Maintenance of C-phase Main Transformer Potential Device

.2 (Closed) Licensee Event Report (LER) 50-390/2002-005, Loss of Offsite Power Due to a Fire at the Watts Bar Hydroelectric Generating Plant (WBH)

On September 27, 2002, a fire at the WBH resulted in the loss of both preferred offsite power lines to the Watts Bar Nuclear Plant (WBNP). A special inspection team was established by NRC Region II management using the guidance contained in Management Directive 8.3, NRC Incident Investigation Program. The results of that inspection are documented in Special Inspection Report 50-390/02-07. The LER was reviewed by the inspectors and was found to contain no additional information related to the WBH fire. No findings of significance were identified. This LER is closed.

4OA5 Other Activities

Unit 2 Layup Inspection (IP 92050)

a. Inspection Scope

Unresolved item (URI) 50-391/02-04-03, Potential Noncompliance of the Unit 2 Lay-up Process to 10 CFR 50, Appendix B, documented in NRC Integrated Inspection Report 50-390/02-04 and 50-391/02-04, was identified as a result of inspections performed in accordance with IP 92050, Review of Quality Assurance for Extended Construction Delay. Subsequent to issuance of the URI, the licensee presented a formal position paper which was reviewed by the NRC. In addition, the inspectors reviewed previous quality assurance (QA) audit reports.

b. Findings

Introduction: An NCV was identified by the inspectors involving reduction in Unit 2 lay-up equipment preservation which constituted a noncompliance of the lay-up process to 10 CFR 50, Appendix B.

Description: The inspectors performed a Unit 2 lay-up inspection during the previous report period and identified that the applicant had decreased the number of structures, systems and components (SSCs) in the Unit 2 lay-up program. This was identified as a URI to allow the applicant additional time to research and evaluate the problem. Subsequently, the applicant presented their formal response to the inspectors. Upon completion of a review by the inspectors and NRC regional management, the following finding was identified.

The applicant's Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A), or NQAP, follows NRC Regulatory Guide (RG) 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, Revision 2, 5/77, to establish a program for equipment preservation. The NQAP lists several alternatives to the RG; one of which states, "Section 6.5 (last sentence) - During a period of installed storage or extended layup after release of an item from permanent storage, vendor recommendations for preventive maintenance, or an engineering evaluation or an engineering requirements document delineating appropriate maintenance requirements will be followed. Engineering evaluation and engineering requirement documents will consider vendor recommendations." The applicant uses engineering document N3M-935, Plant Layup/Equipment Preservation, to meet this part of the NQAP. However, rather than delineating "appropriate maintenance requirements" for equipment preservation on SSCs, the applicant uses this document to terminate "appropriate maintenance requirements" on various SSCs by applying the note in Section 4.1.1.2.6 of N3M-935, which states, "In some cases it may be more economical to restore equipment as necessary following testing by repairing or replacing it than to lay it up. Approval of economic evaluations shall be documented in accordance with SSIIs (site specific implementing instructions). Economic evaluations do not require SE (site engineering) approval." The applicant has reduced the number of Unit 2 SSCs with maintenance requirements for equipment preservation from over 20,200 down to approximately 340. Additionally, implementation of the equipment preservation program uses an uncontrolled, non-QA computer database. The inspectors also reviewed the applicant's Quality Assurance Program, and recent TVA Nuclear Assurance audit reports SSA9904, SSA0004, SSA0103, SSA0203. The inspectors determined that the applicant's audits did not recognize that reduction of SSCs with maintenance requirements was a change that required prior approval from the NRC or that the applicant's use of an uncontrolled computer database for control of maintenance activities on SSCs was inappropriate. The NRC determined that the reduction of SSCs with maintenance requirements was a change in a site activity associated with QA that did not receive prior approval from the NRC.

Analysis: The inspectors reviewed NRC Manual Chapter 0612 and determined that the applicant's Unit 2 layup process satisfied a traditional enforcement criterion of failure to receive NRC approval for a change in licensee activity. In accordance with the NRC Enforcement Policy, Section IV.B, this violation has been characterized as a severity level IV violation.

Enforcement: 10 CFR 50, Appendix B, requirements apply to all activities affecting the safety-related functions of SSCs; in part, these activities include storing and maintaining. Specifically, Criterion XIII, Identification and Control of Materials, Parts, and Components, requires, in part, that measures will be established to control the handling, storage, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. Contrary to these requirements, the applicant's Unit 2 layup process did not met the criteria of 10 CFR 50, Appendix B, due to the reduction of safety-related SSCs receiving maintenance related to equipment preservation. This violation is not an immediate safety concern and is characterized as a Severity Level IV violation in accordance with the NRC Enforcement Policy, Section IV.B. The applicant has identified this problem in their corrective action program as PERs 02-016994-000 and 03-005681-000; therefore, this violation is treated as NCV 50-391/2003-02-02, Noncompliance of the Unit 2 Layup Process to 10 CFR 50, Appendix B.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. W. Lagergren and other members of licensee management at the conclusion of the inspection on April 7, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Annual Assessment Meeting Summary

Subsequent to the end of this inspection period, on April 16, 2003, the NRC's Chief of Reactor Project's Branch 6 and the Resident Inspectors assigned to the Watts Bar Nuclear (WBN) Plant met with the Tennessee Valley Authority (TVA) to discuss the NRC's Reactor Oversight Process (ROP) and the Watts Bar annual assessment of safety performance for the period of January 1, 2002 - December 31, 2002. The major topics addressed were: the NRC's assessment program, the results of the Watts Bar assessment, and NRC security activities. Attendees included Watts Bar site management and staff, corporate staff, and members of some local government emergency preparedness departments.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML031130024. ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations:

The following finding of very low significance (Green) was identified by the licensee and is a violation of NRC requirements and is characterized as an NCV according to the criteria of Section VI of the NRC Enforcement Policy.

- TS 5.7.1.1.a requires written procedures be established, implemented, and maintained specified in RG 1.33, Appendix A. RG 1.33, Appendix A, Item 1.c, requires procedures be implemented for equipment control such as tagging or clearance. Contrary to the above, the licensee failed to properly determine whether the TS could be met prior to implementing a clearance for the containment spray system in accordance with procedure SPP-10.2, Clearance Program, Revision 3W1. The clearance resulted in a breach of the primary containment boundary. A contributing cause of this finding was related to the cross-cutting area of human performance. This issue was documented in the licencee’s corrective action program as PER 03–003677-000. This finding is of very low safety significance because an engineering evaluation determined that the reactor containment leakage during a postulated design basis accident would have been a small fraction of that allowed by design.

SUPPLEMENTAL INFORMATION
KEY POINTS OF CONTACT

Licensee personnel

D. Boone, Radiological Control Manager
L. Bryant, Plant Manager
S. Casteel, Radiological and Chemistry Control Manager
D. Kulisek, Assistant Plant Manager
W. Lagergren, Site Vice President
P. Pace, Licensing and Industry Affairs Manager
K. Parker, Maintenance and Modifications Manager
T. Wallace, Operations Manager
J. West, Site Nuclear Assurance Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-390/2002-005	LER	Loss of Offsite Power Due to a Fire at the Watts Bar Hydroelectric Generating Plant (WBH) (Section 4OA3.2)
50-391/02-04-03	URI	Potential Noncompliance of the Unit 2 Layup Process to 10 CFR 50, Appendix B (Section 4OA5)

Opened and Closed

50-390/2003-02-01	NCV	Failure to Adequately Define the Root of the Weld or Root Pass in G-29 General Welding Procedure Specification (Section 1R02)
50-391/2003-02-02	NCV	Noncompliance of the Unit 2 Layup Process to10 CFR 50, Appendix B (Section 4OA5)
50-390/2003-02-03	FIN	Inadequate Preventative Maintenance of C-phase Main Transformer Potential Device (Section 4OA3.1)

Discussed

50-390/02-04-02	URI	Inadequate Surveillance Instruction Resulting in Gas Accumulation in ECCS Piping (Section 1R22)

LIST OF DOCUMENTS REVIEWED

Section 1R01

- Plant Instruction 1-PI-OPS-1-FP, Freeze Protection
- Technical Instruction 10.17, Freeze Protection Program
- AOI-8, Tornado Watch or Warning
- PER 03-005603-000, NRC-identified problem regarding improper securing of loose lumber and scaffolding materials in yard contrary to AOI-8, Tornado Watch or Warning.

Section 1R02

Evaluations
- DCN 50948-A, Re-gear Valves for Additional Thrust Margin, Rev. 0
- DCN 50965, Clarify Rad Monitor Setpoint Scaling Document
- EDC No. E51334-A, Rev. A, Document Revision for 10% Tube Plugging Margin and Associated UA Value for the Containment Spray Heat Exchanger
- TACF 1-02-005-246, Block Tripping Function for Sudden Pressure Trip Device on USST 1A & 1B, Rev. 0
- TACF 1-02-011-043, Closure of RCS Loop 3 Hot Sample Isolation Valve, 1-SMV-68-578
- FSAR Change 1712, Optional Locations for Processing Service Water in Auxiliary Building

Screened Out Items
- DCN D-50565-A, Change Shutdown Board Degraded Voltage Relay Time Delay, Rev. 0
- DCN 50837-A, Replace Existing Sudden Pressure Relays with 2 out of 3 Logic, Rev. 0
- DCN No. 50895, Rev. A, Add Clean Out Ports in Piping Upstream of the Auxiliary Feedwater Isolation Valve for the ERCW System
- EDC 50951, Reduce Ice Weight Requirements
- DCN 51072A, Replace Analog Rod Position Indication System Electronics with Upgraded System, Rev. 0
- DCN 51125, Modify MDAFP Start Circuit to Improve Seal-in Function, Rev. 0
- EDC E51131A, Rev. A, Evaluation on Adding Alternate Purification Media to Mixed Bed Demineralizer to Complement Zinc Injection
- EDC 51307-A, Increase EQ Life of Solenoid Valves, Rev. 0
- TI-12.15, 161 kV Offsite Power Requirements, Rev. 0
- 1-SI-0-2B-01, Change to Acceptance Criteria for RCS Flow, Rev. 18
- 1-SI-211-3-A, Change Shutdown Board Degraded Voltage Relay Time Delay, Rev. 0
- G-29, Changes to Welding Specification on Parts A (Rev. 59) and B (Rev. 45)
- G-55, Changes to Technical and Programmatic Requirements for Protective Coatings, (Rev. 11)

Problem Evaluation Reports (PERs)
- 02-009411-000, Work orders with temporary configurations open beyond 90 days
- 03-001872-000, Not all 50.59 documents produced by WBN engineering entered separately into EDMS
- 03-003075-000, NRC identified issue related to definition of root pass in TVA Specification G-29

Audits and Self-Assessment Documents

- 50.59 Program Valley Wide Assessment, CRP-ENG-01-012/WBN-SA-01-06
- Self-Assessment, Maintenance and Modifications Performance Evaluation, WBN-M&M-02-004, July 2002
- Self-Assessment, Installation and Testing of Modifications, WBN-M&M-01-009, 3/2001

Other Documents

- SSP-9.3, Rev. 7, Plant Modifications and Engineering Change Control
- SSP-9.4, Rev. 5, 10CFR50.59 Evaluations of Changes, Tests, and Experiments
- 01-002355, Implementation work order for DCN No. 50895
- SPP-5.3, Section D, Purchase Specifications for Resin, Purification Demin Macroporous Mixed (Purolite NRW-35, 160, and 600), Rev. 15
- Purchase Specification for AMBERLITE IRN217, Mixed Bed Ion Exchange Resin
- Drawing No. 47W450-2C and -3C, Rev. 2, Mechanical Essential Cooling Water
- G-29A-S02, G-29 General Welding Procedure Specification (GWPS) for American Society of Mechanical Engineers (ASME) and American National Standards Institute (ANSI), GWPS 1.M.1.2 Rev2, Addendum 1, Rev. 0
- G-29 Detailed Welding Procedure Specification (DWPSs) ASME/ANSI-GWPS 1.M.1.2, DWPS GM11-O-1-N, Revs. 2 and 3
- G-29A-S03, ASME/ANSI DWPS, DWPS GT88-O-2-N, Rev. 1
- G-29A-S04, AWS D1.1 DWPS, DWPS GT-U-1**,** Rev. 3
- G-29B-S02A Standard Material Specifications, PF-1051, Rev. 3

<u>Section 1R04</u>

- SOI-74.01, Residual Heat Removal System
- SOI-62.01, CVCS - Charging and Letdown
- SOI-63.01, Safety Injection System
- SOI-72.01, Containment Spray System
- SOI-72.02, RHR Spray System
- SOI-82.02, Diesel Generator (DG) 1B-B
- SOI-82.04, Diesel Generator (DG) 2B-B
- SOI-82.01, Diesel Generator (DG) 1A-A
- UFSAR Section 8.3, Onsite (Standby) Power System
- UFSAR Section 9.5, Other Auxiliary Systems
- UFSAR Section 5.5.7, Residual Heat Removal System
- TS 3.8.1, AC Sources - Operating
- TS 3.5.2, ECCS - Operating
- TS 3.6.6 Containment Spray System

Section 1R05

- PER 03-003955-000, NRC-identified problem with two cable tray penetrations with missing Kaowool.

Section 1R06

- Maintenance Instruction (MI)-17.001, Flood Preparation - Ventilation of Steam Valve Rooms, Rev. 5
- MI-17.002, Flood Preparation - Opening of Aux Bldg Railroad Hatchways and Access Door, Rev. 7
- MI-17.004, Movement of Equipment, Flood Mode Preparation, Rev. 6
- MI-17.021, Installation of Spool Pieces Between ERCW System and Component Cooling Water, Revision 6
- MI-17.022, Flood Preparation - Installation of Spool Pieces Between SFPC System and RHR System, Rev. 5
- WO 00-016898-000, PM of class 1E manholes
- WO 01-007732-000, PM of class 1E manholes
- WO 01-016117-000, PM of class 1E manholes
- WO 02-006125-000, PM of class 1E manholes

Section 1R11

- TRN-12 Certification of Simulators, Rev. 2
- WBN Simulator Engineering Guideline: Simulator Problem Reports and Simulator Design Change Requests
- Watts Bar Nuclear Plant (WBN) Unit 1 - 10CFR55.45 Plant-Referenced Simulator Certification Report, March 2, 1999
- Simulator Transient Test: Transient # 1: Manual Reactor Trip (2002 and 2003)
- Simulator Transient Test: Transient # 8: Loss of Coolant Accident maximum rate with LOOP (2003)
- Simulator Transient Test: Transient # 4: Four Loop RCP Trip (2003)
- Steady State Test @ 100% power level (2003)
- Procedures Test GO-4 Normal Operations
- Procedures Test GO-5 Unit shutdown from 30% Power to Hot Standby
- Procedures Test GO-6 Unit Shutdown from Hot Standby to Cold Shutdown
- Procedures Test GO-10 Reactor Coolant System Drain and Fill Operation
- Simulator Malfunction Periodic Test # FW05: Loss of all Feedwater
- Simulator Malfunction Periodic Test # ED01: Loss of all Offsite power
- Simulator Malfunction Periodic Test # RX18: Tavg Control signal failure
- Simulator Malfunction Periodic Test # RD07: Dropped Rod
- Simulator Malfunction Periodic Test # TH04: Pressurizer Safety Failure
- Simulator Current/Open Problem Reports (as of 3/5/03)
- Problem Report #2156 Request to allow make replay file capture capability time variable.
- Problem Report #2460 Request for new malfunction for leak in Seal Water Return Heat Exchanger.

- Problem Report #2486 Investigate reason for change in RCS temperature and Main Steam pressure between last year and this year's Transient Test (TT1) And tune as necessary.
- Problem Report #2487 Investigate reason for change in turbine first stage pressure between last year and this year's Transient Test (TT1) and tune as necessary.
- Problem Report #2488 Investigate reason for change in Main Steam header pressure decay rate between last year and this year's Transient Test (TT1) and tune as necessary.
- Problem Report #2490 Add capability to locally open 1-FCV-67-65 and 2-FCV-67-65 to allow performing steps 7 and 8 of AOI-43.02
- Problem Report #2495 Investigate reason for lower containment temperature drops while wide range containment pressure continues to rise during malfunction test RP02 AND RC07. Tune as necessary.
- Problem Report #2497 Adjust feedwater pressure to more closely match the plant ($\pm$2%)
- Problem Report #2498 Adjust PRT pressure to more closely match the plant ($\pm$2%)
- 42 closed Simulator Problem Reports
- Core Performance Test: 1-SI-0-10 Shutdown Margin
- Core Performance Test: 1-SI-92-1 NIS Daily Comparison
- Core Performance Test: TI-6.001 Board Calorimetric

Section 1R12

- List of PERs over previous 24 months regarding B MCR chiller
- WBN Maintenance Rule Information summary
- WBN System Status, 4th quarter, FY02
- PER 03-000028-000, B MCR chiller failure to start on 1/3/03
- WO 03-001238-000, MCR B chiller did not trip on low suction pressure
- PER 03-003755-000, Condenser temperature control valve leaking by seat identified during troubleshooting
- PER 03-003901-000, SSPS A213 board failure on 10-25-02
- Main control room logs (06-10-02, 10-25-02)
- WBN-VTD-W120-2454, table 6-1, logic C Test switch
- Drawing 1082H70, SSPS - Low Steamline Pressure
- Surveillance task sheet (SPP-8.2), procedure 1-SI-99-10-B, dated 10-25-02
- MMDP-3, Guidelines for Planning and Execution of Troubleshooting Activities
- IMI-99.030, Solid State Protection System Universal Logic Board Test
- SSPS Unavailability system engineer's database

Section 1R14

- E-0, Reactor Trip or Safety Injection
- ES-0.1, Reactor Trip Response

Section 1R17

PEG Packages
- 2000-31830, U2-U1 Transfer - ASME valve
- 2000-85679, Rev. 1, U2-U1 Transfer - Snubber & Support 2-01B-003
- 2000-85682, U2-U1 Transfer - Nuclear Instrumentation System Isolation Amplifier
- 2001-99956, U2-U1 Transfer - Westinghouse Barton pressure transmitter
- 2002-91002, U2-U1 Transfer - TDAFW Trip and Throttle Valve

Assessments
- 50.59 Program Valley-Wide Assessment, CRP-ENG-01-012/WBN-SA-01-06
- Self-Assessment, Maintenance and Modifications Performance Evaluation, WBN-M&M-02-004, July 2002
- Self-Assessment, Installation and Testing of Modifications, WBN-M&M-01-009, March 2001

Section 1R19

- 02-014544-000,Take oil sample and change oil on SI 1B-B motor
- 02-014477-000, Repair boron leak on 1-PT-63-18
- 02-014593-000, Cuno filter inspection, pump oil change, heat exchanger cleaning, pump coupling lubrication
- WO 02-018235-000, Safety injection pump room cooler motor, clean/inspect/meggar
- WO 02-018085-000, Safety injection pump Cuno filter inspection, oil heat exchanger cleaning and eddy current inspection
- WO 01-015013-000, Replacement of alarm relays
- WO 03-003995-000, Repair 86 LOR relay failure to toggle
- 0-SI-82-19-A, 184-Day Fast Start and Load Test DG 2A-A
- SOI-82.03, Diesel Generator (DG) 2A-A
- MI-82.003, Two-Year Diesel Generator Engine Inspection
- WO 03-001246-000, Obtain oil sample and freon sample from 'B' MCR chiller
- WO 02-015784-000, Disassemble and inspect 1-CKV-072-0507-B (1B-B CSP suction check valve)
- WO 01-016793-000, Repair leak at 1-PDI-072-0016 (1B-B CSP diff. press.)
- PER 03-006204-000, NRC identified problem with MNT/OPS unaware of normal operating temperature and pressure requirements specified for PMT on 1-ITS-82-A2.

Section 1R20

- Technical Instruction (TI)-127, Reactor/Turbine Trip Report, Event Critique, Root Cause Analysis
- PER 03-004747-000, Turbine/reactor trip on actuation of 121GB generator backup relay
- SPP-7.0, Work Control and Outage Management
- SPP-7.1, Work Control Process
- SPP-7.2, Outage Management
- TI-124, Equipment to Plant Risk Matrix
- Forced outage schedule

Section 1R22

- 1-SI-63-10-A, ECCS pumps and discharge pipes venting - train A
- TS 3.5.2 ECCS - Operating
- TS Bases 3.5.2 ECCS - Operating
- UFSAR 6.3, Emergency Core Cooling System
- UFSAR 5.5.7, Residual Heat Removal System
- UFSAR 9.2.2, Component Cooling System
- UFSAR 8.3.1, A.C. Power System
- TS 3.7.7, Component Cooling System
- TS 3.8.1, AC Sources-Operating
- PER 03-001920-000, NRC-identified problem involving use of a gauge for differential pressure measurement that did not match procedure requirements regarding increments

Section 4OA1

- PER 03-006237-000, NRC-identified problem with documentation of control room log entries