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

April 29, 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: WATTS BAR NRC INTEGRATED INSPECTION REPORT 05000390/2005002 AND 05000391/2005002

Dear Mr. Singer:

On March 31, 2005, 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 results which were discussed on April 6, 2005, 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.

This reports documents one NRC-identified finding concerning procedure adherence which resulted in pressurizer power-operated relief valve (PORV) actuations. This finding has potential significance of greater than very low safety significance. The finding does not present an immediate safety concern because your staff has addressed the procedural problems that led to the PORV actuations. The finding is unresolved pending significance determination assessment. In addition, this reports documents four NRC-identified findings and two self-revealing findings of very low safety significance (Green). The six findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they are entered into your corrective action program, the NRC is treating these six findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. A licensee-identified NCV that was also determined to be of very low safety significance is listed in Section 4OA7 of this report. If you contest any NCV 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 DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Watts Bar facility.

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 Nos. 50-390, 50-391
License No. NPF-90 and Construction
Permit No. CPPR-92

Enclosure: NRC Inspection Report 05000390/2005002, 05000391/2005002
w/Attachment: Supplemental Information

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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 Nos:	05000390/2005002, 05000391/2005002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Watts Bar Nuclear Plant, Units 1 and 2
Location:	1260 Nuclear Plant Road Spring City TN 37381
Dates:	January 1 through March 31, 2005
Inspectors:	J. Bartley, Senior Resident Inspector J. Reece, Resident Inspector A. Vargas-Mendez, Reactor Inspector (Section 1R08) S. Vias, Senior Reactor Inspector (Section 1R08) W. Loo, Senior Health Physicist (Sections 2PS2, 2OS1, 2OS2, 2PS3) R. Carrion, Project Engineer (Sections 2OS2, 2OS1, 4OA1) J. Kreh, Emergency Preparedness Inspector (Sections 2OS1, 2OS2)
Approved by:	Stephen J. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000390/2005002, 05000391/2005002, 01/01/2005 - 03/31/2005, Watts Bar, Units 1 & 2; Equipment Alignment, Maintenance Effectiveness, Post-Maintenance Testing, Refueling Outage, Access Control to Radiologically-Significant Areas, Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

The report covered a three-month period of routine inspection by resident inspectors and announced inspections by regional reactor inspectors, health physicists, and a project engineer. The significance of an issue is indicated by its color (Green, White, Yellow, Red) using the Significance Determination Process in 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. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification (TS) 3.8.1 when the 1A-A Diesel Generator (DG) was inoperable due to both ventilation exhaust fans being out of service. Surveillance Requirement 3.8.1.1 was not performed as required within one hour. A senior reactor operator issued a hold order which tagged out the exhaust fans and did not recognize that this action made the DG inoperable.

  The finding is more than minor because it affected the availability attribute of the Mitigating System Cornerstone. The DG would have started and run but manual action would have been required to shut a breaker to provide power to one of the fans for continued operation. The finding was of very low safety significance (Green) because it did not result in a loss of function per Generic Letter 91-18, did not represent an actual loss of safety function for a single train greater than its TS allowed outage time, and was not potentially risk-significant due to possible external events. The cause of this finding impacts the human performance cross-cutting area. (Section 1R04)

- Green. The inspectors identified a non-cited violation of TS 5.7.1.1, which requires that written procedures be implemented covering the activities in the applicable procedures recommended by Regulatory Guide 1.33, including procedures for maintenance. The procedure and work order for post-maintenance testing (PMT) for a residual heat removal (RHR) pump seal replacement were not followed. The PMT was performed at 215 pounds per square inch gauge (psig) instead of the specified 275-300 psig but was signed as complete and acceptable.

  The finding is more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences to the reactor core and the associated cornerstone attribute of human performance. A failure to perform the PMT as specified had a credible impact on reactor safety because the 1A RHR pump mechanical seal subsequently failed. The finding was of very low safety significance (Green) because it only affected one train of RHR and the steam generators (SGs) were available for heat removal. The cause of the finding impacts the cross-cutting area of human performance. (Section 1R19)

- Green. The inspectors identified a non-cited violation of 10 CFR 50.65 (a)(4) which requires that the licensee assess and manage the increase in risk that may result from the proposed maintenance activities. The licensee did not establish a pre-approved contingency plan for an Orange risk condition involving electrical power as required by procedure Standard Programs and Processes (SPP)-7.2, Outage Management.

  The licensee's failure to establish a contingency plan for a high risk condition is more than minor because it impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences to the reactor core and the associated cornerstone attribute of human performance. The finding did not increase the likelihood of a loss of offsite power or degrade the licensee's ability to cope with a loss of offsite power, resulting in the characterization of very low safety significance (Green). The cause of the finding, failure to implement outage procedural requirements, impacts the cross-cutting area of human performance. (Section 1R20.3)

Cornerstone: Barrier Integrity

- TBD. The inspectors identified a finding associated with TS 5.7.1.1 requirements for procedure adherence which resulted in pressurizer power-operated relief valve (PORV) actuations. The finding is unresolved pending significance determination assessment.

  The inspectors determined that procedural noncompliances had a credible impact on safety involving the challenge of reactor coolant system (RCS) integrity by PORV actuations and the challenge of RCS inventory through the loss of inventory via the open PORVs. The finding was more than minor because it impacted the Barrier Integrity Cornerstone objective to provide reasonable assurance that the RCS physical design barrier protects the public from radionuclide releases caused by accidents or events and the associated cornerstone attributes of human performance and procedure quality. The inspectors reviewed MC 0609, Appendix G, and determined that the finding required quantitative assessment consisting of a Phase 3 analysis because it affected the cold over-pressure mitigation or low temperature over-pressure system required by TS. The cause of the finding impacts the cross-cutting area of human performance. (Section 1R20.2)

- Green. The inspectors identified a non-cited violation of TS 5.7.1.1 which requires that written procedures be implemented covering the activities in the applicable procedures recommended by Regulatory Guide 1.33, including procedures for maintenance. The licensee failed to follow procedures for work control which resulted in de-tensioning the pressurizer PORV mounting nuts when it was a designated operable vent path per TS.

  This finding had a credible impact on safety involving the challenge of RCS integrity by the performance of work on the pressurizer PORVs. The finding was more than minor because it impacted the Barrier Integrity Cornerstone objective to provide reasonable assurance that the RCS physical design barrier protects the public from radionuclide releases caused by accidents or events and the associated cornerstone attributes of human performance. The licensee had the functional ability to establish an alternate core cooling path in the event of a loss of RHR based on the licensee's conclusion that the venting capability of the detensioned PORVs was still functionally available. This resulted in the characterization of Green (very low safety significance). The cause of the finding impacts the cross-cutting area of human performance. (Section 1R12)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a non-cited violation of TS 5.7.1.1 for failure to implement effluent monitoring quality assurance design guidance used to demonstrate representative sampling for the Auxiliary Building Ventilation Monitor (0-RE-90-101) compensatory sampler. This issue was initially identified as an Unresolved Item following an inspection in December 2004.

  This finding is more than minor because it is associated with the program and process attribute of the Public Radiation Safety Cornerstone and affects the cornerstone objective to assure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The failure to conduct appropriate evaluations to assure representative sample collection from the U1 plant ventilation exhaust streams using the compensatory sampling configuration could result in inaccurate measurement of airborne particulate radionuclides in effluent samples and inaccurate dose estimates to members of the public. This finding was evaluated using the Public Radiation Safety SDP and is of very low safety significance (Green) because the licensee's ability to assess offsite dose was not impaired and doses to the public were below 10 CFR 50, Appendix I, and 10 CFR 20.1301 limits. (Section 4OA5.2)

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing non-cited violation of TS 5.11.1 was identified for an un-posted high radiation area. The high radiation area was created when lower containment coordinators sent contaminated trash out of lower containment to upper containment without properly notifying the radcon radwaste technician.

  The finding was more than minor because it was associated with the Occupational Radiation Safety Cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material. The uncontrolled high radiation area created the potential for unplanned and unintended dose to individuals working in the proximity of the trash. The finding was of very low safety significance because the dose rates were not sufficient to produce a substantial potential for an exposure in excess of regulatory limits. This finding impacts the cross-cutting aspect of human performance. (Section 2OS1)

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 actions are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Unit 1 operated at or near 100 percent power until February 22, 2005, when it was shut down to start the Cycle 6 refueling outage (RFO). Unit 1 was returned to service on March 31, 2005. Unit 2 remained in a suspended construction status.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted three equipment alignment partial 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, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and Technical Specifications (TSs) 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. In addition, the inspectors reviewed a hold order for the 1A-A diesel generator (DG) ventilation exhaust fans which the licensee determined did not render the DG inoperable. The inspectors reviewed the hold order to determine if the DG was operable with the exhaust fans out of service.

- 1A-A DG due to emergent work on 1B-B diesel generator day tank fuel oil transfer pump
- 1B residual heat removal (RHR) train with 1A RHR pump inoperable for component outage
- B train spent fuel pool (SFP) cooling and B train component cooling system (CCS) with A train essential raw cooling water (ERCW) drained for maintenance and high pressure differential on 1B ERCW strainer

b. Findings

Introduction. A Green non-cited violation (NCV) was identified by the NRC regarding the failure to comply with TS 3.8.1 when the 1A-A DG was inoperable due to both ventilation exhaust fans being out of service.

Description. While performing a plant status review of the main control room logs on February 16, the inspectors identified a series of log entries for issuing and picking up hold order 1-30-0705A for the 1A-A DG air intake damper. There were no log entries for entering or exiting TS 3.8.1 for one DG inoperable. The inspectors reviewed the hold order and determined that it tagged out the intake dampers and both of the exhaust fans. System Operating Instruction (SOI) 82.01, Diesel Generator 1A-A, states that for outside air temperatures less than 86 degrees Fahrenheit (EF) a minimum of one exhaust fan is required for DG operability. Outside air was less than 86 EF throughout the licensee's work. The inspectors discussed the hold order with Operations personnel and determined that the senior reactor operators (SROs) who reviewed and approved the hold order mistakenly thought the hold order was tagging out the DG electric board room exhaust fan, which is not required for DG operability with outside temperatures less than 80 EF. An SRO mis-read a table in the procedure and did not recognize that the DG was inoperable with both exhaust fans tagged out. Two other SROs and two auxiliary unit operators involved in the hold order process did not catch the error.

Analysis. The finding adversely affected the availability of emergency AC power supply during a loss of offsite power. The inspectors referred to Manual Chapter (MC) 0612 and determined that the finding is more than minor because it affected the availability attribute of the Mitigating System Cornerstone. The DG would have started and run but manual action would be required to shut a breaker to provide power to one of the fans for continued operation. The inspectors evaluated this finding using MC 0609 and determined that it was of very low safety significance (Green) because it did not result in a loss of function per Generic Letter 91-18, did not represent an actual loss of safety function for a single train greater than its TS allowed outage time, and was not potentially risk-significant due to possible external events. The cause of this finding impacts the human performance cross-cutting area.

Enforcement. TS 3.8.1.B requires that with one required DG inoperable then perform SR 3.8.1.1 for the offsite circuits within one hour. Contrary to this, DG 1A-A was inoperable from February 15th at 11:58 p.m. through February 16th at 5:59 a.m., and SR 3.8.1.1 was not performed. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as PER 76827, this violation is being treated as an Non-cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000390/2005002-01, DG Fans Removed From Service and Tech Spec SR 3.8.1.1 Not Peformed.

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.

- Control room emergency ventilation system (CREVS)
- Vital DC Boardroom I
- Vital DC Boardroom II
- Vital DC Boardroom III
- Vital DC Boardroom IV
- A 6.9-kV SDBR
- B 6.9-kV SDBR
- Motor-driven auxiliary feedwater (MDAFW) pumps/CCS pumps

.2 Fire Protection - Drill Observation

Based on previous a previous performance problem as documented in Integrated Inspection Report 05000390/2004004 and 05000391/2004004, the inspectors observed two fire drills. On January 6, 2005, the inspectors observed an unannounced fire drill performed at the 250-VDC Board Room. On March 30, 2005, the inspectors observed the fire brigade's response to an announced fire drill in the raw cooling water transformer room at the intake pumping structure. The drill was observed to evaluate the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient fire fighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's program for maintenance and testing of three risk-important heat exchangers in the essential raw cooling water (ERCW) system. The inspectors reviewed three heat exchangers because of the past history of silt accumulation and clams in the ERCW system. Specifically, the review included the program for testing and analysis of the 'A', 'B', and 'C' component cooling system (CCS) heat exchangers which was cleaned, inspected, and evaluated by WOs 03-821246-000, 04-811505-000, and 05-812466-000 during the Cycle 6 refueling outage. The inspectors observed the physical condition of the heat exchangers during the cleaning activities and verified that the frequency of inspection was sufficient to detect degradation prior to loss of heat removal capabilities below design requirements, that the inspection results were appropriately categorized against pre-established engineering acceptance criteria including the impact of tubes plugged on the heat exchanger performance, and that the licensee had developed adequate acceptance criteria for bio-fouling controls.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Reactor Coolant Pressure Boundary and Piping Systems

a. Inspection Scope

The inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk-significant piping system boundaries for Unit 1. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examination and code components in order of risk priority as identified in Section 71111.08-03 of NRC Inspection Procedure 71111.08, Inservice Inspection Activities, based upon the ISI activities available for review during the onsite inspection period.

The inspectors conducted an onsite review of nondestructive examination (NDE) activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspectors observed the following examinations:

Ultrasonic Testing (UT):
- Safety Injection Pipe to Pipe Weld SIF-D079-11

Visual Testing (VT):
- Pressurizer Safe-End Welds WP-11-SE-12, WP-11-SE-13, WP-11-SE-14, WP-11-SE-15
- Pressurizer Spray Nozzle Weld WP-11-SE
- Pressurizer Safety Nozzles Welds WP-13-SE, WP-14-SE, WP-15-SE
- Pressurizer Safety Relief Weld WP-12-SE

Inspectors also reviewed the following records of completed NDE activities:

UT:
- Reactor Coolant Pump (RCP) #3 Shaft
- Safety Injection Pipe to Elbow Weld SIF-D079-01
- Safety Injection Pipe to Pipe Weld SIF-D092-15

Flow Accelerated Corrosion Ultrasonic Testing (FAC/UT):

- Steam Generator 4 Elbow 103BE134
- Steam Generator 1 Pipe Reducer 115X005

VT:
- Reactor Coolant Bolt RC-04-BC

The inspectors reviewed the licensee's ASME Section XI Inservice Inspection Summary Report- 5th Refueling Cycle to verify that the licensee had no relevant/recordable conditions/indications accepted for continued service.

The inspectors reviewed pressure boundary welds for Code Class 1 or 2 systems which were completed since the last refueling outage and during the present refueling outage, to verify that the welding acceptance and pre-service examinations (e.g., visual, dye penetrant, radiography, and weld procedure; and personnel qualifications) were performed in accordance with the ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed welds associated with the following work activities:

RFO 6:
- 2-inch (") CVCS Seal Water Injection Bonnet to Seal Weld 1-062B-T118-01A
- 2" CVCS Seal Water Injection Bonnet to Seal Weld 1-062B-T217-01A
- Valve 1-DRV-068-0581 Replacement Weld 1-068B-T003-08A

The inspectors performed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. In addition, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.2 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors reviewed the Unit 1 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.

The inspectors conducted an onsite record review as well as an independent walkdown of the BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. In particular, the inspectors reviewed information to determine that the visual examinations focused on locations where boric acid leaks can cause degradation of safety-significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed engineering evaluations performed for boric acid found on reactor coolant system piping and components to verify that the minimum design code required section thickness had been maintained for the affected component(s). Specifically, the inspectors reviewed:

- Initial Evaluation of Borated Water Leaks WID 03-016076-000, PER 03-015746-000, Residual Heat Removal Heater Test Vent
- Initial Evaluation of Borated Water WID 03-013204-000, PER 03-015746-000, Reactor Coolant Pump #4 Injection Flow Electronic Transmitter
- Initial Evaluation of Borated Water WID 03-015777-000, PER 03-15746-000, Cold Leg 1 Safety Injection Check Valve
- Initial Evaluation of Borated Water WID 03-016207-000, PER 03-015746-000, Residual Heat Removal Line 1-Pipe-074-B

The inspectors reviewed licensee corrective actions implemented for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code and 10 CFR 50, Appendix B, Criterion XVI. Specifically, the inspectors reviewed:

- PER 78273, Boron Leakage
- PER 75192, Valve 1-FCV-72-21, RWST to CNTMT Spray Pump leaking
- PER 75536, Boron Leakage
- PER 75862, Boron Leakage

b. Findings

No findings of significance were identified.

.3 Steam Generator (SG) Tube ISI

a. Inspection Scope

The inspectors reviewed the Unit 1 SG tube examination activities conducted pursuant to TS and ASME Code, Section XI, requirements.

The inspectors reviewed the SG examination scope, expansion criteria, eddy current testing (ET) acquisition procedures, ET analysis procedures, the SG Operational Assessment, in-situ tube pressure testing procedures and records and examination reports to confirm that:

- In-situ SG tube pressure testing screening criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620, Steam Generator In Situ Pressure Test Guidelines, and the licensee's screening criteria included allowances for ET probe flaw sizing error bands;

- The in-situ SG tube pressure testing screening criteria were properly applied in terms of specific tubes selected and number of tubes required to be tested based upon a review of a list of tubes with measured/sized flaws (e.g., tubes with I-Code type ET calls);

- In-situ SG tube pressure testing was conducted in accordance with the licensee procedures and that these procedures were consistent with (EPRI) TR-107620, Steam Generator In Situ Pressure Test Guidelines (e.g., pressure verses time traces, pressure achieved and hold times);

- In-situ SG tube pressure test results of degraded tubes met the licensee's performance criteria for tube structural and leakage integrity. Additionally, the inspectors reviewed the licensee's tube integrity performance criteria to determine that it had been developed using a methodology which was consistent with that discussed in EPRI TR-107621, Steam Generator Integrity Assessment Guidelines, Revision 1;

- The numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee's previous outage operational assessment predictions;

- The SG tube ET examination scope and expansion criteria were sufficient to identify tube degradation based on site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures and plant TS requirements. In addition, the inspectors reviewed the SG tube ET examination scope to determine if it was consistent with that recommended in EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6, and included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, U-bends and support plates;

- The licensee did not identify any new tube degradation mechanisms other than what was predicted in the SG tube degradation assessment;

- The SG tube repair criteria and process (plugging and sleeving) implemented were consistent with TS requirements and that the licensee was only applying the TS plugging limit at tube wear locations (e.g., licensee was not depth-sizing cracks to allow returning cracked tubes to service);

- The licensee identified degraded tube SG2R26C21, which was the source of the primary-to-secondary leakage that reached a maximum of 2.4 gallons per day during the previous operating cycle, and also that the cause of tube degradation was identified and the tube placed on the tube repair and in-situ testing list;

- The ET probes and equipment configurations used to acquire ET data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6;

- The licensee adequately examined for loose parts indications;

- The licensee adequately evaluated for any contractor deviations from their ET data acquisition or analysis procedures or EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6.

The inspectors performed a review of SG ISI-related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. In addition, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requirements.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On February 9, 2005, the inspectors observed operators in the plant simulator during licensed operator annual requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and 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. The inspectors observed a shift crew's response to scenario 3-OT-SRT0059B, Refueling Outage Just-in-Time Startup, Respond to an Ejected Control Rod/Reactor Trip.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed a performance-based problem relating to the auxiliary control air system exceeding the maintenance rule (MR) performance criteria and a maintenance work control issue. The focus of the reviews was to assess the effectiveness of maintenance efforts that apply to scoped structures, systems, or components (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) appropriate work practices; (2) identification and resolution of common cause failures; (3) scoping in accordance with 10 CFR 50.65; (4) characterization of reliability issues; (5) charging unavailability time; (6) trending key parameters; (7) 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification; and (8) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). Additional documents reviewed by the inspectors are listed in the attachment to this report.

- PER 73762, Train B auxiliary control air system has failed MR performance criteria of not more than one functional failure in a 24-month period.
- PER 77744, NRC-identified problem with failure to follow required work process on WOs 04-813661-000 and 04-813662-000

b. Findings

Introduction. A Green NCV was identified by the NRC regarding a failure to implement procedures which impacted TS requirements of the pressurizer power-operated relief valve (PORV).

Description. On February 25, 2005, the inspectors observed the morning outage meeting in which a comment was made concerning a completed work activity to de-tension the pressurizer PORVs. The inspectors contacted the SRO to verify that the appropriate TS had been entered as current plant conditions required one PORV and one RHR suction relief valve to be operable per TS 3.4.12, Cold Overpressure Mitigating System (COMS). The inspectors determined that none of the control room personnel were aware of the existing work activity on the PORVs. Work Orders (WOs) 04-813661-000 and 04-813662-000 (one for each PORV) were originally planned to remove the PORVs after the licensee had considered the PORVs out-of-service. However, due to a critique item from the previous refueling outage, the licensee had added a schedule activity to de-tension the PORVs prior to their actual removal. The inspectors reviewed the WOs and determined the following:

- A technical review was not performed for a handwritten note that changed the work scope to de-tension the PORVs prior to other work directed by the WO. This is contrary to the requirements of MMDP-1, Maintenance Management System, Section 3.2.5, Revisions to Work Orders, and Section 3.8.1, Independent/Technical Review, in regards to scope changes and required reviews.

- Prerequisite steps of the WO were signed and dated February 26, 2005, when the work designated by the additional note was reported to be complete on February 24, 2005, to outage work control. This is contrary to SPP-2.2, Administration of Site Technical Procedures, Section 3.2.2, Reference Use Procedure, in regards to completion of appropriate signoffs to verify that each segment of the procedure has been performed.

- Maintenance Instruction (MI)-68.021, Pressurizer PORV Maintenance, Section 1.3, Frequency and Conditions, states: "This Instruction can be performed in mode 5 or 6, provided the pressurizer has been vented per MI-68-020 and the requirements of Technical Specification 3.4.12 are adhered to." Contrary to this procedure condition, TS 3.4.12 was not adhered to in that the PORVs were de-tensioned while still considered operable to fulfill TS 3.4.12..

Analysis. The inspectors determined that the procedural noncompliances identified above had a credible impact on safety involving the challenge of reactor coolant system (RCS) integrity by the performance of work on the pressurizer PORVs. The inspectors reviewed MC 0612 and determined that the finding was more than minor due to the impact on the Barrier Integrity Cornerstone objective to provide reasonable assurance that the RCS physical design barrier protects the public from radionuclide releases caused by accidents or events and the associated cornerstone attributes of human performance. The inspectors reviewed MC 0609, Appendix G, and determined that the finding did not require a quantitative assessment because the licensee had the functional ability to establish an alternate core cooling path in the event of a loss of RHR based on the licensee's conclusion that the venting capability of the detensioned PORVs was still functionally available. This resulted in the characterization of Green (very low safety significance). The errors that caused the failure to follow three procedures for controlling maintenance resulting in this finding involved the cross-cutting area of human performance.

Enforcement. Technical Specification 5.7.1.1 requires that written procedures shall be implemented and maintained covering the activities in the applicable procedures recommended by Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978 of which Part 9.e requires general procedures for control of maintenance, repair and replacement work; Part 1.d requires administrative procedures for control of procedure adherence; and Part 9.a requires procedures for performing maintenance activities affecting safety-related equipment. Contrary to the above on February 25, 2005, the inspectors identified that:

- The requirements of MMDP-1, Maintenance Management System, Section 3.2.5, Revisions to Work Orders and section 3.8.1, Independent/Technical Review, in regards to scope changes and required reviews, a technical review was not performed for a change to WOs 04-813661-000 and 04-813662-000 involving a handwritten note stating a change in the work process to de-tension prior to other work directed by the WO.
- The requirement of SPP-2.2, Administration of Site Technical Procedures, Section 3.2.2, Reference Use Procedure, in regards to completion of appropriate signoffs to verify that each segment of the procedure has been performed, prerequisite steps of WOs 04-813661-000 and 04-813662-000 were not completed, signed, and dated prior to performing work on February 24, 2005, but were signed and dated on February 26, 2005.
- The requirements of MI-68.021, Pressurizer PORV Maintenance, Section 1.3, Frequency and Conditions, which stated, “This Instruction can be performed in mode 5 or 6, provided the pressurizer has been vented per MI-68-020 and the requirements of Technical Specification 3.4.12 are adhered to.” TS 3.4.12 was not adhered to in that the PORVs were de-tensioned while still considered operable per the TS.

This violation is being treated as a NCV, consistent with Section VI of the NRC Enforcement Policy, and is identified as NCV 50-390/2005002-02, Failure to Implement Procedures which Impacted TS Requirements of the Pressurizer PORV. This issue is in the licensee's corrective action program as PER 77744.

1R13 <u>Maintenance Risk Assessments and Emergent Work Evaluation</u>

a. <u>Inspection Scope</u>

The inspectors evaluated, as appropriate for the five work activities 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; TI-124, Equipment to Plant Risk Matrix; and TI-133, Risk Assessment Methodology for Implementation of LCO 3.0.4/SR 3.0.4 Mode Restraint Requirements.

- WO 05-810042-000, Emergent work on 1B-B DG
- WO 04-022833-000, Perform 1-SI-99-301-B coincident with PORV block valve shut
- WO 05-811393-000, Performing maintenance on the standby main feedwater pump coincident with PORV block valve shut
- TI-133, Attachment 1, LCO 3.0.4 mode change risk assessment for 1A RHR pump inoperable
- WO 05-811284-00, Replace 24" ERCW line at outlet of B CCS heat exchanger

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed five operability evaluations affecting risk-significant mitigating systems, 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.

- PER 74133, ABGTS operability with the 676 ft pipe chase blowout panel actuated
- WO 05-810042-000, Repair 1B-B DG engine 2-day tank fuel oil transfer pump
- PER 75115, Fuses in breakers for containment electrical penetration protection are wrong size
- PER 76275, During performance of 1-SI-99-301-B, Section 6.4, the B train phase B did not actuate as expected
- PER 76287, DG 1A-A inoperable from placement of clearance 1-30-075A which removed power from both DG room exchanger fans

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; and TI-126, Post-Maintenance Testing Matrices. Additional documents reviewed by the inspectors are listed in the attachment to this report.

- WO 05-810655-000, Troubleshoot and repair SG 4 Ch IV lo-lo SG level trip circuit
- WO 03-019248-001, Install refurbished motor for C-A ERCW pump
- WO 04-815133-000, Implement DCN for Loop 1 main steam isolation valve (MSIV)
- WO 05-811232-000, Repair/replace 1A-A RHR pump mechanical seal as required per MI-74.001
- WO 04-813875-000, Replace 1A-A centrifugal charging pump rotating assembly
- WO 04-810497-000, Replace 1B-B DG sequence timer

b. Findings

Introduction. A Green NCV was identified by the NRC regarding the failure to perform an adequate PMT for the 1A RHR pump seal replacement.

Description. On March 17, 2005, the inspectors reviewed completed WO 05-811232-000, which replaced the mechanical seal on the 1A RHR pump and installed a refurbished motor, to verify that the PMT was adequately defined and completed. The inspectors observed that PMT template #5 description stated that the RHR pump 1-PMP-074-0010-A must be in service for shutdown cooling at 275-300 pounds per square inch gauge (psig) and any associated seal leakage from the vent openings must not exceed 10 drops per minute (as per instructions from the vendor manual). The inspectors also observed that MI-74.001, Removal, Inspection, and Replacement of Residual Heat Removal Pump, Section 7.0, Post Performance Activities, Step 7.1 [2] stated that the licensee must perform 1-SI-74-901-A, Residual Heat Removal Pump 1A-A Quarterly Performance Test. Step 7.1 [3] of MI-74.001 required the licensee to verify no visual leakage during pump operation. A review of the completed performance of 1-SI-74-901-A revealed that the pump outlet pressure was recorded as 215 psig which was below the PMT pressure specified above. The inspectors also observed that, in addition to PMT template #5 of the WO signed as complete on March 14, 2005, Part F, Operations Acceptance for Work Closure, was also signed and dated March 14, 2005, for PMTs completed and equipment ready for return to service. The inspectors informed the licensee of the finding, and the licensee changed the WO status from complete to PMT-required. The inspectors performed a walkdown of the 1A RHR pump on March 24th when its discharge pressure, as indicated on the plant computer, was approximately 350 psig. During the walkdown the inspectors identified a pencil size flow of water from the mechanical seal vent opening indicating a seal leakage much greater than allowed by the WO PMT acceptance criteria.

Analysis. The failure to perform the PMT as specified had a credible impact on reactor safety because the 1A RHR pump mechanical seal failed when run at normal system pressures. The inspectors reviewed MC 0612 and determined that the finding was more than minor due to the impact on the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences to the reactor core and the associated cornerstone attribute of human performance. The inspectors reviewed MC 0609, Appendix G, for the SDP and determined that the finding did not require quantitative assessment because it only affected one train of RHR and the steam generators (SGs) were available for heat removal. This resulted in the characterization of Green (very low safety significance). A human performance error contributed to the finding in that the PMT was signed off as complete and the WO was closed even though the PMT was performed at a lower pressure than specified. Therefore, the cause of the finding was considered to involve the cross-cutting area of human performance.

Enforcement. TS 5.7.1.1 requires that written procedures shall be implemented and maintained covering the activities in the applicable procedures recommended by RG 1.33, Revision 2, Appendix A, February 1978, of which Part 9 requires procedures for performing maintenance. SPP-6.3, Pre-/Post-Maintenance Testing, Section 3.4, Review and Approval of PMT, Step L, states that the operations shift manager/SRO designee must ensure that the PMTs are performed at the appropriate system operating conditions or plant modes. Further, MMDP-1, Step 3.10.2 A., states that the operations organization shall review the WO to verify that the PMT is successfully completed. Contrary to the above, on March 14, 2005, the PMT for WO 05-811232-000 was not performed at the specified pressure, and a seal failure was subsequently identified by the inspectors at a pressure of 350 psig. This violation is being treated as a NCV, consistent with Section VI of the NRC Enforcement Policy, and is identified as NCV 05000390/2005002-03, Failure to Perform an Adequate PMT for RHR Pump Seal. This issue is in the licensee's corrective action program as PER 78875.

1R20 Refueling and Outage Activities

a. Inspection Scope

The licensee began its Unit 1 Cycle 6 (U1C6) refueling outage RFO on February 22, 2005. From that date through the end of the report period, the inspectors observed portions of the shutdown, cooldown, refueling, maintenance activities, and startup activities to verify that the licensee maintained defense-in-depth (DID) commensurate with the outage risk plan and applicable TS. The inspectors monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including daily outage reports, to evaluate defense-in-depth commensurate with the outage safety plan and compliance with the applicable TS when taking equipment out of service.
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met.
- Licensee implementation of clearance activities to ensure equipment was appropriately configured to safely support the work or testing.

- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling.
- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after-core offload.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Reactivity controls to verify compliance with TS and that activities which could affect reactivity were reviewed for proper control within the outage risk plan.
- Refueling activities for compliance with TS, to verify proper tracking of fuel assemblies from the spent fuel pool to the core, and to verify foreign material exclusion was maintained.
- Reduced inventory and mid-loop conditions for commitments to Generic Letter 88-17 to verify that these commitments were in place, that plant configuration was in accordance with those commitments, and that distractions from unexpected conditions or emergent work did not affect operator ability to maintain the required reactor vessel level.
- Heatup and startup activities to verify that TS, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant conditions. RCS integrity was verified by reviewing RCS leakage calculations and containment integrity was verified by reviewing the status of containment penetrations and containment isolation valves.
- Containment closure activities, including a detailed containment walkdown prior to startup, to verify no evidence of leakage and that debris had not been left which could affect the performance of the containment sump or ice condenser.

b. Findings

.1 Inadequate Procedures for Containment Closure

Introduction. The inspectors identified a finding for an inadequate procedure for containment closure during loss of shutdown cooling events. The inadequate procedure could have resulted in not being able to restore containment availability due to the use of a seal which was not rated for containment pressure. This finding is an Unresolved Item (URI) pending review of licensee testing of the seal configuration which is scheduled to be completed by June 2005.

Description. On February 28, 2005, the inspectors reviewed the licensee’s containment penetration closure controls due to a short calculated time-to-boil in the reactor and upcoming fuel handling activities. Containment closure is directed by Abnormal Operating Instruction (AOI)-14, Loss of RHR Shutdown Cooling. The licensee controlled containment penetrations using TI-68.002, Containment Penetrations and Closure Controls, and MI-88.003, Opening Primary Containment Penetrations and Shield Building Penetrations for Maintenance Activities. The technical bases for using the temporary penetration are documented in the safety evaluation for MI-88.003; Drawing 47A472-11, Mechanical Penetration Seal Details; WBNAPS2-092, Containment

Response to Loss of RHR During Mid-Loop Operation; and Report on Hydrostatic Test on Dow Corning Silicon Foam in an Electrical Blockout (Test Number HT-E01-24).

The licensee opens containment penetrations X-54, X-108, X-109, X-117, and X-118 during refueling outages to run cables and hoses into containment for outage support. For this outage the penetrations were opened on February 22 and 23, the first and second days after shutdown, after the unit entered Mode 5. AOI-14, Section 3.2, RHR Pump Cavitation during Mid-Loop, Step 6, directed: "Immediately INITIATE actions to establish cntmt closure," and "ENSURE Closure initiated in accordance with TI-68.002." In addition, Section 3.9, RCS Alternate Cooling Method with the Reactor Vessel Head Off, Step 3, directed implementation of containment closure in accordance with TI-68.002. TI-68.002 directed closure of containment penetrations when directed by the Unit SRO. MI-88.003 provided the emergency closure procedures for the containment penetrations except for X-117. The emergency closure requirements for a fuel handling accident or loss of RHR cooling were to immediately close manual isolation valves in the hoses and within four hours disconnect or cut any lines and re-install the blind flanges. However, MI-88.003 stated that X-117 did not need emergency closure since only solid cables would be run through the penetration and the qualified foam seal was sufficient in the event of a loss of RHR or a fuel handling accident.

The inspectors reviewed the technical documentation and determined that it did not support the no emergency closure requirements for X-117 nor the four-hour time to re-install the blind flanges on the other four penetrations. Specifically, the inspectors determined that:

- the safety evaluation for MI-88.003 incorrectly stated that the temporary penetration seals were rated for 3 psig when Test HT-E01-24 documented the test seal failed after 50 minutes at 3 psig;
- the safety evaluation for MI-88.003 stated that containment pressure would not exceed 2 psig; however, there were periods of very short time-to-boil (approximately 14 minutes) early in the outage during reactor vessel disassembly. Under these conditions the licensee's evaluation of containment response for a loss of shutdown cooling indicated containment pressure would quickly rise to over 3 psig within 15-20 minutes after boiling starts;
- the safety evaluation for MI-88.003 stated that containment pressure would not exceed 2 psig; however, the containment response evaluation for a loss of shutdown cooling during cold mid-loop (500 hours after shutdown) incorrectly assumed that the containment equipment hatch was open. The licensee's procedure requires immediate actions to shut the containment equipment hatch on a loss of shutdown cooling. The licensee's evaluation showed containment pressure would reach approximately 1.75 psig at 5000 seconds after the loss of shutdown cooling with the equipment hatch open.

Analysis. The finding adversely affected the containment availability during loss of shutdown cooling events. The inspectors referred to MC 0612 and determined that the finding is more than minor because it affected the configuration control attribute of the Barrier Integrity Cornerstone for the reactor containment. The significance determination will be performed upon the completion of planned licensee testing of the specific seal configuration (vice the generic test HT-E01-24) used in penetration X-117 to determine if there was a loss of function.

Enforcement. TS 5.7.1.1 requires that written procedures be established, implemented, and maintained for the activities specified in RG 1.33, Revision 2, Appendix A. Item 6.h. of RG 1.33 states that implementing procedures are required for combating the loss of shutdown cooling. AOI-14, Loss of RHR Shutdown Cooling, was established and implemented to combat the loss of shutdown cooling. AOI-14 directed establishing containment closure using TI-68.002. Contrary to this, AOI-14 and TI-68.002 were not adequately established to assure that containment closure would be achieved prior to the time at which a core uncovery and fission product release could result from a loss of shutdown cooling. Specifically, there were no emergency closure actions for penetration X-117, and four hours were allowed for emergency closure of the blind flanges for penetrations X-54, X-108, X-109, and X-118. This finding does not present an immediate safety concern because these penetrations have been restored to their required condition for containment integrity. The licensee discussed their plans at the exit meeting to perform testing on a full scale mockup of penetration X-117 within 60 days of the exit. The significance determination of this finding will be performed after the inspectors review the results of the testing. Pending review of the licensee's penetration testing, this finding is identified as URI 05000390/2005002-04, Inadequate Procedures for Containment Closure.

.2 Failure to Follow Procedures Results in PORV Actuations

Introduction. A self-revealing finding was identified for two examples of failure to comply with TS 5.7.1.1 which resulted in pressurizer PORV actuations.

Description. On February 23, 2005, the inspectors identified a control room log entry which described the initiation of PER 77176 for cycling of the pressurizer PORV as a result of problems associated with charging flow control valve 1-FCV-62-93 erratic control and implementation of a design change notice (DCN) to raise control air pressure on the actuator for 1-FCV-62-93 to eliminate the erratic control. The inspectors performed a review of the reactor coolant, and charging system parameters for the period in question and determined that the Cold Over-Pressure Mitigating System as required by TS 3.4.12 was challenged by the actuation of one or both PORVs multiple times during a 2 hour period. The block valve for PORV, 1-RFV-63-340A, had been closed to reduce containment gas problems via leakage from the valve packing and as such the PORV did not relieve actual pressure during a total of 7 actuations. However, PORV, 1-RFV-63-334D, actuated a total of 4 times to reduce pressure in parallel with a group of 5 actuations by 1-RFV-63-340A. The inspectors determined that the first single actuation and a group of five/four actuations of 1-RFV-63-340A/1-RFV-63-334D were due to a failure to follow procedure regarding

General Operating Instruction (GO)-6, Unit Shutdown from Hot Standby to Cold Shutdown. To transition to solid water operations, Section 5.5, Step [1] [e] states: “Slowly RAISE charging to fill Pzr at less than 30 gpm.” Contrary to this, the licensee exceeded the 30 gpm requirement and experienced the first PORV actuation when 1-FCV-62-93 exhibited erratic operation following activities to swap from bypass to normal charging. The inspectors noted that, while the DCN had been previously implemented while the plant was on bypass charging, all of the post-maintenance testing had not yet been completed. Since 1-FCV-62-93 operation was still erratic, the licensee swapped back to bypass charging resulting in the group of five PORV actuations. The inspectors had the following observations:

- The RCS system description states that when the RCS is operated in the water solid mode, the charging flow to the RCS is set at a constant value.

- WO 04-825584-000, which implemented the DCN, contained the PMT statement: “Equipment cannot be declared operable (by release of hold order/caution order, if applicable) until modification turnover package is complete for DCN No: D-51812-A.” This statement also had a note stating that, after implementation of the DCN and completion of a test to stroke 1-FCV-62-93 while still isolated, the valve may be returned to operation with outstanding PMTs to be done when plant conditions allow. This PMT step was signed by a licensee operations shift manager and dated February 22, 2005. Subsequently the licensee returned 1-FCV-62-93 to service during the transition to solid plant operations, while the remaining PMTs, which included a stroke under high differential pressure, were not yet complete.

- The history of erratic control with 1-FCV-62-93 had resulted in a precaution and limits statement in GO-6 stating that charging flow control valve 1-FCV-62-93 may cycle with RCS pressure below 500 psig when manually attempting to control low charging flow rates. During the transition to solid plant operations, RCS pressure was less than 400 psig.

The inspectors also determined that, contrary to the original PER 77176 problem description, the last Pressurizer PORV actuation was due to RCS heatup and resultant pressure increase from the closure of the 1A RHR heat exchanger outlet valve per SOI-74.01, Residual Heat Removal, Section 8.11, Flush of A Train RHR Heat Exchanger Bypass during Shutdown Cooling. The inspectors determined that this procedure was not maintained in that the following action was contained in a procedure note, “The effect on RCS heatup/cooldown should be evaluated.” This is contrary to SPP-2.2, Administration of Site Technical Procedures, Procedure Verification Review Checklist, which prohibits action steps in a note. This action was not appropriately implemented in that the performance of Section 8.11 during solid plant operation allowed sufficient RCS heatup to result in the actuation of the Pressurizer PORV.

<u>Analysis</u>. The inspectors determined that the procedural noncompliances identified above had a credible impact on safety involving the challenge of RCS integrity by Pressurizer PORV actuations and the challenge of RCS inventory through the loss of inventory via the open Pressurizer PORVs. The inspectors reviewed MC 0612 and determined that the finding was more than minor due to the impact on the Barrier Integrity Cornerstone objective to provide reasonable assurance that the RCS physical design barrier protects the public from radionuclide releases caused by accidents or events and the associated cornerstone attributes of human performance and procedure quality. The inspectors reviewed MC 0609, Appendix G, and determined that the finding required quantitative assessment consisting of a phase 3 analysis because it affected the cold over-pressure mitigation or low temperature over-pressure system required by TS. The cause of the finding involved the cross-cutting area of human performance, in that licensee personnel failed to follow procedure guidance for controlling charging flow and for evaluating the effects of RCS heatup/cooldown.

<u>Enforcement</u>. TS 5.7.1.1 states that written procedures shall be implemented and maintained covering the activities in the applicable procedures recommended by RG 1.33, Revision 2, Appendix A, February 1978, of which Part 2.j requires a procedure for hot standby to cold shutdown, Part 3.c requires a procedure for shutdown cooling system, and Part 1.e requires administrative procedures for procedure review and approval. GO-6, Unit Shutdown from Hot Standby to Cold Shutdown, Section 5.5, Step [1] [e] states, “Slowly RAISE charging to fill Pressurizer at less than 30 gpm.” SOI-74.01, Residual Heat Removal, Section 8.11, implemented a flush of the A train RHR heat exchanger bypass during shutdown cooling and contained a note which stated, “The effect on RCS heatup/cooldown should be evaluated.” SPP-2.2, Administration of Site Technical Procedures, Procedure Verification Review Checklist, prohibits action steps in a note.

Contrary to the above, on February 22, 2005:

- GO-6, Section 5.4, step [1] [e] was not adequately implemented and net charging flow exceeded the 30 gpm requirement which resulted in PORV actuations. A single PORV actuation occurred when net charging exceeding 30 gpm due to erratic control of 1-FCV-62-93, and a subsequent group of five PORV actuations during a swap from normal to bypass charging.
- SOI-74.01, Section 8.11, was not adequately implemented in that the effect of RCS heatup/cooldown was not adequately evaluated for performance during solid plant operations. The performance of Section 8.11 during solid plant operation allowed sufficient RCS heatup to result in the actuation of the Pressurizer PORV.
- SPP-2.2 was not adequately implemented in that a note in SOI-74.01 contained an action to evaluate the effect of performing a flush of the bypass line.

This finding is considered an unresolved item (URI) pending completion of the risk significance determination. This finding is identified as URI 05000390/2005002-05, Failure to Implement and Maintain Shutdown Procedures Resulting in Pressurizer PORV Actuations. This issue is in the licensee's corrective action program as PERs 77176 and 79910.

.3 Failure to Establish Contingency Plan for Electrical Orange Risk Condition

Introduction. A Green NCV was identified by the NRC regarding the failure to establish a contingency plan for an Orange risk condition involving electrical power.

Description. On March 1, 2005, the licensee's refueling outage schedule for maintenance activities resulted in an Orange risk condition involving the Electrical Power category monitored by the licensee's DID risk assessment process. The licensee's non-quality related procedure, Standard Programs and Process (SPP) 7.2, Outage Management, Appendix C, Outage Risk Assessment Monitoring, states that a contingency plan must be in place prior to entry into an Orange plant condition (a significant reduction in DID). In addition, Outage and Site Scheduling Directive Manual (O&SSDM) 4.0, Operational Defense-In-Depth Assessment, states that written guidance/contingency plans should be made before entering a pre-planned Orange condition. The inspectors contacted the licensee's control room staff to request a copy of the contingency plan for review. While the control room staff was aware of the current Orange risk condition, the control room SRO was unaware of the need for a contingency plan, and the staff was unable to find a documented plan. The inspectors subsequently determined that contrary to SPP-7.2 and O&SSDM requirements, the refueling outage overall risk evaluation documented in the outage Safety Plan contained neither the current plant system alignment resulting in the Orange risk condition, nor the respective contingency plan. The inspectors also observed that implementation of the DID assessments by the licensee included a check for contingency plans. However, this check failed to recognize the lack of a documented contingency plan. Instead, inconsistent handwritten notes on DID forms from each shift either referred to the Safety Plan or listed one or two risk management actions regarding barricades. The inspectors subsequently identified weaknesses regarding licensee personnel disregarding barricades and signs during a high-risk evolution involving RCS mid-loop operations which was documented in PER 79069. On March 1, 2005, the completed DID form had a note that referred to the Safety Plan for a contingency plan which did not exist. Further review by the licensee determined that the outage risk plan did not document a planned Orange condition and respective contingency plan impacting the Spent Fuel Cooling category of the DID process. The licensee's failure to establish a contingency plan for a high risk condition is contrary to 10 CFR 50.65(a)(4), which requires that the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.

Analysis. The failure to assess and manage the increase in risk had a credible impact on reactor safety because the failure to have a documented contingency plan prevented a pre-planned response to any further reduction of DID while in the elevated risk configuration. The inspectors reviewed MC 0612 and determined that the finding was more than minor due to the impact on the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences to the reactor core and the associated cornerstone attribute of human performance. The inspectors reviewed MC 0609, Appendix G, and determined that the finding did not require quantitative assessment because it did not increase the likelihood of a loss of offsite power or degrade the licensee’s ability to cope with a loss of offsite power. This resulted in the characterization of very low safety significance (Green). Since licensee personnel failed to develop a contingency plan prior to entering an Orange risk condition as required by plant procedures and since control room personnel were unaware that a pre-approved contingency plan was required, the cause of the finding was determined to involve the cross-cutting area of human performance.

Enforcement. 10 CFR 50.65(a)(4) requires, in part, that, “Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.” Contrary to this, on March 1, 2005, the licensee failed to manage the increase in risk associated with a planned Orange risk condition involving the electrical power category monitored by the licensee’s DID risk assessment process in that a contingency plan as required by SPP-7.2 had not been established. This violation is being treated as an NCV, consistent with Section VI of the NRC Enforcement Policy, and is identified as NCV 05000390/2005002-06, Failure to Establish a Contingency Plan for an Orange Risk Condition Involving Electrical Power. This issue is in the licensee's corrective action program as PER 77673.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed nine surveillance tests and/or reviewed test data of 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. Additional documents reviewed are listed in the attachment.

- WO 04-818752-000, Perform 1-SI-74-901-B, RHR Pump 1B-B Quarterly Performance Test*
- WO 04-822052-000, Perform 1-SI-63-901-A, Safety Injection Pump 1A-A Quarterly Performance Test*
- WO 04-823229-000, Perform 0-SI-82-18-B, 184-Day Fast Start DG 2B-B
- WO 04-822833-000, Perform 1-SI-99-301-B, Engineered Safety Features Actuation System Slave Relay Block Test Train B
- WO 04-813842-000, Perform 1-SI-1-907, Testing and Setpoint Adjustment of Main Steam Safety Valves Using Trevitest Equipment
- WO 04-813477-000, Perform 1-SI-1-904 Full Stroke Exercising of MSIVs
- WO 04-816031-000, Perform 1-SI-61-5, 18-Month Ice Condenser Lower Inlet Doors Inspection
- WO 04-822848-000, Perform 1-SI-30-701, Containment Isolation Valve Local Leak Rate Test Purge Air**

- WO 04-824001-000, Perform 1-SI-68-32, Reactor Coolant System Water Inventory Balance***

*This procedure included inservice testing requirements.
**This procedure included testing of a containment isolation valve.
***This procedure included RCS leak detection.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed a licensee-evaluated requalification scenario on February 9, 2005, which included a formal evaluation of the event classification. The inspectors observed the 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; EIPIP-4, Site Area Emergency; EPIP-5, General Emergency; and the Radiological Emergency Plan. In addition, the inspectors verified that licensee evaluators were identifying deficiencies and properly dispositioning performance against the performance indicator criteria in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

Access Control Licensee program activities for monitoring workers and controlling access to radiologically significant areas and tasks were inspected. The inspectors evaluated procedural guidance; directly observed implementation of administrative and established physical controls; assessed worker exposures to radiation and radioactive material; and appraised radiation worker and technician knowledge of, and proficiency in, the implementation of radiation protection (RP) program activities.

During the inspection, radiological controls for ongoing refueling activities were observed and discussed. Reviewed tasks included the preparation for and the installation of nozzle dams in the SGs and radiography activities conducted in the annulus. In addition, licensee controls for selected tasks scheduled and ongoing during the refueling outage were assessed. The evaluations included, as applicable, radiation work permit (RWP) details; use and placement of dosimetry and air sampling equipment; electronic dosimeter setpoints; and monitoring and assessment of worker dose from direct radiation and airborne radioactivity source terms. Effectiveness of established controls was assessed against area radiation and contamination survey results, and occupational doses received. Physical and administrative controls and their implementation for locked high radiation areas (LHRAs) were evaluated through discussions with cognizant licensee representatives, direct field observations, and record reviews.

Occupational workers' adherence to selected RWPs and Health Physics Technician proficiency in providing job coverage were evaluated through direct observations of staff performance during job coverage and routine surveillance activities, review of selected exposure records and investigations, and interviews with cognizant licensee staff. Radiological postings and physical controls for access to designated high radiation (HRA) and LHRA locations within the Containment, Auxiliary Building, and Refuel Floor areas were evaluated during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for selected auxiliary building locations. Occupational exposures associated with direct radiation, potential radioactive material intakes, and from discrete radioactive particle or dispersed skin contamination events for calendar year (CY) 2004 were reviewed and discussed.

RP program activities were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; Updated Final Safety Analysis Report (UFSAR) details in Section 12, Radiation Protection; TS 5.7, Procedures and Programs, and TS 5.11, High Radiation Area; and approved licensee procedures. Licensee procedures, guidance documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with SPP-3.1, Corrective Action Program, Revision 7. Licensee self-assessments and PER documents related to access control that were reviewed and evaluated in detail during inspection of this program area are identified in Section 2OS1 of the report attachment.

b. Findings

Introduction. A self-identifying Green NCV was identified when radioactive material was moved which created an unposted and unbarricaded HRA.

Description. On March 22, 2005, a plant services worker received an unexpected dose rate alarm while working in upper containment. RP technicians performed surveys to identify the cause of the unexpected dose rate alarm and identified an unposted high radiation area. The licensee investigation determined that the lower containment coordinators sent high dose rate trash out of lower containment to upper containment without proper notification to the RP radiological waste technician. The lower containment coordinators did not realize that they had to specifically contact the RP technician prior to sending up the high dose trash. The trash dose rates were sufficient to create a high radiation area (greater than 100 millirem per hour at 30 centimeters) in the vicinity of the trash, resulting in an unposted high radiation area in upper containment. After discovery that trash with high radiation area dose rates may not have been properly controlled and posted, RP technicians performed surveys, identified the trash with high dose rates and placed the trash into a posted high radiation area.

Analysis. The inspectors determined that the movement of the trash which resulted in the creation of an unposted and unbarricaded HRA was a performance deficiency warranting significance evaluation. The Occupational Radiation Safety cornerstone was impacted by this issue. The inspectors reviewed the samples of minor issues in MC 0612, Power Reactor Inspection Reports, Appendix E, Examples of Minor Issues, and determined that there were no examples similar to this issue. The inspectors concluded that the finding was more than minor because the finding was associated with the Human Performance and Program & Processes attributes of the Occupational Radiation Safety Cornerstone. The finding affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material because the issue involved the occurrence of the potential for unplanned, unintended dose to other individuals working near the unposted, unbarricaded HRA. Utilizing MC 0609, Significance Determination Process (SDP), Appendix C, Occupational Radiation Safety SDP, the inspectors determined that the finding: (1) did not involve ALARA/work controls; (2) was not associated with an overexposure; and (3) based on the surveys of the radioactive trash, did not result in a substantial potential for an overexposure or compromise the licensee’s ability to assess dose. Consequently, the finding screened out as Green and was of very low safety significance. Since a human performance error contributed to the finding in that the licensee personnel were not aware of procedural requirements to notify a RP technician prior to sending the high radiation dose trash, the inspectors determined this finding involved the cross-cutting aspect of Human Performance.

Enforcement. TS 5.11.1 requires, in part, that each entryway to an HRA shall be barricaded and conspicuously posted as an HRA. Contrary to the above, on March 22, 2005, radioactive material, consisting of trash bags, was relocated from lower containment to upper containment creating an HRA which was not posted and barricaded. However, because this violation was associated with a finding of very low safety significance and because the finding was entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section VI of the NRC Enforcement Policy, and is identified as NCV 05000390/2005002-07, Radioactive Material Movement Created an Unposted and Unbarricaded High Radiation Area. This issue is in the licensee's corrective action program as PER 79240.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) Implementation of the licensee's ALARA program during the RFO-6 outage was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included installation of steam generator nozzle dams and radiography of welds in the annulus. Also, incorporation of planning, established work controls, expected dose rates and dose expenditure into the ALARA pre-job briefings, and RWPs for those activities were reviewed. The inspectors directly observed performance of the steam generator nozzle dam installation while evaluating the licensee's use of engineering controls, low-dose waiting areas, and on-the-job supervision.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Shutdown chemistry program implementation and the resultant effect on Containment and Auxiliary Building dose rate trending data were reviewed and discussed with cognizant licensee representatives.

Trends in individual and collective personnel exposures at the facility were reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. The inspectors examined the dose records of all declared pregnant workers during 2003 to 2004 to evaluate total or current gestation dose. The applicable RP procedure was reviewed to assess licensee controls for declared pregnant workers. Trends in the plant's three-year rolling average collective exposure history, outage, non-outage, and total annual doses for selected years were reviewed and discussed with licensee representatives.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 12, Sections 1-5, Radiation Protection; 10 CFR 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in Section 2OS2 of the report attachment.

Problem Identification and Resolution The inspectors reviewed CAP documents listed in Section 2OS2 of the report Attachment that are related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with SPP-3.1, Corrective Action Program, Revision 7.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization The inspectors evaluated licensee methods for processing and characterizing radioactive waste (radwaste). Inspection activities included direct observation of processing equipment for solid and liquid radwaste and evaluation of waste stream characterization data.

Solid and liquid radwaste equipment was inspected for material condition, configuration compliance with the UFSAR, and consistency with Process Control Program (PCP) requirements. The inspectors reviewed the status of non-operational or abandoned-in-place radwaste equipment. The inspectors reviewed the licensee's administrative and physical controls of non-operational or abandoned-in-place radwaste equipment to prevent unmonitored releases, determine impact to operating systems, or to contribute to unnecessary personnel exposure. Inspected equipment included liquid radwaste hold-up tanks, resin transfer piping, filters, and elements of the Mobile Demineralization System. The inspectors discussed system changes, component function, and equipment operability with licensee staff.

In addition, procedural guidance for resin transfer was evaluated and compared with current equipment configuration. Reviewed documents are listed in Section 2PS2 of the report attachment.

Licensee radionuclide characterizations for selected waste streams were reviewed and discussed with radwaste staff. For primary resin, radwaste filters, and dry active waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides and appropriate use of scaling factors. Comparison results between licensee waste stream characterization data and outside laboratory data were reviewed for the period June 2002 to October 2004. For selected shipment records, waste classification calculations were performed and the methodology used for resin waste stream mixing and concentration averaging was evaluated. The inspectors also interviewed cognizant radwaste staff and reviewed procedural guidance to evaluate the licensee's program for monitoring changing operational parameters.

Radwaste processing activities were reviewed for consistency with the licensee 's PCP, Revision 1, dated April 14, 2004, and UFSAR, Chapter 11, Amendment 4, dated April 20, 2004. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR 61.55 and guidance provided in the Branch Technical Position on Waste Classification and Waste Form, 1983.

Transportation The inspectors evaluated the licensee's activities related to the transportation of radioactive material. The evaluation included direct observation of shipment preparation activities and review of shipping-related documents.

The inspectors directly observed transportation activities including shipment packaging, surveying, blocking and bracing, vehicle placarding, vehicle checks, emergency instructions, preparation of disposal manifest, and the provision of shipping papers and special instructions to drivers. Specifically, the inspectors observed one shipment of dry active waste (DAW) and one shipment of laundry. Both shipments observed were shipped as exclusive use only and as low specific activity level II (LSA-II).

As part of the document review, the inspectors evaluated five shipping records for consistency with licensee procedures and compliance with NRC and DOT regulations. In addition, training records for two individuals currently qualified to ship radioactive material were checked for completeness and the training curriculum provided to these workers was evaluated. Documents reviewed during the inspection are listed in Section 2PS2 of the report attachment.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 170-189; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172, Subpart H.

Problem Identification and Resolution The inspectors reviewed the licensee's events reports and self-assessment related to radioactive material processing and transportation areas, to determine if problems were identified and entered into the system for resolution. Specifically, the inspectors reviewed PERs and interviewed cognizant licensee personnel to determine if problems were identified, properly characterized, prioritized, evaluated, and corrected. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP-3.1, Corrective Action Program, Revision 7. Reviewed documents are listed in Section 2PS2 of the report attachment.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

The inspectors followed up on an event that was an issue of agency-wide concern regarding an uncontrolled release of radioactive materials from the RCA.

b. Findings

No findings of significance were identified; however, a minor violation was identified for failure to control and maintain constant surveillance of licensed material that is not in storage, as prescribed by 10 CFR 20.1802. This self-revealing violation occurred when a worker, upon in-processing at another facility after working at Watts Bar Nuclear Plant (WBNP), was determined to have approximately 7,000 disintegrations per minute of Co-58 on his clothes. It was assumed that the licensed material had come from WBNP because the individual was a decon worker who had just conducted decon work at WBNP in October 2003. During those activities conducted at WBNP, the individual did receive an intake of radioactive material and had been whole body counted to confirm this. Also, a PER had been written to document this internal exposure. Because the general quantities reported were near or below the detection capabilities of the personnel monitors used to screen workers exiting the radiologically controlled areas, no performance deficiency occurred. The severity of the violation was screened using Supplement IV of the Enforcement Policy which states that a violation involving an isolated failure to secure, or maintain surveillance over, licensed material in an aggregate quantity that does not exceed 10 times the quantity specified in Appendix C to Part 20 is a minor violation. Since this issue was determined to be a violation of minor significance, it is not subject to enforcement action in accordance with Section IV of the Enforcement Policy. This minor violation is being documented because the uncontrolled release of radioactive material to the public domain is an issue of agency-wide concern.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verifications

a. Inspection Scope

The inspectors sampled licensee data submitted to the NRC for the performance indicators (PIs) listed below for the period from July 1, 2003, through December 31, 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Revision 2, were used to verify the basis for each data element.

Occupational Radiation Safety Cornerstone For the specified period, the inspectors assessed the Occupational Exposure Control Effectiveness PI data by reviewing Corrective Action Program (CAP) documents to determine whether HRA, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred. For the specified period, the inspectors evaluated data reported to the NRC, and subsequently sampled and assessed applicable CAP documents and selected Health Physics Program records. The reviewed records included personnel exposure investigation reports. Reviewed documents relative to this PI are listed in Section 4OA1 of the report attachment.

Public Radiation Safety Cornerstone The inspector reviewed the Radiological Effluent Technical Specification (RETS) / Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurences PI data. The inspectors reviewed and evaluated selected radiological liquid and gaseous effluent release data, abnormal release results, cumulative and projected doses to the public, and selected PER records for the period of July, 2003, through December, 2004. Documents reviewed are listed in Section 4OA1 of the report attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification & Resolution of Problems

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

4OA4 Cross-cutting Issues

Section 1R04 describes a finding associated with failing to follow the system operating procedure for the 1A-A DG. The inspectors identified that a human performance error resulted in the 1A-A DG being inoperable without the knowledge of licensee personnel. A SRO mis-read a table in the procedure and did not recognize that the DG was inoperable with both exhaust fans tagged out. Two other SROs and two auxiliary unit operators involved in the hold order process did not catch the error.

Section 1R12 describes a finding associated with failing to follow three procedures for controlling maintenance. The inspectors identified that multiple human performance errors resulted in de-tensioning the pressurizer PORVs while they were the TS operable vent path.

Section 1R19 describes a finding associated with failing to follow the procedure for conducting the PMT on the 1A-A RHR pump. The inspectors identified a human performance error contributed to the finding in that the PMT was signed off as complete and the WO was closed even though the PMT was performed at a lower pressure than specified.

Section 1R20.2 describes a finding associated with failing to follow procedures which resulted in lifting the pressurizer PORVs in cold overpressure mitigation mode. The inspectors identified that human performance errors contributed to the finding in that licensee personnel failed to follow procedure guidance for controlling net charging flow when entering solid plant operations and for evaluating the effects of RCS heatup/cooldown when performing a flush of a RHR heat exchanger bypass line while the RCS was solid.

Section 1R20.3 describes a finding associated with failing to manage risk as required by 10 CFR 50.65(a)(4). The inspectors identified that a human performance error contributed to the finding in that licensee personnel failed to develop a contingency plan prior to entering an Orange risk condition as required by plant procedures. In addition, control room personnel were unaware that a pre-approved contingency plan was required.

Section 2OS1 describes a finding associated with creating an unposted, unbarricaded high radiation area. The inspectors identified that a human performance error contributed to the finding in that the licensee personnel were not aware of procedural requirements to notify a RP technician prior to sending high radiation dose trash to upper containment.

4OA5 Other

.1 Temporary Instruction 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

a. Inspection Scope

The inspectors reviewed the licensee’s 60-day response to NRC Bulletin 2004-01, dated May 28, 2004. The inspectors verified that the licensee’s examinations conducted during March 6, 2005 were consistent with the licensee’s response.

The inspectors observed the bare metal visual (BMV) examination performed on a sample of the welds that fall under the scope of the bulletin. BMV examinations were observed on the following welds:

Pressurizer Safe-End Welds:
- WP-11-SE-12
- WP-11-SE-13
- WP-11-SE-14
- WP-11-SE-15

Spray Nozzle Weld:
- WP-11-SE

Safety Nozzle Welds:
- WP-13-SE
- WP-14-SE
- WP-15-SE

Safety Relief Nozzle Weld:
- WP-12-SE

Reporting Requirements are as follows:

a. For each of the examination methods used during the outage, was the examination:

   1) performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.) The inspectors verified that the examination personnel were VT-1 and VT-2 qualified in accordance with the licensee written practice, and response to Bulletin 2004-01.

   2) performed in accordance with demonstrated procedures? The inspectors reviewed the licensee's BMV examination procedure for compliance to inspection requirements, and to ensure that it contained specific instructions related to the identification, disposition, and resolution of deficiencies.

   3) able to identify, disposition, and resolve deficiencies? Through application of qualified procedures and examination personnel, the licensee was able to identify, disposition, and resolve any boric acid indications.

   4) capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01? The inspectors verified that the licensee's examination personnel were capable of identifying any leakage in pressurizer penetration nozzles or steam space piping components.

b. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)? There were no viewing obstructions, the insulation was completely removed from the identified components.

c. How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)? The examination was conducted by the direct visual examination technique.

d. How complete was the coverage (e.g., 360° around the circumference of all the nozzles)? The licensee was able to view the entire circumference, 360° around each component.

e. Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized? The examination personnel were appropriately trained and qualified to identify small boron deposits as described in the bulletin.

f. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair? There were no deficiencies identified that required repair.

g. What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)? There were no impediments for an effective examination.

h. If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations? In accordance with the licensee’s response, only a BMV examination was conducted this outage, and there were no indications identified that required further examination.

I. Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system? There were no indications of boric acid leaks from pressure-retaining components in the pressurizer system.

.2 (Closed) URI 05000390/2004005-03: Review Plant Ventilation Compensatory Sample Line Particulate Transmission Factor Calculations to Determine if ‘T’ Connection Configuration Data Were Included.

Introduction. A Green NCV of TS 5.7.1.1 was identified for failure to implement effluent monitoring quality assurance design guidance used to demonstrate representative sampling for the Auxiliary Building Ventilation Monitor (0-RE-90-101) compensatory sampler. This issue was initially identified as an Unresolved Item following an inspection in December, 2004.

Description. During field observations of 0-RE-90-101 sample line configurations, the inspectors questioned the adequacy of the main sample line to provide representative particulate samples to the compensatory sampling skid equipment. Specifically, the inspectors noted a ‘T’ connection for taking suction from the main sample line to supply the compensatory sample line when the primary skid is declared out-of-service (OOS). During the inspection, 0-RE-90-101 was declared OOS and the temporary skid was installed to obtain compensatory samples for particulate, iodine, and noble gas effluent monitoring as required by Offsite Dose Calculation Manual (ODCM) Table 1.1-2. The inspectors noted that the connection was not in accordance with acceptable industry practices regarding sample lines as outlined in American National Standards Institute (ANSI) N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities, nor in accordance with licensee sample line design documents. The inspectors noted that Watts Bar Design Criteria Document 40-24, Radiation Monitoring, Revision 14, specifies that for required sampling delivery lines, 90-degree bends which can affect the collection of particulates should be avoided, or an evaluation of deposition (loss/transmission) of iodine and particulates in these lines should be made. Licensee representatives stated that the ‘T’ connection originally was designed as a connection point for taking noble gas grab samples only, which are unaffected by elbows and bends in sample lines, but later was modified to collect particulate and iodine samples as well. The inspectors noted that a transmission factor is applied to effluent release permits to account for sample line losses; however, at the time of the inspection, the licensee was unable to provide documentation to show that the observed ‘T’ connection was properly evaluated as part of the re-analysis of sample line loss when the modification was made to allow particulate and iodine sampling.

On January 14, 2005, the licensee provided the inspectors with calculation WBNTSR-108, Evaluation of Sample Line Plateout with Reduced Flow, that showed how the ‘T’ connection was included in the compensatory sample skid re-analysis. The calculation was based on a flow of 2 cubic feet per minute (cfm) in the main sample line and assumed that the ‘t’ connection was a 90E bend with all of the flow directed into the compensatory sample line. Normally, the 0-RE-90-101 sample pump provides a flow of 10 cfm in the main sample line; however, calculation WBNTSR-108 assumed that the main sample pump is turned off and all sample line flow is due to the 2 cfm capacity sample pump attached to the compensatory sampling skid. Based on discussions with licensee representatives, the inspectors noted that during times when 0-RE-90-101 is declared OOS, the system is usually configured so that both sample pumps are left running. This results in 10 cfm continuing to flow through the main sample line to the 0-RE-90-101 skid with a suction of 2 cfm diverted through the ‘T’ connection to the compensatory sampling skid. Based on review of the submitted calculation and discussions with cognizant licensee representatives, the inspectors determined that leaving both pumps running during periods of compensatory sample collection represents a condition that had not been evaluated for sample line loss.

<u>Analysis</u>. The inspectors noted that a failure to adequately evaluate the effect of the observed 'T' connection on the sample line transmission factor is a performance deficiency because the licensee is expected to establish, implement, and maintain quality assurance activities. The ODCM requirements for effluent measurements and the missed calculations were reasonably within the licensee's ability to foresee and correct. The failure to adequately evaluate sample line loss is considered greater than minor because it is associated with the program and process attribute of the Public Radiation Safety Cornerstone and affects the cornerstone objective to assure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The failure to conduct appropriate evaluations to assure representative sample collection from the U1 plant ventilation exhaust streams using the compensatory sampling configuration could result in inaccurate measurement of airborne particulate radionuclides in effluent samples and inaccurate dose estimates to members of the public.

A follow-up licensee calculation for the observed Auxiliary Building compensatory sampling configuration, WBNAPS3-122, Auxiliary Building Radiation Monitor 0-RE-90-101, was provided to the inspectors on February 16, 2005. The calculation demonstrated that the unanalyzed sample line configuration had a negligible impact on particulate effluent sample accuracy and that the transmission factors currently in use are appropriate for the compensatory sample system as it is currently configured. This finding was evaluated using the Public Radiation Safety SDP and is of very low safety significance (Green) because the licensee's ability to assess offsite dose was not impaired and doses to the public were below 10 CFR 50, Appendix I, and 10 CFR 20.1301 limits.

<u>Enforcement</u>. TS 5.7.1.1©) requires the licensee to establish, implement, and maintain procedures for QA for effluent monitoring. Watts Bar Design Criteria Document 40-24, Radiation Monitoring, Revision 14, specifies that for required sampling delivery lines, sharp 90-degree bends which can affect the collection of particulates should be avoided, or an evaluation of deposition (loss/transmission) of iodine and particulates in these lines should be made. In addition, Section 1.0 of the ODCM specifies effluent release methodologies to be developed using the guidance in RG 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants. RG 1.21, Section C.6, states that the guidance for sampling from ducts and stacks contained in ANSI N13.1-1969 are generally acceptable and provide adequate bases for the design and conduct of monitoring programs for airborne effluents. Appendix B of ANSI N13.1-1969 states that for cases where sampling delivery lines are required, an evaluation should be made of deposition (loss/transmission) in these lines.

Contrary to TS 5.7.1.1, the licensee failed to implement design procedures for effluent monitoring quality assurance in that an analysis of sample line particulate radionuclides loss/transmission had not been made for the observed compensatory sampling configuration. Because the failure to comply with TS 5.7.1.1 is of very low safety significance and has been entered into the licensee's corrective action program (PER

04-01757), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 050000390/2005002-08, Failure to Evaluate Effluent Sample Line Losses for the Auxiliary Building Ventilation Monitor (0-RE-90-101) Compensatory Sampling Skid.

4OA6 Meetings, including Exit

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. William Lagergren and other members of licensee management at the conclusion of the inspection on April 6, 2005. 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 18, 2005, the NRC's Chief of Reactor Project's Branch 6 and the Senior Resident Inspector assigned to the Watts Bar Nuclear 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 through December 2004. The major topics addressed were: the NRC's assessment program, the results of the Watts Bar assessment, and NRC inspection plans. Attendees included Watts Bar site management and members of site staff. No members of the public attended.

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 ML051100154. ADAMS is accessible from the NRC Web site at http://www/reading-rm/pdr.html (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

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

- TS 3.7.12 requires that two Auxiliary Building Gas Treatment System (ABGTS) trains be operable for movement of irradiated fuel assemblies in the fuel handling area. If both trains of ABGTS are inoperable, the movement of irradiated fuel assemblies will be immediately suspended. Contrary to this, on March 11th, fuel handling was conducted with both trains of ABGTS inoperable. This item is in the corrective action program as PER 78414. This finding is of very low safety significance as listed in MC 0609, Appendix H, Containment Integrity Significance Determination Process, Table 4.1, because fuel handling accidents in the spent fuel pool are not important to LERF because of the small fission product inventory in an individual fuel assembly and the scrubbing effect of the water.

SUPPLEMENTAL INFORMATION
PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. DeRoche, Site Nuclear Assurance Manager
R. Evans, Acting Training Manager
A. Hinson, Maintenance and Modifications Manager
W. Justice, Engineering and Site Support Manager
W. Lagergren, Site Vice President
G. Laughlin, Plant Manager
D. Nelson, Business and Work Performance Manager
R. O'Rear, Operations Superintendent
P. Pace, Licensing and Industry Affairs Manager
T. Wallace, Operations Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000390/2005002-04	URI	Inadequate Procedures for Containment Closure (Section 1R20.1)
05000390/2005002-05	URI	Failure to Implement and Maintain Procedures Resulting in Pressurizer PORV Actuations (Section 1R20.2)

Opened and Closed

05000390/2005002-01	NCV	DG Fans Removed From Service and Tech Spec SR 3.8.1.1 Not Performed (Section 1R04)
05000390/2005002-02	NCV	Failure to Implement Procedures which Impacted TS Requirements of the Pressurizer PORV (Section 1R12)
05000390/2005002-03	NCV	Failure to Perform an Adequate PMT for RHR Pump Seal (Section 1R19)
05000390/2005002-06	NCV	Failure to Establish a Contingency Plan for an Orange Risk Condition Involving Electrical Power (Section 1R20.3)
05000390/2005002-07	NCV	Radioactive Material Movement Created an Unposted and Unbarricaded High Radiation Area (Section 2OS1)
05000390/2005002-08	NCV	Failure to Evaluate Effluent Sample Line Losses for the Auxiliary Building Ventilation Monitor (0-RE-90-101) Compensatory Sampling Skid (Section 4OA5.2)

Closed

05000390/2004005-03	URI	Review Plant Ventilation Compensatory Sample Line Particulate Transmission Factor Calculations to Determine if 'T' Connection Configuration Data Were Included (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R04
- SOI-82.01, Diesel Generator (DG) 1A-A
- SOI-74.01, Residual Heat Removal System
- SOI-78.01, Spent Fuel Pool Cooling and Cleaning System
- SOI-70.01, Component Cooling Water (CCS)

Section 1R07
- MI-70.002, Component Cooling Heat Exchanger Maintenance and Testing

Section 1R08
Corrective Action Documents Prompted by NRC Inspection:
- PER 78273, NRC identified evidence of boron leakage

Nondestructive Examination Procedures:
- Procedure N-UT-80, Ultrasonic Examination of Westinghouse Reactor Coolant Pump Shafts (PWR), Revision 1
- Procedure N-VT-1, Visual Examination Procedure for ASME Section XI Preservice And Inservice, Revision 37
- Procedure N-UT-64, Generic Procedure for Ultrasonic Testing of Austenitic Pipe Welds, Revision 7
- Procedure N-VT-8, Visual Examination of PWR Vessel Interiors and Core Support Structures, Revision 8
- Procedure N-VT-19, Visual Inspection of Alloy 600/82/182 Pressure Boundary Components, Revision 1
- Procedure N-VT-17, Visual Examination for Leakage of PWR Reactor Head Penetrations, Revision 4
- TI-68.017, Reactor Building Post Shutdown Walkdown, Revision 7
- Procedure IswT-PDI-AUT2, Automated Inside Surface Ultrasonic Flaw Evaluation and Sizing, Revision 0
- Procedure IswT-PDI-AUT1, Automated Inside Surface Ultrasonic Examination of Ferritic Vessel Wall Greater than 4.0 Inches in Thickness, Revision 0
- Procedure IswT-PDI-AUT4, Automated Inside Surface Ultrasonic Nozzle-To-Shell Welds Using Phased Array, Revision 0
- Procedure IswT-NDE2, Ultrasonic Linearity Measurements, Revision 0
- Procedure IswT-PDI-AUT11, Automated Inside Surface Ultrasonic Examination of Piping Welds Using Phased Array, Revision 0

Other Documents:
- Technical Requirement Instruction 1-TRI-0-10, ASME Section XI ISI/NDE Program, Revision 12
- TVA Business Practice (BP)-257, Integrated Material Issues Management Program, Revision 1
- Tennessee Valley Authority’s Watts Bar Nuclear Plant Unit 1 ASME Section XI Inservice Inspection Summary Report 5th Refueling Outage Cycle

Steam Generator:

- WBNP Unit 1 Cycle 6, Degradation Assessment, Rev. 1
- WBNP Unit 1 Cycle 6, Steam Generator Tubing Examination Scan Plan, Rev. 0
- WBNP Unit 1 Cycle 5, Steam Generator Final Operational Assessment, March 1, 2004
- Steam Generator Eddy Current Examination Guideline, Rev. 7
- TVA-400-001, Multifrequency Eddy Current Examination of Non-Ferromagnetic Steam Generator Tubing, Rev. 10
- WBN-006, Standard In Situ Pressure Test Using the Computerized Data Acquisition System, Rev. 04

Section 1R12

- MI-68.020, Purging of the Pressurizer, Pressurizer Relief Tank, and Reactor Head
- PER 77699 Reportability Evaluation
- Drawing S-1714-00, PORV Superbolt

Section 1R19

- SPP-9.1, Part B, ASME Section XI system Pressure Test Program
- TI-100.010, System Pressure Testing
- TI-100.009, ASME Section XI System Pressure Test Program Basis Document
- TI-100.001, Inservice Testing of Pumps
- PER 02-002407-000, Functional Evaluation of RHR Pump Seal Leakage
- PEG Package No. CWA-WBN-2005-014
- Vendor Manual, WBN-VTD-1075-0430
- N-VT-4, System Pressure Test Visual Examination Procedure
- IT-126, Post-Maintenance Testing Matrices
- SPP-6.3, Post Maintenance Testing
- MI-62.001, Centrifugal Charging Pump
- 1-SI-62-901-A, Centrifugal Charging Pump 1A-A Quarterly Performance Test
- MI-67.003, ERCW Motor Removal, Disassembly, and Reassembly

Section 1R20

- Tagout 1-TO-2005-0006, Replace PORV with a new or rebuilt valve
- SSD-1-LPT-68-1B, Scaling and setpoint document for 1-TM-68-1F, COMS function generator
- N3-68-4001, RCS System Description including RCS Pressure and Temperature Limits report
- SOI-62.01, Section 8.17, Bypassing 1-FCV-62-93, CVCS Charging Header Flow, for Local Control
- General Operating Instruction (GO) -10, Reactor Coolant System Drain and Fill Operations
- GO-6, Unit Shutdown from Hot Standby to Cold Shutdown
- GO-5, Unit Shutdown from 30% Reactor Power to Hot Standby

Section 1R22

- PER 75906, NRC identified issue with incorrect test instrumentation calibration due dates documented.
- PER 76101, NRC identified issue on preconditioning regarding the addition of oil to pump bearings just prior to performance of the quarterly surveillance test.
- PER 75885, NRC identified issue on inconsistent documentation regarding implementation of ASME XI ISI requirements.
- PER 76957, NRC identified issue for the failure to perform a vendor procedure review for critical steps.

Section 2OS1

Procedures, Instructions, Guidance Documents, and Operating Manuals:

- Standard Programs and Processes (SPP) - 3.1, Corrective Action Program (CAP), Revision (Rev.) 7
- SPP-5.1, Radiological Controls, Rev. 5
- Tennessee Valley Authority (TVA), TVA Nuclear (TVAN) Standard Department Procedure (SDP), Radiation Control Departmental Procedure (RCDP) - 3, Administration of Radiation Work Permits (RWP), Rev. 3
- TVA, TVAN, SDP RCDP - 5, TLD Operations, Rev. 0
- TVA, TVAN SDP, RCDP - 7, Bioassay and Internal Dose Program, Rev. 0
- TVA, TVAN Standard Programs and Processes (SPP), SSP - 3.1, Corrective Action Program, Rev. 7
- TVA, TVAN SPP, SPP - 5.0, Radiological and Chemistry Control, Rev. 1
- TVA, TVAN SPP, SPP - 5.1, Radiological Controls, Rev.
- TVA, Watts Bar Nuclear Plant (WBNP), Radiation Control Instruction (RCI)-100, Control of Radiological Work, Rev. 24
- TVA, WBNP, RCI-101, Radiation, Contamination, and Airborne Surveys, Rev. 20
- TVA, WBNP, RCI-102, Contamination and Hot Particle Control, Rev. 7
- TVA, WBNP, RCI-103, Radioactive Material Control, Rev. 22
- TVA, WBNP, RCI-111, Special Exposure Monitoring, Rev. 9
- TVA, WBNP, RCI-119, Use and Control of Portable HEPA Ventilation Units and HEPA Vacuum Cleaners, Rev. 9
- TVA, WBNP, RCI-129, Radiographic Operations, Rev. 3
- TVA, WBNP, RCI-144, Field Implementation of Remote Monitoring, Rev. 0
- TVA, WBNP, System Operating Instruction (SOI), SOI-74.01, Residual Heat Removal System, Rev. 45
- TVA, WBNP, Technical Instruction, TI-7.005, Storage of Material in the Spent Fuel Pool, Cask Pit & New Fuel Vault, Rev. 19

Records and Data Reviewed:

- Active Hot Spot Database Report, 03/02/05
- Air Sample Results for Survey Nos. 110305201 - 110305214 (conducted 03/10 - 03/11/05), 220205012, 220205013, 220205015, and 220205018 (conducted on 02/22/05), and 270205202 - 270205213 (conducted 02/26 - 02/27/05)
- Airborne Radiological Survey Sample Number Log for Radcon Lab (02/22/05), and Upper Containment (02/27/05, 03/10 - 03/11/05)
- HRA/LHRA/VHRA Area Access Control/Posting Inspection Log
- Industrial Radiography Operating and Emergency Procedures, Procedure No. IEP-400, Rev. 2, Daily Radiological Survey Report, 03/02/05

- RWP Number (No.) 05006070, U1C6 - Steam Generator Jump for Installation/Removal of Nozzle Dams
- RWP No. 050007011, U1C6 - General Plant Access for Nuclear Security, Engineering, Job Planners, Plant Management and NRC, Performing Plant Tours, Inspections, Surveillances and Walkdowns in High Radiation Areas
- RWP No. 05007030, U1C6 - NSS Filter Changeout and Associated Work During the RFO
- RWP No. 05007072, U1C6 - Radiography of Various Plant System Components Located Outside of U1 Containment
- RWP No. 05007081, U1C6 - Snubber Inspection, Testing, and Repair in High Radiation Areas in the Aux Bldg and Annulus to Support the RFO
- RWP No. 05007111, U1C6 - Insulation Removal/Replacement in High Radiation Areas in the Aux Bldg and Annulus to Support the RFO
- RWP No. 05007301, U1C6 - MSA Performing Corrective and Preventive Maintenance in High Rad Areas in the Aux Bldg and Annulus
- RWP No. 05007305, U1C6 - Modifications Group Performing Corrective and Preventive Maintenance in High Rad Areas in the Aux Bldg and Annulus to Support the RFO
- RWP No. 05007311, U1C6 - RHR Pump 1A-A Seal Replacement
- RWP No. 05007501, U1C6 - Radcon Surveillances and Job Coverage Activities in Support of Refueling Outage
- RWP No. 05008072, U1C6 - Radiography of Various Plant System Components Located Inside U1 Containment
- RWP No. 05009155, U1C6 - Retrieve Boron Associated With Canopy Seal Weld Leakage. Includes Camera Installation, Boron Retrieval, Radcon Surveys, and Other Activities Allowed by the RCSS
- RWP No. 05009165, U1C6 - Recovery Efforts in Upper Containment Associated with Spray Down of Upper Containmnet. Includes Decon of Head, Megger Check of CRDM Coil Stacks and RPI Coils, Area Cleanup and Other Work Required Due to Spray Down
- RWP No. 05009501, U1C6 - Upper Containment and Refuel Floor Radcon Surveillances and Job Coverage Activities
- RWP No. 05009610, U1C6 - Cavity and Equipment Pit Decon Including All Support Work Such as Wet Masslin, Hydrolazing, and Hand Scrubbing in an Airborne Radioactivity Area in U-1 Upper Containment
- TVA, WBNP, Batch Gaseous Effluent Permit No. 2005056.080.015G, 03/11/05
- Watts Bar Radiological Survey (WBRS) # 022205-60 - U1 1A-A RHR Pump Room 676', 02/22/05
- WBRS # 022305-61 - CVCS Hold-Up Tank Room-A 692', 02/23/05
- WBRS # 022705-5 - Unit 1 Containment Bldg. U/C G/A 802', 02/27/05
- WBRS # 022705-9 - Unit 1 Containment Bldg. U/C G/A 802', 02/27/05
- WBRS # 022705-37 - Unit 1 Containment Bldg. U/C G/A 802', 02/27/05
- WBRS # 022705-39 - Unit 1 Containment Bldg. Reactor Head & Vessel, 02/27/05
- WBRS # 022705-43 - Unit 1 Containment Bldg. U/C G/A 802', 02/27/05
- WBRS # 022705-47 - Unit 1 Containment Bldg. U/C G/A 802', 02/27/05
- WBRS # 022705-48 - Unit 1 Containment Bldg. Rx. Cavity Elev. 713, 02/27/05
- WBRS # 022705-51 - U1 1A-A Containment Spray Pump Room 676', 02/27/05
- WBRS # 022805-18 - U1 1B-B RHR Pump Room 676', 02/28/05
- WBRS # 030105-11 - U1 Charging Pump Room 1B-B 692', 03/1/05

Corrective Action Program (CAP) Documents:

- Focused Self Assessment Report (FSAR), Assessment No. WBN-OPS-04-003, Radiation Worker Knowledge and Performance
- FSAR Assessment No. WBN-RP-04-001, Personnel Contaminations from WBN RFO5
- Nuclear Assurrance (NA) - TVAN-Wide - Audit Report No. SSA0302 - Radiological Protection and Control Audit, dated December 31, 2003
- Problem Evaluation Report (PER) 77282, Six workers were exposed to unanticipated radioactive airborne I-131 while investigating a leak in the “A” RHR Pump Room
- PER 77568, Ten workers were exposed to unanticipated radioactive airborne when the Cold Leg Accumulator was inadvertently discharged into the RCS
- PER 77984. Operations employee entered Steam Generator Eddy Current laydown/platform areas without RADCON approval
- PER 78380, A Westinghouse individual working on the steam generator platforms received an uptake of 78.3 nCi
- PER 78391, On nightshift ending 03/11/2005, containment purge was taken out of service without Radcon/Chemistry notification which contributed to the increase of iodine airborne radioactivity
- PER 78446, Trend PER to evaluate the U1C6 dosimetry investigations due to lost electronic dosimeters
- Self-Assessment Report, Assessment No. WBN-RP-04-002, Electronic Dosimeter Use
- Self-Assessment Report, Assessment No. WBN-RP-04-003, Radworker Practices

Section 2OS2

Procedures, Instructions, Guidance Documents, and Operating Manuals:

- ALARA Pre-Planning Report (APR) 05-0026, 1A-A RHR Pump Seal Replacement and Support Work During RFO6
- APR 05-023, Work Associated with Reactor Pressure Vessel Head Assembly and Disassembly
- APR 05011, U1C6 Refueling Outage ISI Work
- APR 05019, Setup Preparation and Restoration of Work Area for Steam Generator Maintenance Activities to Support U1C6 Refueling Outage
- APR 05020, Installation and Removal of Nozzle Dams
- APR 05021, Eddy Current Testing U1C6 Refueling Outage
- RCI-128, ALARA Program Implementation, Rev. 5
- TVA, TVAN RCTP - 105, Personnel Inprocessing and Dosimetry Administrative Processes, Rev. 0
- TVA, TVAN SPP, SPP - 5.2, ALARA Program, Rev. 2
- TVA, WBNP, Chemistry Manual, Chapter 5.09, Shutdown Primary Control Chemistry, Rev. 13

Records and Data:

- Dose Records of all declared pregnant workers (4) during the period 01/01/2003 to 03/15/2005
- RWP No. 05006030, U1C6 - Steam Generator Primary Manway and Insert Remove/Install Including Insert Cleaning/inspection and Stud Hole and Bolt Cleaning
- RWP No. 05006040, U1C6 - Eddy Current Test, Tube Plugging, Tube Sleeving and In-Situ Test
- RWP No. 05006070, U1C6 - Steam Generator Full Jump for Installation/Removal of Nozzle Dams
- RWP No. 05006200, U1C6 - Steam Generator Scaffolding Installation/inspection/Removal
- RWP No. 05007311, U1C6 - RHR Pump 1A-A Seal Replacement
- RWP No. 05007501, U1C6 - Radcon Surveillances and Job Coverage Activities in Support of Refueling Outage
- RWP No. 05009120, U1C6 - Electrical Valve Maintenance Conducted in U1 Upper Containment to Support Refueling Outage
- RWP No. 05009150, U1C6 - Disassembly/Reassembly of Reactor Head, Remove and Install Seismic Restraints, Remove/Reinstall Head Insulation, Disconnect /Reconnect Head Vent Piping, Instrument Tubing, Spool Pieces and RVLIS, Remove/Reinstall CRDM Ventilation Duct, Disconnect/Reconnect Electrical Cables, De-Tension/Re-Tension RPV Head Bolts, Lift and Store RPV Head, Install/Remove Upper Internals,(exlcude CRM Drive Shaft) and Install/Remove Head Shielding
- RWP No. 05009151, U1C6 - Disassemble/Reassemble of Reactor Head: To Include Remove/Install Missile Shield, Prepare RX Cavity Including Install/Remove Cavity Drain Covers, and NIS Covers, Remove/Install Fuel Transfer Tube Blind Flange, Install/Remove RPV Guide Pins, Stud Hole Plugs and Head Lift Rig, Install/Test/Remove Rx Cavity Seal Ring, Lift and Store RPV Head, Remove/Reinstall Upper Internals, and Install/Remove Head Shielding.
- RWP No. 05009600, U1C6 - Plant Services Activities in U1 Upper Containment to Support Refueling Outage. Includes Firewatch, Access Control, General Labor Support, Laundry Pickup and Decon Activities.
- Supervisory Brief, Unit 1 Cycle 6 Outage Briefing, 03/05/05
- U1C6 ALARA Summaries dated 02/28/05, 03/14/05 and 03/17/05
- WBN U1C6 RCS Sum of Hard Gamma Activity (first 160 hours of outage)
- WBNP - Quarterly ALARA Committee Meeting Minutes dated 12/03/03, 02/25/04, 03/25/04, 06/15/04, 07/30/04, 08/13/04, 09/07/04, 11/05/04, 01/28/05 and 01/31/05
- WBNP RFO-5 ALARA Outage Report
- WBNP Site Dose Reduction Strategy
- WBNP TEDE ALARA Worksheets for RWP No. 05007311 dated 02/22/05, 02/24/05, and 03/03/05, and RWP No. 05009502 dated 03/03/05

CAP Documents:

- PER 77867, Nozzle Dam installation dose goal and schedule goal was [*sic*] exceeded

Section 2PS2

Procedures, Guidance Documents and Manuals:

- TVA, Engineering and Support (E&S), QA Record, WBNP, No. N3-77B-4001, Solid Waste Disposal
- TVA, E&S, QA Record, WBNP, No. N3-77C-4001, Liquid Radwaste Processing System
- Radioactive Materials Shipment Manual, TVA, Rev. 37 and 37A (Volumes I and II)
- TVA, TVAN Common Technical Procedure (CTP), RWTP - 100, Radioactive Material/Waste Shipments, Rev. 2
- TVA, TVAN CTP, RWTP - 101, 10 CFR Waste Characterization, Rev. 0
- TVA, TVAN SPP, SPP - 3.1, Corrective Action Program, Rev. 7
- TVA, TVAN SPP, SPP - 5.7, Radwaste Management, Rev. 1
- TVA, TVAN SPP, SPP - 9.6, Appendix C, Rev. 7
- TVA, WBNP, Offsite Dose Instruction, 0-ODI-90-1, Liquid Radwaste Tank Release, Rev. 24
- TVA, WBNP, Process Control Program, Rev. 1
- TVA, WBNP, RCI-105, Shipping Radioactive Materials, Rev. 11
- TVA, WBNP, RCI-125, Operation of the Mobile Demineralizers, Rev. 8
- TVA, WBNP, RCI-131, Radioactive Waste Management and Minimization, Rev. 2
- TVA, WBNP, RCI-136, Loading Radioactive Material for Shipment, Rev. 2
- TVA, WBNP, SOI - 77.03, Spent Resin Handling, Rev. 29
- TVA, WBNP, TI-209, Pages 4-5 and 17, Rev. 1

Records and Data:

- 10 CFR 61 Analysis Reports ERM&I Sample Nos. 326385-2, 424793-8, and 520302-1
- CNS Demineralizer CNS Cation Resin Report dated 06/18/04
- CNS Demineralizer CNS Charcoal Report dated 01/22/04
- CNS Demineralizer CNS Mixed Bed Resin Report dated 02/26/04
- CNS Resin Report (CNS CAT/MB Mix) dated 02/26/04
- CVCS Resin Reports dated 10/07/04 and 12/21/04
- Designation of Individuals Responsible for the Safe Packaging, Transfer and Transport of Radioactive Materials, letter dated 11/17/04
- DAW Report dated 02/10/04
- Radioactive Material Manifests: WBN 011, 04-003, 04-030, 05-001, 05-024, and 05-030
- RCS Letdown Spent Cartridge Filters Report dated 07/27/04
- SWIF Filter Clippings Reports dated 03/03/04 and 02/07/05
- Waste Stream Reports for CNS Cat/MIB Mix, CNS Cation Resin, CNS Charcoal, CNS Mixed Bed Resin, CVCS Resin, DAW and Spent Cartridge Filters

CAP Documents:

- Self-Assessment Report, Assessment No. WBN-RP-04-005, Radioactive Waste Control, Liquid Effluents

Section 2PS3

Procedures, Guidance Documents and Manuals:

- TVA, TVAN Standard Programs and Processes, SSP - 3.1, Corrective Action Program, Rev. 7
- TVA, WBNP, RCI-134

CAP Documents:

- PER Work Order 03-017797-000

Section 4OA1

Procedures, Guidance Documents and Manuals:

- Desktop Guide for Chemistry Reporting
- TVA, TVAN Standard Programs and Processes, SSP - 3.1, Corrective Action Program, Rev. 7

Records and Data:

- Monthly Liquid and Gaseous Dose Reports; July 2003 through December, 2004
- Search of PER Archives for Radioactive Contamination, Personnel Contamination, Worker Contamination, Dosimetry Investigation Report, and Overexposure