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

January 30, 2008

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

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2007005 AND 05000328/2007005

Dear Mr. Campbell:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on January 9, 2008, with Mr. Timothy Cleary 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 report documents one NRC-identified finding of very low safety significance (Green), which was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this 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 Sequoyah Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Rebecca L. Nease, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-327, 50-328
License Nos.: DPR-77, DPR-79

Enclosure: Inspection Report 05000327/2007005 and 05000328/2007005
w/Attachment: Supplemental Information

cc: w/encl: (See page 3)

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

James R. Douet
Vice President, Nuclear Support
Tennessee Valley Authority
Electronic Mail Distribution

H. Rick Rogers, Vice President
Nuclear Engineering & Technical Services
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Timothy P. Cleary
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy-Daisy, TN 37384-2000

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

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

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

James D. Smith, Manager
Licensing and Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Christopher R. Church
Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

Lawrence E. Nanney, Director
TN Dept. of Environment & Conservation
Division of Radiological Health
Electronic Mail Distribution

County Mayor
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Larry E. Nicholson, General Manager
Performance Improvement
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Ann Harris
341 Swing Loop
Rockwood, TN 37854

James H. Bassham, Director
Tennessee Emergency Management
Agency
Electronic Mail Distribution

Letter to William R. Campbell from Rebecca L. Nease dated January 30, 2008

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2007005 AND 05000328/2007005

Distribution w/encl:
Bob Pascarelli, NRR
B. Moroney, NRR
C. Evans, RII
L. Slack, RII
OE Mail
RIDSNRRDIRS
PUBLIC

NRC Resident Inspector
U.S. Nuclear Regulatory Commission
2600 Igou Ferry
Soddy-Daisy, TN 37379

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report No:	05000327/2007005 and 05000328/2007005
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	October 1, 2007 - December 31, 2007
Inspectors:	S. Freeman, Senior Resident Inspector M. Speck, Resident Inspector W. Loo, Senior Health Physicist J. Griffis, Health Physicist A. Nielsen Health Physicist B. Miller, Reactor Inspector A. Rogers, Reactor Inspector R. Taylor, Reactor Inspector
Approved by:	R. Nease, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2007-005, IR 05000328/2007-005; 10/01/2007 - 12/31/2007; Sequoyah Nuclear Plant, Units 1 and 2; Refueling and Outage Activities.

The report covered a three-month period of inspection by resident and regional inspectors and health physicists. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, for failure to implement licensee Procedure SPP-6.5, Foreign Material Control. During a review of the core verification video following refueling and reactor vessel head installation, the inspectors identified debris within the Reactor Coolant System (RCS), not previously identified by the licensee. The licensee took immediate action to enter the problem into their corrective action program and evaluate whether the reactor coolant system could safely operate with the material left behind.

  The finding was more than minor because the material could have been removed had it been properly identified and because an evaluation was required to justify leaving it after the reactor head was installed. The finding was of very low safety significance because it affected only the fuel barrier and not the RCS barrier. The finding had no cross-cutting aspects. (Section 1R20.1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the period at 79% rated thermal power (RTP) and operated there until October 4, 2007 when it was shutdown for a refueling outage. Unit 1 achieved criticality on November 15, 2007 and reached Mode 1 on November 16, 2007. The unit was shut down on November 17, 2007 due to rod control system power supply problems. Following repairs, Unit 1 returned to critical on November 17, 2007 and reached 100% RTP on November 21, 2007 where it remained for the duration of the reporting period.

Unit 2 operated at or near 100% RTP until November 1, 2007 when power was reduced to 58% RTP to correct problems on the Number 3 Heater Drain Tank level control system. Unit 2 returned to 100% RTP on November 2, 2007 and remained there for the duration of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed design features and licensee preparations for protecting both Unit 1 and Unit 2 refueling water storage tanks (RWSTs) and the essential raw cooling water (ERCW) intake structure from extreme cold and freezing conditions. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TS), reviewed and observed implementation of licensee freeze protection procedures, and walked down portions of the systems to assess deficiencies and the system readiness for extreme cold weather and discussed prioritization and status of correcting deficiencies with licensee personnel. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns. The inspectors performed a partial walkdown of the Unit 2 Emergency Core Cooling system Train A during Train B maintenance to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control system components and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted a tour of the nine areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with the licensee's administrative procedure; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan.

- Auxiliary Building Elevation 749 (Pressurizer Heater Transformer Rooms and CRDM Equipment Rooms)
- Control Building Elevation 706 (Cable Spreading Room)
- Emergency Diesel Generator Building
- Control Building Elevation 669 (Mechanical Equipment Room, 250-VDC Battery and Battery Board Rooms)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Control Building Elevation 732 (Mechanical Equipment Room and Relay Room)
- Auxiliary Building Elevation 690 (Corridor)
- Auxiliary Building Elevation 714 (Corridor)
- Essential Raw Cooling Water Building

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the turbine building internal flood protection design to determine the strategy for mitigating a flood caused by a break in a large circulating water pipe and the potential flood propagation to the auxiliary and control buildings. The inspectors reviewed the Sequoyah Probabilistic Safety Assessment Individual Plant Examination to verify that assumptions and mitigating elements of various flood scenarios were addressed by plant procedures and operator actions. The inspectors reviewed the most recent performances of the preventative maintenance work orders on the various turbine building sumps to verify that flooding in the turbine building would be detected. The inspectors also walked down selected areas of the turbine building to view flood protection doors and level detection devices to assess material condition and general condition. Documents reviewed are listed in the Attachment.

b. Findings

No Findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed performance and reviewed the results of licensee Procedure 2-PI-SFT-070-001.0, Performance Testing of Component Cooling Heat Exchangers 2A1, 2A2, Revision 15, to verify that the acceptance criteria and results appropriately considered differences between testing conditions and design conditions; that test results were appropriately categorized against pre-established acceptance criteria; that the frequency of testing was sufficient to detect degradation prior to loss of heat removal capability below design basis values; and that test results considered test instrument inaccuracies and differences. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (ISI)

.1 Piping Systems ISI

a. Inspection Scope

From October 7-17, 2007, the inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the RCS boundary and risk significant piping system boundaries. The inspectors reviewed a sample from activities performed during the Unit 1-Fall 2007 / Refueling Outage (1SR21) including nondestructive examinations (NDE) required by the 1989 Edition, no addenda, of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, and augmented examination commitments.

The inspectors observed and reviewed non-destructive examination (NDE) activities. Specifically:

Ultrasonic Examination (UT):
- CVC socket weld, elbow to pipe, weld # CVC-2599
- CVC socket weld, pipe op elbow, weld # CVC-2600
- Safety Injection valve to pipe weld, weld # SI-1605

- Safety Injection pipe to elbow weld, weld # SI-1606
- Pressurizer spray line elbow to safe end weld, weld # RCF-23
- Pressurizer relief line safe end to elbow weld, weld #: RCF-24
- Pressurizer safety line safe end to elbow weld, weld #: RCF 36
- Pressurizer safety line safe end to elbow weld, weld #: RCF 42
- Pressurizer safety line safe end to elbow weld, weld #: RCF 45
- Pressurizer nozzle to shell weld, weld # RCW 15

Penetrant Testing (PT):

- Reactor Coolant System (RCS) hanger, ID # 1-RCH-027-IA Visual Examination (VT):
- Chemical Volume Control System (CVCS) rigid support, ID # 1-CVCH-007
- Chemical Volume Control System (CVCS) rigid support, ID # 1-CVCH-010
- Reactor Vessel Bottom head penetrations #'s 29, 43, 8, 55, 49, 48, 14, 46

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the previously referenced ISI examination activities were reviewed and compared to requirements stated in ASME Section V, ASME Section XI, and other industry standards.

The inspectors reviewed welding activities from the previous outage. The inspectors reviewed drawings, work instructions, weld process sheets, weld travelers, pre-heat requirements and radiography records for welding of an ASME Class 1 and 2 pressure boundary weld.

The inspectors reviewed and observed weld overlay activities associated with the Pressurizer weld overlay activities. Specifically:

- Pressurizer spray nozzle, weld # RCW-24-SE
- Pressurizer safety relief valve nozzle, weld # RCW-25-SE
- Pressurizer safety relief valve nozzle, weld # RCW-27-SE
- Pressurizer safety relief valve nozzle, weld # RCW-28-SE

b. Findings

No findings of significance were identified.

.2 PWR Vessel Upper Head

The inspectors completed this activity by performance of Temporary Instruction 2515/150, Revision 3, which is documented in Section 4OA5.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

The inspectors reviewed the licensee's Boric Acid Corrosion Control (BACC) program to ensure compliance with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

The inspectors conducted an on-site record review as well as an independent walkdown of parts of the reactor building that are not normally accessible during at-power operations to evaluate compliance with licensee BACC program requirements and 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In particular, the inspectors verified 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 a sample of engineering evaluations completed 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 components. The inspectors also reviewed licensee corrective actions and problem evaluation reports (PERs), as well as corrosion assessments implemented for evidence of boric acid leakage to confirm that they were consistent with requirements.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

From October 12-17, 2007, the inspectors reviewed the Unit 1 SG tube eddy current testing (ECT) examination activities to ensure compliance with Technical Specifications (TS), applicable industry operating experience and technical guidance documents, and ASME Code Section XI requirements.

The inspectors reviewed licensee SG inspection activities to ensure that ECT inspections were conducted in accordance with the licensee's SG Program and applicable industry standards. The inspectors reviewed the SG examination scope, ECT acquisition procedures, site-specific Examination Technique Specification Sheets (ETSS), ECT analysis guidelines, the most recent SG degradation assessment, and the last operational assessment. The inspectors reviewed documentation to ensure that the ECT probes and equipment configurations used were qualified to detect the expected types of SG tube degradation, and a sampling of tube data was reviewed with a Level III analyst. The inspectors ensured that all tubes with relevant indications were appropriately screened for in-situ pressure testing. No tubes met the criteria for in-situ testing. Additionally, the inspectors monitored the licensee's secondary side activities, which included a foreign object search and recovery for loose parts, and sludge lancing.

b. Findings

No findings of significance were identified

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of SG and 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 Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed licensed operator requalification testing on December 3, 2007. The testing consisted of two scenarios requiring an Alert declaration. The first involved a steam generator tube rupture. An emergency diesel generator engine developed a lubricating oil leak requiring it to be taken out of service followed by a loss of main condenser vacuum. This resulted in operators initiating a manual reactor trip. Following the trip, a steam generator tube rupture occurred with the associated main steam isolation valve failing to operate, both requiring operator actions. The second scenario involved a motor trip of the running centrifugal charging pump and a loss of one main feed pump resulting in a turbine runback followed by a failed open pressurizer safety relief valve. This resulted in a manual reactor trip and safety injection initiation. This was compounded by all A-train emergency core cooling pumps failing to start automatically and a failed intermediate range nuclear instrument. The inspectors observed crew performance in terms of communications; ability to take timely and proper actions; prioritizing, interpreting and verifying alarms; correct use and implementation of procedures, including the alarm response procedures and emergency plan event classification; timely control board operation and manipulation, including high risk operator actions; oversight and direction provided by shift manager, including the ability to identify and implement appropriate TS actions; independent event classification by the Shift Technical Advisor; and group dynamics involved in crew performance. The inspectors also observed the examining staff's assessment of the crew's performance and compared them to inspector observations. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following two maintenance activities to verify the effectiveness of the activities in terms of: 1) appropriate work practices; 2) identifying and addressing common cause failures; 3) scoping in accordance with 10 CFR 50.65 (b); 4) characterizing reliability issues for performance; 5) trending key parameters for condition monitoring; 6) charging unavailability for performance; 7) classification in accordance with 10 Code of Federal Regulations (CFR) 50.65(a)(1) or (a)(2); 8) appropriateness of performance criteria for structure, system, or components (SSCs) and functions classified as (a)(2); and 9) appropriateness of goals and corrective actions for SSCs and functions classified as (a)(1). Documents reviewed are listed in the Attachment.

- PER 134770, Incorrect Containment Purge Exhaust Filter Replaced
- Unit 1 and Unit 2 Main Steam Isolation Valves

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following four activities to verify that the appropriate risk assessments were performed prior to removing equipment from service for maintenance. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65 (a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee’s risk assessment tool and risk categories in accordance with Procedure SPP-7.1, On-Line Work Management, Revision 10, and Instruction 0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 8. Documents reviewed are listed in the Attachment.

- ERCW B-Train Outage for Cross-Tie Piping Installation
- Testing Emergency Diesel Generator (EDG) 1B While in ORAM Orange Condition Due to Reduction in RCS Vent Path Area

- Removal of Unit 2 Turbine-Driven Auxiliary Feedwater (AFW) Pump from Service for Testing
- ERCW B Train Inoperable due to Motor-Operated Valve Testing on Component Cooling System (CCS) Heat Exchanger Outlet Valve 0-FCV-67-152

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the five operability evaluations described in the PERs listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred. The inspectors reviewed the UFSAR to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PERs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

- PER 132400, Unplanned Limiting Condition for Operation (LCO) Entry on EDG 1B
- PER 132653, Vital Inverter Declared Inoperable
- PER 133211, Foreign Material on Core Baffle Former Plate
- PER 133270, ERCW Low Flow Condition
- PER 120682, Tornado Effects on EDG Ventilation System

b. Findings

No findings of significance were identified.

1R17 Current Review of Ongoing Modifications

a. Inspection Scope

The inspectors reviewed DCN D22161-A, Install High Point Vents in ERCW Discharge Headers, and interviewed engineering personnel regarding the modification and associated post-modification testing to verify that (1) the design bases, licensing bases, and performance capability had not been degraded through this modification, and (2) the modification was not performed during increased risk-significant configurations that placed the plant in an unsafe condition. The inspectors also reviewed applicable sections of the UFSAR, plant modification procedures, system drawings, supporting analyses, technical specifications, and related PERs.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the four post-maintenance tests listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). Documents reviewed are listed in the Attachment.

- Work Order (WO) 07-780665-000, Adjust Motor-Operated Valve Striker Plates on 1-FCV-62-132 and 1-FCV-62-133
- WO 04-783124-000, Unit 1 Reactor Trip Switch Replacement
- WO 07-780645-000, Unable to Manually Stroke Atmospheric Relief Valve 1-PCV-01-005
- WO 05-777910-000, Troubleshoot/Repair Leakage of Valve 1-VLV- 063-0561

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 1 Refueling Outage

a. Inspection Scope

For the Unit 1 refueling outage that began on October 4, 2007, the inspectors evaluated licensee activities to verify that the licensee considered risk in developing outage schedules, followed risk reduction methods developed to control plant configuration, developed mitigation strategies for the loss of key safety functions, and adhered to operating license and TS requirements that ensure defense-in-depth. The inspectors also walked down portions of Unit 1 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between October 4 and November 17, 2007, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment.

- Outage Plan. The inspectors reviewed the outage safety plan and contingency plans to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth.
- Reactor Shutdown. The inspectors observed the shutdown in the control room from the time the reactor was tripped until operators placed it on the Residual Heat Removal (RHR) system for decay heat removal to verify that TS cooldown restrictions were followed. The inspectors also toured the lower containment as soon as practicable after reactor shutdown to observe the general condition of the RCS and emergency core cooling system components and to look for indications of previously unidentified leakage inside the polar crane wall.

- Licensee Control of Outage Activities. On a daily basis, the inspectors attended the licensee outage turnover meeting, reviewed PERs, and reviewed the defense-in-depth status sheets to verify that status control was commensurate with the outage safety plan and in compliance with the applicable TS when taking equipment out-of-service. The inspectors further toured the main control room and areas of the plant daily to ensure that the following key safety functions were maintained in accordance with the outage safety plan and TS: electrical power, decay heat removal, spent fuel cooling, inventory control, reactivity control, and containment closure. The inspectors also observed a tagout of the ERCW supply and discharge headers to verify that the equipment was appropriately configured to safely support the work or testing. To ensure that RCS level instrumentation was properly installed and configured to provide accurate information, the inspectors reviewed the installation of the Mansell level monitoring system. Specifically, the inspectors discussed the system with engineering, walked it down to verify that it was installed in accordance with procedures and adequately protected from inadvertent damage, verified that Mansell indication properly overlapped with pressurizer level instruments during pressurizer draindown, verified that operators properly set level alarms to procedurally required setpoints, and verified that the system consistently tracked RCS level while lowering to reduced inventory conditions. The inspectors also observed operators compare the Mansell indications with locally-installed ultrasonic level indicators during entry into mid-loop conditions.

  During the outage, the inspectors also reviewed the licensee's control of heavy loads to ensure the licensee was properly handling heavy loads in areas where a load drop could impact fuel in the reactor core or equipment that would be required to achieve safe shutdown. To do this the inspectors examined the licensee's basis for considering the containment polar crane to be single-failure proof in order to verify that it met industry standards, reviewed the polar crane testing and inspection done prior to lifting the reactor head, and observed the initial head lift to verify that it complied with the safe load path specified in licensee procedures.

- Refueling Activities. The inspectors observed fuel movement at the spent fuel pool and at the refueling cavity in order to verify compliance with TS and that each assembly was properly tracked from core offload to core reload. In order to verify proper licensee control of foreign material, the inspectors verified that personnel were properly checked before entering any foreign material exclusion (FME) areas, reviewed FME procedures, and verified that the licensee followed the procedures. To ensure that fuel assemblies were loaded in the core locations specified by the design, the inspectors independently reviewed the recording of the licensee's final core verification.

- Reduced Inventory and Mid-Loop Conditions. Prior to the outage, the inspectors reviewed the licensee's commitments to Generic Letter 88-17. Before entering reduced inventory conditions the inspectors verified 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. While in mid-loop conditions, the inspectors verified that licensee procedures for closing the containment upon a loss of decay heat removal were in effect, that operators were aware of how to implement the procedures, and that other personnel were available to close containment penetrations if needed.

- Heatup and Startup Activities. The inspectors toured the containment prior to reactor startup to verify that debris that could affect the performance of the containment sump had not been left in the containment. The inspectors reviewed the licensee's mode change checklists to verify that appropriate prerequisites were met prior to changing TS modes. To verify RCS integrity and containment integrity, the inspectors further reviewed the licensee's RCS leakage calculations and containment isolation valve lineups. In order to verify that core operating limit parameters were consistent with core design, the inspectors also observed portions of the low power physics testing, including approach to reactor criticality.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion V, for failing to remove or evaluate foreign material in the reactor vessel prior to installing the reactor vessel head as required by licensee procedure.

Description: On October 20, 2007 the licensee completed core reload as part of the outage. Following core verification per Procedure TI-45, Physical Verification of Core Load Prior to Vessel Closure, Revision 25, the licensee continued with reactor reassembly, set the reactor vessel head, and returned the RCS to service. The inspectors reviewed the video made during the core verification and identified several pieces of foreign material adjacent to the fuel assemblies that had not been previously identified by the licensee. The inspectors then reviewed licensee Procedure, SPP-6.5, Foreign Material Control, Revision 12, and concluded that the licensee failed to prescribe instructions or procedures of a type appropriate to the circumstances as required by SPP-6.5 and failed to identify and remove the foreign material prior to closing the system. Procedure TI-45 contained no instructions to look for foreign material in the core and no other process existed which verified foreign material had been removed prior to closing the reactor vessel. This programmatic gap in the licensee's foreign material control process could result in future recurrence.

Analysis: The finding was more than minor because the RCS was returned to service with unevaluated foreign material that could have been removed had it been properly identified by the licensee. This is similar to the more than minor example in IMC 0612, Appendix E, Example 5a. Furthermore, the inspectors determined the finding impacted the Human Performance attribute (FME Loose Parts) of the Barrier Integrity Cornerstone to maintain fuel cladding functionality. Licensee evaluation of the size, location, and characterization of the foreign material showed that the material was a small piece of metal, most likely a steel badge clip, in a low flow area, and would not likely carry over into other parts of the RCS. The licensee also determined that if it did carry over, it had a low probability of causing any fuel cladding damage, would not affect reactor coolant system integrity, and was bounded by previous foreign material evaluations. Since the finding affected only the fuel barrier and not the RCS barrier, the finding was of very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances. Licensee Procedure SPP-6.5 required that the responsible supervisor perform a final inspection of the system for foreign material with the intent that all foreign materials were accounted for and had not been left within the open system or component. Contrary to this, on October 20, 2007, the licensee failed to effectively implement Procedure SPP-6.5 by allowing foreign material, which could have been removed, to be left in the reactor coolant system. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as PER 133211, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000327/2007005-01, Failure to Effectively Implement Foreign Material Control Requirements in the RCS.

1R22 Surveillance Testing

a. Inspection Scope

For the five surveillance tests identified below, the inspectors verified that the SSCs involved in these tests satisfied the requirements described in the TS surveillance requirements, satisfied the UFSAR, applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions. This was accomplished by witnessing testing and/or reviewing the test data. Documents reviewed are listed in the Attachment.

- 1-SI-SXP-074-202.0, RHR Pump 1A-A and 1B-B Comprehensive Performance and Check Valve Test, Revision 2*
- 0-SI-SLT-088-259.4, Upper Personnel Airlock Interlock Operability Test, Revision 1
- 0-SI-MIN-061-105.0, Ice Condenser - Ice Weighing, Revision 5**

- 0-SI-MIN-061-109.0, Ice Condenser Intermediate and Lower Inlet Doors and Vent Curtains, Revision 5
- 1-SI-SLT-088-156.0, Containment Integrated Leak Rate Test, Revision 3

*This procedure included inservice testing requirements.
**This procedure included an ice condenser system surveillance.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification described in Temporary Alteration Control Form (TACF) 0-06-011-077, Piping From Floor Drain Collector Tank to the Condensate Demineralizer Waste Evaporator, Revision 2, and the associated 10CFR 50.59 screening, and compared it against the UFSAR and TS to verify that the modification did not affect the operability or availability of any safety system. The inspectors walked down the TACF to ensure it was installed in accordance with the modification documents and reviewed post installation and removal testing to verify the actual impact on permanent systems was adequately verified by the tests. The inspectors also verified that permanent plant documents were updated to reflect the TACF to ensure that plant configuration control as maintained. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls The inspectors evaluated licensee guidance and its implementation for controlling worker access to radiologically significant areas and monitoring jobs in-progress. The inspectors evaluated the adequacy of procedural guidance; directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation areas and high radiation areas (HRAs) within the Unit 1 (U1) and Unit 2 (U2) containment buildings, shared auxiliary building, external buildings, and the independent spent fuel storage installation (ISFSI). The inspectors independently measured radiation dose rates and contamination levels or directly observed conduct of licensee radiation surveys for selected Radiologically Controlled Area (RCA) locations. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were observed and evaluated for selected Locked High Radiation Area (LHRA) locations. Implementation of procedural guidance for LHRA and Very High Radiation Area controls were discussed in detail with health physics supervisors and management. Physical controls for storage of irradiated material within the spent fuel pool were observed. In addition, licensee controls for areas where dose rates could change significantly as a result of refueling operations or radwaste activities were reviewed and discussed.

The inspectors observed pre-job RWP briefings and reviewed RWP details, including engineering controls for potential airborne radioactivity and surface contamination, to assess communication of radiological control requirements. Radworkers' adherence to RWP guidelines and HPT proficiency in providing job coverage, including use of contamination controls and airborne surveys, were evaluated through observation of jobs in-progress. Jobs observed included U1 Steam Generator nozzle dam removal and manway closure, U1 Reactor Coolant System filter change-outs, a U2 at-power containment entry, and various refueling activities. Electronic dosimeter (ED) alarm set points were evaluated against area radiation survey results and ED alarm response actions were discussed with radworkers and HP supervisors. In addition to the jobs directly observed, inspectors also reviewed activities and documents associated with the U1 Pressurizer Overlay activities, and the removal and replacement of Hold-Up Tank "A."

The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Licensee evaluations of skin dose resulting from discrete radioactive particle or dispersed skin contamination events during the last refueling outage were reviewed and assessed.

For HRA tasks involving significant dose rate gradients, the inspectors evaluated procedural guidance and implementation for the use and placement of whole body and extremity dosimetry to monitor worker exposure, including the use of multiple badging during U1 Cycle 15 Refueling Outage (C15 RFO), and the last refueling outage. The inspectors also reviewed and discussed selected whole-body count analyses conducted during U1 C15 RFO and the last refueling outage.

Radiation protection (RP) activities were evaluated against the requirements of UFSAR Chapter 12; TS Sections 6.8 and 6.12; 10 Code of Federal Regulations (CFR) Part 20; and approved licensee procedures. Records reviewed are listed in Section 2OS1 and 4OA1 of the report Attachment.

Problem Identification and Resolution The inspectors reviewed and assessed select PERs associated with access control to radiologically significant areas. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure SPP-3.1, Corrective Action Program, Revision (Rev.) 12. In addition, the inspectors reviewed self-assessments related to the area of access controls. Specific corrective action program documents associated with access control issues, personnel radiation monitoring, and personnel exposure events reviewed and evaluated during inspection of this program area are identified in Section 2OS1 and 4OA5 of the report Attachment.

The inspectors completed 21 of the required line-item samples described in Inspection Procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) Implementation of the licensee's ALARA program during the U1 C15 RFO 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 U1 Initial Containment Entry, U1 Letdown Heat Exchanger work, and U1 Steam Generator nozzle dam removal and manway closure. Incorporation of planning, established work controls, expected dose rates, and dose expenditure into the ALARA pre-job briefings and RWPs for those activities were also reviewed. The inspectors directly observed performance of these activities while evaluating the licensee's use of engineering controls, low-dose waiting areas, and on-the-job supervision. The inspectors reviewed the licensee's exposure tracking system to determine whether it adequately supported control of collective exposures. RWPs were job-specific, with approximately 150 written for the current outage. EDs included administrative limits (warning to employee and manager, followed by ED lockout, if designated limits were exceeded) on individual worker exposure.

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 personnel. A review of chemistry activities included the U1 crudburst and use of hydrogen peroxide which resulted in a dose rate reduction of ~8 times from what they had observed before these activities had been conducted. Also, the inspectors reviewed the licensee's source-term control strategy.

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 October 2005 to September 2007 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 Part 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 PER documents listed in Section 2OS2 of the report Attachment that were 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, Rev. 12.

The inspectors completed 15 of the required line-item samples described in IP 71121.02.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Groundwater Monitoring The inspectors discussed current and future programs for onsite groundwater monitoring with licensee corporate staff, including number and placement of monitoring wells and identification of plant systems with the most potential for contaminated leakage. The inspectors also reviewed procedural guidance for identifying and assessing onsite spills and leaks of contaminated fluids. In addition, the inspectors reviewed records of historical contaminated spills retained for decommissioning purposes as required by 10 CFR Part 50.75(g).

In 2007, hydrological studies were performed and several new groundwater monitoring wells were installed. Analyses are performed for tritium and, for selected samples, hard-to-detect radionuclides. To date, tritium has been the only radionuclide identified in the well samples. One of the wells shows elevated levels of tritium due to historical spills. No levels exceeding the EPA drinking water limit of 20,000 picocuries per liter (corresponding to 4 millirem per year to a member of the public) have been identified in the onsite or offsite environs.

The inspectors completed one of the required line-item samples described in IP 71122.01.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included liquid waste demineralizer skids; resin transfer piping; floor drain collector tanks; and abandoned radwaste evaporators. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2006 Radioactive Effluent Report and radionuclide characterizations from 2005 - 2007 for each major waste stream were reviewed and discussed with radwaste staff. For Chemical Volume and Control Systems (CVCS) Resin and Dry Active Waste the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for resinous waste was evaluated and discussed with licensee staff.

The inspectors also reviewed the licensee’s procedural guidance for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee’s Process Control Program and UFSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, and guidance provided in the Branch Technical Position on Waste Classification and Waste Form. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Transportation The inspectors directly observed preparation activities for a shipment of contaminated laundry and Type A package containing equipment. The inspectors noted package markings and placarding, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations.

Five shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. For selected shipment records, the licensee's handling of Type B shipping casks was compared to Certificate of Compliance (CoC) requirements. In addition, training records and training curricula for individuals currently qualified to prepare shipments of radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution Selected PERs in the area of radwaste/shipping were reviewed in detail and discussed with licensee personnel. 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, Rev. 12. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Documents reviewed for problem identification and resolution are listed in Section 2PS2 of the report Attachment.

The inspectors completed 6 of the required line-item samples specified in IP 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the six PIs listed below for the period from July 1, 2006 through September 30, 2007 for both Unit 1 and Unit 2. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Revision 5, were used to verify the basis in reporting for each data element.

Cornerstone: Mitigating Systems

- Mitigating Systems Performance Index: Emergency AC Power
- Mitigating Systems Performance Index: High Pressure Injection System
- Mitigating Systems Performance Index: Heat Removal System (AFW)
- Mitigating Systems Performance Index: Residual Heat Removal System

- Mitigating Systems Performance Index: Cooling Water System
- Safety System Functional Failures

The inspectors reviewed portions of the operations logs and raw PI data developed from monthly operating reports and discussed the methods for compiling and reporting the PIs with engineering personnel. The inspectors also independently calculated selected reported values to verify their accuracy and compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report. Specifically for the Mitigating Systems Performance Index (MSPI), the inspectors reviewed the basis document and derivation reports to verify that the licensee was properly entering the raw data as suggested by NEI 99-02. For Safety System Functional Failures, the inspectors also reviewed LERs issued during the referenced timeframe. Documents reviewed are listed in the Attachment.

Cornerstone: Occupational Radiation Safety

The inspectors reviewed the Occupational Exposure Control Effectiveness PI results from May 2006 through September 2007. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and assessed corrective action program documents to determine whether HRA, VHRA, or unintended radiation exposures had occurred. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. In addition, the inspectors reviewed selected personnel contamination event data and internal dose assessment results. Report section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas. Documents reviewed are listed in sections 2OS1 and 4OA1 of the report Attachment.

Cornerstone: Public Radiation Safety

The inspectors reviewed records used by the licensee to identify occurrences of quarterly doses from liquid and gaseous effluents in excess of the values specified in NEI 99-02 guidance. Those records included monthly effluent dose calculations for October 2006 through September 2007. The inspectors also interviewed licensee personnel that were responsible for collecting and reporting the PI data. In addition, licensee procedural guidance for classifying and reporting PI events was evaluated. Reviewed documents are listed in Section 4OA1 of the report Attachment.

The inspectors completed two of the required samples for IP 71151, one sample for the OS PI and one sample for the PS PI.

b. Findings

Introduction: The inspectors identified an unresolved issue (URI) regarding licensee failure to submit accurate information regarding the Emergency AC Power Mitigating Systems Performance Indicator (MSPI).

Description: The inspectors reviewed the importance weighting ratios for both the unavailability and unreliability portions of the five different MSPI indicators as delineated in the MSPI basis document. The inspectors noted that, for Emergency AC Power, a separate ratio was specified for each EDG on each unit so that when calculating MSPI for Unit 1 there was one ratio for EDG 1A, one ratio for EDG 1B, one ratio for EDG 2A, and one ratio for EDG 2B. The importance of the Unit 1 EDGs was higher for the Unit 1 indicator than the Unit 2 EDGs with each Unit 2 EDG having an identical importance for Unit 1. The opposite was true for Unit 2. The Unit 2 EDGs were more important. However, when reviewing the derivation reports for the Emergency AC Power indicator, the inspectors noted that the same importance ratios were used on each EDG for each unit so that each EDG was equally important to each unit. The inspectors determined that the basis document was correct but the importance ratios had been improperly entered into the CDE database that calculated the Emergency AC Power MSPI. In addition, while reviewing the indicator as part of addressing inspector questions, engineering personnel determined that three previous failures not had not been classified properly.

The licensee had originally classified a failure of EDG 2A on October 3, 2005, which involved a broken sight glass on one of the generator bearings, as a demand failure based on the inability of the EDG to complete its function. After reviewing it further the licensee realized that the EDG would have started but would not have been able to complete its mission because the bearing would have failed due to loss of oil. Therefore they reclassified the failure of October 3, 2005, as a failure to run. The licensee also determined that two previous failures on July 20, 2006 and August 7, 2007, were not actually failures because the affected equipment was outside the boundary of the system.

Analysis: The as-reported numbers for the Emergency AC Power MSPI for the quarter ending September 2007 were -5.3E-7 for Unit 1 and -5.3E-7 for Unit 2. The effect of the improper use of the importance measures was to change the unreliability portion of the indicator from a negative to positive number while the total indicator remained negative. After adjusting for importance measures the numbers would have been -2.0E-7 for Unit 1 and -7.0E-8 for Unit 2. With the additional classification changes to the failure data the numbers became -1.12E-6 for Unit 1 and -1.17E-6 for Unit 2. While these numbers remained in the green band, the changes also affected earlier time periods. In two previous quarters, June 2006 and March 2007, the Unit 2 indicator was 1.04E-6. Because this information involved licensee failure to provide complete and accurate information concerning a ROP performance indicator, the inspectors determined that it had the potential to impact NRC ability to perform its regulatory function.

Enforcement: 10 CFR 50.9 requires that information provided to the NRC be complete and accurate in all material respects. Contrary to this, from July 1, 2006 until December 31, 2007, the licensee provided information regarding the Emergency AC Power MSPI indicator that was inaccurate. Specifically, the importance ratios for both the unavailability and unreliability portions of the indicator were improperly entered into the calculation for determining the indicator resulting in inaccurate reporting of the MSPI for Emergency AC Power. However, this item will remain unresolved pending NRC review of the previous data for the indicator and is identified as URI 05000327,328/2007005-02, Improper Information Provided for MSPI. This item has been entered into the licensee's corrective action program as PER 135288.

4OA2 Identification and Resolution of Problems

.1 Daily Review

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 was accomplished by reviewing the description of each new PER and attending daily management review committee meetings.

.2 Annual Sample Review of Breaker Problems that Resulted in Missed Preventive Maintenance (PM)

a. Inspection Scope

In March 2007, the licensee identified two breakers that required 60-month PM by Technical Requirements Manual (TRM) Surveillance Requirement 4.8.3.3 had exceeded the surveillance interval and the 25% grace period. In addition, several other safety-related breakers had also exceeded the PM frequency by more than 25%. While operators complied with TRM and TS requirements, problems controlling PM can develop into more significant issues. Therefore, in order to understand the cause and the work control process, the inspectors reviewed licensee actions to resolve this issue. The inspectors reviewed the PER dealing with this event, PER 120990; interviewed maintenance, engineering, and training personnel; and reviewed several of the corrective actions. Documents reviewed are listed in the Attachment.

b. Findings and Observations

There were no findings of significance identified during this review. The inspectors determined that the root cause was thorough and that immediate and long term corrective actions appeared to be adequate. The root cause team performed both a barrier analysis and event & causal factor analysis and determined that the transition of breaker preventive maintenance from the Surveillance Instruction (SI) scheduling program to the PM scheduling program lacked independent barriers to ensure breaker maintenance was kept current. They also concluded that the change management process had not provided an adequate barrier. The licensee developed several actions to address these causes and implemented them beginning in June 2007. These included designating a breaker program manager, training non-maintenance personnel on the change management process, developing a job familiarization worksheet on the change management process for maintenance supervisors, a review of breaker surveillance packages and breaker swap WO's to ensure the documentation was accurate, and a walkdown of all 480V and 6.9kV drawout type breakers to ensure that breaker data was properly entered in the plant database. The actions also included updating PM and WO scheduling tools with the correct information. The inspectors reviewed these actions and verified that they addressed the cause and were actually being implemented.

However, the inspectors noted that actions to train supervisors in the change management process were not completed even though stated in the PER. Additionally, the inspectors noted that the package review and breaker walkdown actions were not complete. The inspectors reviewed what had been completed on these actions and determined that the plant database had not been completely updated but that the majority of breakers were tracked properly. For those breakers that were not tracked, the shop personnel were able to determine the correct information. The inspectors found the process to be cumbersome and complicated, which could increase the potential for mistakes.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee’s corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors’ review was focused on repetitive equipment issues, but also included licensee trending efforts and licensee human performance results. The inspectors’ review nominally considered the six-month period of July 2007 through December 2007, although some examples expanded beyond those dates when the scope of the trend warranted. Specifically, the inspectors consolidated the results of daily inspector screening discussed in Section 4OA2.1 into a log, reviewed the log, and compared it to licensee trend reports for the period from January 2007 through October 2007 in order to determine the existence of any adverse trends that the licensee may not have previously identified. The inspectors also independently reviewed RCS leakage data for the six-month period of July 2007 through December 2007.

b. Findings and Observations

No findings of significance were identified. In general, the licensee had identified trends and appropriately addressed them in their corrective action program. The inspectors evaluated the licensee trending methodology and observed that the licensee had performed a detailed review. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in their data. The inspectors compared the licensee process results with the results of the inspectors’ daily screening and did not identify any discrepancies or potential trends that the licensee had failed to identify. There were two issues that had potential significance, both of which were tracked in the corrective action program.

Following the End-of-Cycle 14 Outage in December 2006 the licensee noticed higher levels of formaldehyde, two to three ppm, in the Unit 2 containment atmosphere prior to each weekly entry. To ensure worker safety, the licensee began increased purging of the Unit 2 containment in order to reduce concentrations down to less than 0.3ppm. This was a conservative decision as other methods were available to protect workers; however, by the end of the inspection period the licensee had used all of the 1000 hours allowed by TS for purging the containment and had not yet been successful in locating the source. During the inspection period, the licensee applied for and received a one time change to the TS to add 400 hours. Formaldehyde concentrations had previously decreased to approximately 1ppm prior to containment entries and that trend remained stable throughout the inspection period. The licensee has installed a filtration unit in the Unit 2 containment and has indicated they would proceed with a TS amendment request to remove the purge limit. The licensee has also contracted the services of a specialist to perform more detailed sampling on the Unit 2 containment as a further attempt to locate the source of the formaldehyde. Before the End-of-Cycle 15 Outage on Unit 1 the licensee also identified formaldehyde in the Unit 1 containment. These concentrations ranged from 0.2 to 0.6ppm and have been lower than that since the beginning of the new cycle in November 2007, typically less than 0.3ppm. Because of this, less purging has been required. The licensee has continued to monitor levels in Unit 1. The inspectors had no further concerns with the licensee's corrective actions associated with formaldehyde.

During the End-of-Cycle 15 Outage in November 2007, there were several instances on Unit 1 where foreign material was found in or entered unexpected places. These included the reactor vessel, the main condenser, the ice condenser, the main generator, and the reactor cavity. While each incident was properly entered into the corrective action program and none had more than minor safety significance, the actual introduction of foreign material into unexpected places represented a declining performance trend from recent outages. Prior to the Unit 1 outage foreign material incidents involved process issues by site personnel as opposed to actual foreign material intrusion. The licensee initiated plans to institute specific FME control plans for different areas like the spent fuel pit, reactor cavity, main turbine, and main generator. The inspectors had no current concerns with the licensee's corrective actions associated with FME but determined the trend should be monitored.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/150, Rev. 3, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009) (Unit 1)

a. Inspection Scope

From October 10 - 17, 2007, the inspectors reviewed the licensee's activities associated with the non-destructive examination (NDE) of the reactor pressure vessel head (RPVH) penetration nozzles, the bare metal visual examination of the top surface of the RPVH, and the visual examination to identify potential boric acid leaks from pressure-retaining components above the RPVH. These activities were performed in response to NRC Bulletins 2001-01, 2002-01, 2002-02, and the first revision of NRC Order EA-03-009, "Modifying Licenses," dated February 20, 2004, (hereafter referred to as the NRC Order).

The inspectors' review of the NDE of RPVH penetration nozzles included independent observation and evaluation of ultrasonic (UT) examinations (for both data acquisition and analysis), review of NDE procedures, personnel qualifications and training, and NDE equipment certifications. The inspectors also held interviews with contractor representatives (Areva) and other licensee personnel involved with the RPVH examination. The activities were reviewed to verify licensee compliance with the NRC Order and to gather information to help the NRC staff identify possible further regulatory positions and generic communications.

The inspectors reviewed a sample of the results from the volumetric UT examinations of RPVH penetration nozzles. Specifically, the inspectors reviewed or observed the following:

- Observed portions of in-process UT data acquisition scanning of RPVH penetration nozzle 23.
- Reviewed the UT electronic data with the Level III analyst for RPVH nozzles 1, 5, 19, 25, 45, 47, 68, 75, vent line, and the calibration block for the Auxiliary Head Adaptor (AHA) probe. Nozzles reviewed included CRDM penetrations both with and without thermal sleeves and one AHA.
- Reviewed the results of the UT examination performed to assess for leakage into the annulus (interference fit zone) between the RPVH penetration nozzle and the RPVH low-alloy steel for all penetrations listed in the previous bullet.
- Reviewed the procedures and results for the visual exam performed to identify potential boric acid leaks from pressure-retaining components above the RPVH.
- Reviewed the RPVH susceptibility ranking and calculation of effective degradation years (EDY), including the basis for the RPVH temperature used in the calculation.

b. <u>Observations and Findings</u>

In accordance with the requirements of TI 2515/150, the inspectors evaluated and answered the following questions:

1) Were the examinations performed by qualified and knowledgeable personnel?

Yes. The inspectors reviewed personnel training and qualifications to verify that volumetric and surface NDEs were performed by trained and qualified personnel. All examiners were qualified in accordance with the ASME Code and had additional training on RPVH examination, as required in Areva's "Written Practice for the Qualification and Certification of NDE Personnel" document.

2) Were the examinations performed in accordance with demonstrated procedures?

Yes. The Sequoyah Unit 1 RPVH has 57 control rod drive mechanism (CRDM) nozzles with thermal sleeves, 13 with open housings (including 5 instrument column nozzles), 8 with part lengths, 4 upper head injection (UHI) nozzles, and 1 vent line nozzle, for a total of 83 nozzles. All penetration nozzles, including the vent line, were examined by remote automated UT from the inside diameter (ID) surface in accordance with Areva approved procedures 54-ISI-604-004 for open bore penetrations not using a dummy sleeve, 54-ISI-603-003 for sleeved penetrations (including open bores which did utilize dummy sleeve), and 54-ISI-605-03 for small bore penetrations (i.e. vent line).

The inspectors found that Areva examination procedures for CRDM nozzles were demonstrated to be able to detect and size flaws in the RPVH nozzles in accordance with Electric Power Research Institute (EPRI) NDE Center's protocol contained in "Materials Reliability Program: Demonstration of Vendor Procedures for the Inspection of Control Drive Mechanism Head Penetrations (MRP-89)." Areva's equipment demonstration took place from August 14 to August 24, 2006. Areva had performed a similar demonstration in 2002 as documented in MRP-89. However, because Areva modified its equipment including changing the essential variables of the demonstration in 2002, the demonstration was repeated. The 2006 demonstration was performed with three RPVH nozzle mockups with multiple tube flaws representing the expected field degradations. These mockups were different from those used during the demonstration performed in 2002 (i.e. demonstration documented in MRP-89). The demonstration adopted security provisions from the EPRI Performance Demonstration Initiative protocol by restricting the access to the mockups and making them available to Areva only when the EPRI NDE personnel were present. EPRI letter to Mr. Joel Whitaker of Tennessee Valley Authority, dated October 8, 2007, documents the comparison of the recent Areva equipment demonstration with the previous demonstration performed in 2002. The letter states that the scatter observed is within the variability of the examination and the reliability of the examinations conducted with the new instrumentation will be comparable to the previous demonstration.

The procedure used for the RPVH vent line was not demonstrated under a specific program such as the EPRI MRP. This procedure was developed with NDE techniques similar to the CRDM procedures with regard to basic fundamental ultrasonic techniques. The procedure used for the PT examination of the vent line weld surface was developed in accordance with the ASME Code.

3) Was the examination able to identify, disposition, and resolve deficiencies?

Yes. All indications of cracks or interference fit zone leakage are required to be reported for further examination and disposition. Based on observation of the examination process, the inspectors considered deficiencies would be appropriately identified, dispositioned, and resolved. UT indications associated with the geometry and surface features of the examined volume were identified in several penetration tubes. None of the indications exhibited crack-like characteristics and were appropriately dispositioned in accordance with procedures.

4) Was the examination capable of identifying the primary water stress corrosion cracking (PWSCC) and/or RPVH corrosion phenomena described in the NRC Order EA-03-009?

Yes. The NDE techniques employed for the examination of RPVH nozzles had been previously demonstrated under the EPRI MRP/Inspection Demonstration Program as capable of detecting PWSCC-type manufactured cracks as well as cracks from actual samples from another site. Based on the demonstration, observation of in-process examinations, and review of NDE data, the inspectors determined that the licensee was capable of identifying PWSCC and/or corrosion as required by the NRC Order.

5) What was the physical condition of the RPVH (e.g. debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The licensee performed a 100% bare metal visual (BMV) inspection of the top of the RPVH, including 360° around each penetration using a remote visual robotic crawler for areas inside the lead shielding and underneath the raised insulation package. The surface sloping down from the shielding to the flange was visually inspected directly by a qualified VT-2 examiner. The inspectors independently reviewed portions of the remote inspection video, particularly in the area around penetration 75, which revealed boric acid crystals around the penetration and on the sloping head surface both above and below the penetration. This area was reinspected after cleaning, and the boric acid was justifiably attributed to an identified conoseal leak above the vessel head. The ultrasonic inspection confirmed that there was no through-wall leakage at the penetration. For the other areas of the head, no insulation, dirt, or other general debris was present that caused viewing obstructions in the areas of interest. The inspectors determined that the physical condition of the head and the actions taken by the licensee were adequate to meet the requirements of the NRC Order.

6) Could small boron deposits, as described in NRC Bulletin 2001-01, be identified and characterized?

Yes. The BMV examination was determined by the inspectors to be capable of identifying and characterizing small boron deposits as described in NRC Bulletin 2001-01. The remote exam was VT-2 qualified and able to resolve, at a minimum, the 0.105-inch characters on an ASME IWA-2210-1 Visual Illumination Card.

7) What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

There were no identified examples of RPVH penetration cracks, leakage, material deficiencies, head corrosion, or other flaws that required repair. As discussed previously, there were some UT indications at J-groove welds that were dispositioned as metallurgical/geometric indications (not service related). Additionally, there were some minor surface indications detected on some of the tubes, likely due to thermal sleeve centering pad wear.

8) 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)?

The penetration nozzles with thermal sleeves and centering pads did not impede effective collection of data. Concerning examination coverage, the NRC Order requires that each tube's volume is inspected from a minimum of 2 inches above the highest point of the J-groove weld to 2 inches below the lowest point of the J-groove weld, or 1 inch with a stress analysis. The licensee had performed a stress analysis and the inspectors verified that the minimum examination coverages required by the NRC Order were met.

9) What was the basis for the temperature used in the susceptibility ranking calculation?

NRC Order EA-03-009 requires that licensees calculate the effective degradation years (EDY) of the RPVH to determine its susceptibility category, which subsequently determines the scope and frequency of required RPVH examinations. The operating temperature of the RPVH is an input to this calculation. Therefore, an incorrect temperature input could result in placing the RPVH in an incorrect susceptibility category. The licensee uses the cold leg temperature in this calculation.

In Supplement No. 1 to the NRC's Safety Evaluation Report (SER) dated February 1980, the NRC concluded that scale model tests provided reasonable assurance that the upper head would operate at the cold leg temperature. However, the NRC staff also required that plant data be acquired to confirm the head temperature. The inspectors reviewed this data which confirmed that the head operated at approximately cold leg temperature with some minor thermocouple variations. In addition, both units underwent a modification since this testing to increase bypass flow to the head from 4% to about 7%. This gives further assurance that the RPVH operates at cold leg temperature. For these reasons, the inspectors concluded that the licensee had an adequate basis for their temperature input to the susceptibility ranking calculation, which results in Unit 1 being placed in the Low category.

10) During non-visual examinations, was the disposition of indications consistent with the NRC flaw evaluation guidance?

There were no indications considered to be flaws found during the RPVH examination.

11) Did procedures exist to identify potential boric acid leaks from pressure-retaining components above the RPVH?

Yes. Procedure 0-PI-DXX-068-100.R, Rev. 1, "Monitoring of Reactor Head Canopy Seal Welds for Leakage," is implemented every outage and meets the requirements of the NRC Order. However, inspection of conoseals and other bolted connections above the RPVH, such as the RVLIS line, are covered under the Boric Acid Program. The inspectors determined that the program and procedure implementation met the requirements of the NRC Order. The inspectors reviewed the inspection results for this outage and found that no indications of boric acid leakage from canopy seal welds were identified. However, as discussed previously, a boric acid leak from a conoseal connection was identified (see further discussion below).

12) Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPVH?

Yes. A conoseal leak was identified during inspection. The licensee performed appropriate follow-on examinations by tracing the leak down to the RPVH and noting the affected areas and penetrations. The noted areas were then cleaned and re-inspected to verify the integrity of the RPVH base metal. An action to fix the conoseal leak was also taken to prevent further leakage onto the RPVH. The inspectors reviewed the licensee's actions and determined that they were in accordance with the requirements of the NRC Order.

.2 (Closed) Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Units 1 and 2

a. Inspection Scope

The inspectors verified the Unit 1 implementation of the licensee's commitments documented in their September 1, 2005, response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors. The commitments included a permanent screen assembly modification, a license amendment request to change the UFSAR description of the sump screen analysis methodology, and submittal of a supplemental response to GL 2004-02. This review included the sump screen assembly installation procedure, screen assembly modification 10 CFR 50.59 evaluation, structural (debris) loading calculation, and validation testing of the modified sump screen design. The inspectors also reviewed the foreign materials exclusion controls and the completed Quality Assurance / Quality Control records for the screen assembly installation. The inspectors conducted a visual walkdown to verify the installed screen assembly configuration was consistent with drawings and the tested configuration. The inspectors also verified the design criteria for screen gap. Additionally, the inspectors reviewed the status of Unit 2 GL 2004-02 commitment items that were not verified complete during the Unit 2 TI 2515/166 inspection performed on December 12-13, 2006.

b. Findings and Observations

No findings of significance were identified.

The inspectors determined the following answers to the Reporting Requirements detailed in TI 2515/166-05 issued 5/16/07:

05.a TVA implemented plant modifications and procedure changes at Sequoyah committed to in their GL 2004-02 response for Unit 1.

05.b TVA updated the Sequoyah Unit 1 licensing bases to reflect the corrective actions taken in response to GL 2004-02.

05.c No extensions of 12/31/2007 deadline for GL 2004-02 commitment completions have been applied for or granted to Sequoyah Unit 1. An extension may be sought based on the results of ongoing chemical effects testing to validate the design.

Unit 2 GL 2004-02 commitment items were complete.

TI 2515/166 is closed for Sequoyah Unit 1 and Unit 2, no additional modifications or procedural changes under GL 2004-02 are anticipated.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 9, 2008, the resident inspectors presented the inspection results to Mr. Timothy Cleary and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

An interim exit was conducted on October 19, 2007, to discuss the findings of the TI2515/166 inspection. Although proprietary information was reviewed during the inspection, no proprietary information is included in this report.

In addition, on October 26, 2007, the inspectors discussed results of the onsite radiation protection inspection. The inspectors noted that proprietary information was reviewed during the course of the inspection but would not be included in the documented report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Bodine, Chemistry/Environmental Manager
D. Boone, Radiation Protection Manager
C. Church, Plant Manager
K. Clayton, Maintenance Manager
T. Cleary, Site Vice President
L. Cross, Maintenance Shop Superintendent
B. Dungan, Outage and Site Scheduling Manager
K. Jones, Engineering Manager
Z. Kitts, Licensing Engineer
A. Little, Acting Site Security Manager
T. Marshall, Operations Superintendent
G. Morris, Licensing Manager
M. Palmer, Operations Manager
K. Parker, Maintenance and Modifications Manager
J. Proffitt, Licensing Engineer
J. Smith, Licensing Supervisor and Industry Affairs Manager
N. Thomas, Licensing Engineer
K. Wilkes, Emergency Preparedness Manager

NRC personnel:

R. Bernhard, Region II, Senior Reactor Analyst
B. Moroney, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000327,328/2007005-02	URI	Improper Information Provided for MSPI (4OA1).

Opened and Closed

05000327/2007005-01	NCV	Failure to Effectively Implement Foreign Material Control Requirements in the RCS (Section 1R20).
TI 2515/150	TI	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles Sequoyah Unit 1 (NRC Order EA-03-009) (Section 4OA5.1)

TI 2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) - Units 1 and 2) (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section R01: Adverse Weather Protection

SPP-10.14, Freeze Protection, Revision 0
0-PI-OPS-000-006.0, Freeze Protection, Revision 46
0-GO-14-2, Operator Rounds – Aux Bldg Round, Revision 19
1-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Revision 33
2-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Revision 19
1,2-45W1635-92, Wiring Diagrams Local Instrument Panels Connection Diagrams, Revision 6

Section R06: Flooding

Sequoyah Nuclear Plant Probabilistic Safety Assessment Individual Plant Examination Volume 3, Revision 1
WO 04-775349-000, Turbine Building Station Sump Level Switch Functional Test
WO 05-775248-000, Turbine Building Demin Sump Alarm Level Switch Functional Test
WO 06-777324-000, Turbine Building Oil Sump Level Alarm Level Switch Functional Test

Section R07: Heat Sink Performance

FSAR 9.2.1, Component Cooling System
1,2-47W859-1, Mechanical Flow Diagram – Component Cooling System, Revision 52

Section R08: Inservice Inspection Activities

0-TI-DXX-000-097.1, Boric Acid Corrosion Control program, Rev. 0001
N-UT-65, Generic Procedure for the Ultrasonic Through Wall Sizing in pipe Welds, Rev. 4
1-SI-SXI-068-114.3, Steam Generator Tubing Inservice Inspection and Augmented Inspections, Rev. 0001
0-SI-DXI-000-114.3, ASME SECTION XI ISI/NDE PROGRAM UNIT 1 and UNIT 2, Rev. 0002
1-SI-SXI-068-114.3, Steam Generator Tubing Inservice Inspection and Augmented Inspections, Revision 1
Degradation Assessment for Sequoyah Unit 1 Cycle 15
Operational Assessment Report from Unit 1 Cycle 13 Refueling Outage
Unit 1 Cycle 15 Replacement Steam Generator Tubing Examination Scan Plan, Revision 0
Self Assessment CRP-ENG-009 SQN ASME Section XI Program
Self Assessment 06SQN-12-ENG-XI ASME Section XI Inservice Inspection (ISI) Program
SQN-ENG-03-007 Boric Acid Program Effectiveness Assessment
SPP-9.7, Corrosion Control Program, Rev. 13
Technical Instruction 0-TI-DXX-000-097.1, Rev. 01, Boric Acid Corrosion Control Program
BP-257, Rev. 5, TVA Business Practice, Integrated Material Issues Management Plan, App. A

N-UT-76, Rev. 6, Generic Procedure for Ultrasonic Examination of Ferritic Pipe Welds.
N-UT-64, Rev. 9, Generic Procedure For The UT Examination of Austenitic Pipe Welds
N-VT-1, Visual Examination Procedure for ASME Section XI Preservice and Inservice
N-VT-15, Rev. 5, Visual Examination of Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants

0-VI-MOD-068-001.0, Vendor Instruction for Welding Activities Associated with Alloy 690 Weld Overlays, Rev. 0002
PER 100694, Leak check of the main generator bushings for H2 leakage prior to depressurizing the generator
PER 100931, U-1 wafer valve floor penetration has signs of leakage of dried boric acid.
PER 100940, Boric acid corrosion: 1-VLV-67-661A, Borated Water Leak.
PER 104474, SQN-1-VLV-062-0546 has packing leakage (one drop per minute) identified under WO 06-775022-000
PER 101708, A crack indication was identified on Unit 1 Low Pressure Turbine (TN 12751)at balance hole 27 on the inlet side of disc number 2 (generator end) with depth measuring 1.15 in.
PER 100929, When preparing to add chemicals to U1 S/G #2 a leak was discovered at a hose coupling in the mainsteam vent line.
PER 101471, During the Unit 1 Cycle 14 Reactor Vessel internal examination by the Inservice Inspection Organization (ISO) two indications were noted on the vessel outlet nozzle mating surface for the # 4 Hot Leg (ISI inspection location N-18).
PER 117945, Missed inspection of 11 SG tubes during U2C13 contrary to Technical Specifications

Section R11: Licensed Operator Requalification

E-0, Reactor Trip or Safety Injection, Revision 29
ES-0.5, Equipment Verifications, Revision 0
E-1, Loss of Reactor or Secondary Coolant, Revision 23
E-3, Steam Generator Tube Rupture, Revision 17
FR-Z.1, High Containment Pressure, Revision 17
AOP-S.01, Loss of Normal Feedwater, Revision 12
AOP-S.02, Loss of Condenser Vacuum, Revision 10

Section R12: Maintenance Rule Implementation

WO 07-779781-000, Replace Unit 2 B-train Containment Purge Air Exhaust Charcoal Filter
Tagout 2-TO-2007-0018, Clearance 2-30-0712
SPP-10.2, Clearance Procedure to Safely Control Energy, Revision 11
SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting - 10CFR50.65, Revision 9
TI-4, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting - 10CFR50.65, Revision 0020
Sequoyah System Status Report, Main Steam System, October 2007

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

1,2-47W845-2, Mechanical Flow Diagram - Essential Raw Cooling System, Revision 94

1,2-47W845-4, Mechanical Flow Diagram - Essential Raw Cooling System, Revision 16
1,2-47W845-6, Mechanical Flow Diagram - Essential Raw Cooling System, Revision 30
Sentinel Runs - October 9 to October 23, 2007
Tagout 1-TO-2007-0017, Component Cooling Heat Exchanger Discharge Valve to Header B
ORAM-Sentinel Outage Safety Assessment for October 30, 2007
Sentinel Run – November 12 to December 2, 2007
Sentinel Run – December 10 to December 23, 2007

Section R15: Operability Evaluations

1,2-45N779-6, Wiring Diagram, 480V Shutdown Aux Power Schematic Diagrams Sheet 6, Revision 13
1,2-45N765-3, Wiring Diagram, 6900 Volt Shutdown Aux Power Schematic Diagram Sheet 3, Revision 22
1,2-45N765-4, Wiring Diagram, 6900 Volt Shutdown Aux Power Schematic Diagram Sheet 4, Revision 3
1,2-45N700-1, Key Diagram – 12V AC and 125V DC Vital Plant Control Power System, Revision 40
Functional Evaluation 42301 – Vital Inverter Switch Lug Load Capacity
NEDP-22, Functional Evaluations, Revision 5
OPN218E.018, Electrical Training Lesson Plan – 120V AC Systems
1-SI-SFT-067-739.0, ERCW Lower Containment Flow Balance, Revision 6
MDQ0300-980037, Diesel Generator Operabililty with Outside Air Dampers Closed Based on Room Temperature, Revision 0

Section R19: Post Maintenance Testing

1-SI-OPS-000-009.0, Actuation of ECCS and Boron Injection Flowpath Valves via SI Signal, Revision 1
PER 132143, MOV Striker Plates Misadjusted
SI-93, Reactor Trip Instrumentation Functional Tests Conditional 31 Days Prior to Startup, Revision Draft 28B
PER 132552, Reactor Trip Breakers Did Not Close
PER 132554, New Reactor Trip Switch is Sensitive
1,2-45N699-1, Wiring Diagrams Reactor Protection System Schematic Diagrams Sheet 1, Revision 7
1-45N1624-11, Wiring Diagram Reactor Trip Switchgear Connection Diagram Sheet 11, Revision 1
1-PI-OPS-000-003.0, Periodic Stroking of Unit 1 Time Critical Valves, Revision 1
1-SI-SXV-063-206.0, Residual Heat Removal Primary and Secondary Check Valve Integrity Test, Revision 9
1-47W811-1, Flow Diagram Safety Injection System, Revision 72

Section R20: Refueling and Outage Activities

1-PI-IXX-068-005.0, Installation and Removal of the Mansell Level Monitoring System During Refueling Outages, Revision 12

0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 58
1,2-47W813-1, Flow Diagram Reactor Coolant System, Revision 52
0-MI-MRR-068-005.0, Removal of Reactor Pressure Vessel Head and Attachments, Revision 28
0-MI-MXX-000-026.0, Control of Heavy Loads in Critical Lifting Zones NUREG-0612, Revision 17
0-MI-ECR-303-911.0, Reactor Building (Polar) Crane Periodic Inspection, Revision 5
WO 06-781659-000, Reactor Pressure Vessel Disassembly
WO 06-777976-000, Unit 1 Polar Crane Inspection
TVA Response to Phase I Requests of Generic Letters 80-110 and 81-07, dated March 1, 1982
TVA Response to Phase I Requests of Generic Letters 80-110 and 81-07, dated February 25, 1983
TVA Response to Phase I Requests of Generic Letters 80-110 and 81-07, dated February 28, 1984
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TVA Response to Phase I Requests of Generic Letters 80-110 and 81-07, dated December 7, 1984
TVA Response to Phase II Requests of Generic Letters 80-110 and 81-07, dated January 24, 1985
NRC Safety Evaluation Report on Control of Heavy Loads, dated March 26, 1985
DCN M-06332-A, Install Removable Shielding to Reactor Head Lifting Columns, Revision 0
SQN Unit 1 Cycle 15 Outage Safety Plan, Revision C Schedule
NUMARC 91-06 Guidelines for Industry Actions to Assess Shutdown Management
0-GO-15, Containment Closure Control, Revision 23
TVA 90-Day Response Letter to Generic Letter 88-17, dated February 2, 1989,
FHI-3, Movement of Fuel, Revision 51
0-PI-OPS-068-673.W, Weekly Requirements for Modes 5 and 6 Operations, Revision 10
SQN-SQS2-0133, Midloop Design Information Calculation, Revision 6
0-TI-OXX-068-001.0, Reactor Coolant System Hot Leg Vents and Generic Letter 88-17 Issues, Revision 15
1-PI-OPS-068-673.D, Daily Requirements for Reduced Inventory/Midloop Operation, Revision 11
TI-45, Physical Verification of Core Load Prior to Vessel Closure, Revision 25
PER 126835, Containment Closure Under Adverse Conditions
0-SI-OPS-000-011.0, Containment Access Control During Modes 1-4, Revision 28
0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 21
0-RT-NUC-000-008.0, Low Power Physics Testing Acceptance Criteria, Revision 8
0-GO-2, Unit Startup From Hot Standby to Reactor Critical, Revision 28

Section R22: Surveillance Testing

FSAR Section 6.3, Emergency Core Cooling System
FSAR Figure 6.3.2-5, RHR Pump Minimum ECCS Performance Curve
1,2-47W810-1, Flow Diagram Residual Heat Removal System, Revision 50
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USNRC memo of December 13, 1999, NRR Response to TIA 99-02, Adequacy of Sequoyah Ice Condenser Ice Bed and Baskets

R.G. 1.163, Performance Based Containment Leak Test Program
NEI 94-01, Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50 Appendix J, Revision 0
ANSI/ANS 56.8-1994, Containment System Leakage Testing Requirements

Section R23: Temporary Plant Modifications

WO 06-772888-000, Install CDWE TACF
0-SO-77-7, Floor Drain Collector Tank, Revision 6
UFSAR Section 9.3.3, Equipment and Floor Drainage
UFSAR Section 11.2, Liquid Waste Systems
1,2-47W830-2, Mechanical Flow Diagram – Waste Disposal System, Revision 29
1,2-47W830-2, Mechanical Flow Diagram – Waste Disposal System, Revision 30
1,2-47W830-7, Mechanical Flow Diagram – Waste Disposal System, Revision 18
1,2-47W830-7, Mechanical Flow Diagram – Waste Disposal System, Revision 19
SPP-9.5, Temporary Alterations, Revision 8

Section 2OS1: Access Control To Radiologically Significant Areas

Tennessee Valley Authority (TVA), Sequoyah Nuclear Plant (SNP), Radiological Control Instruction (RCI)-01, Radiation Protection Program, Rev. 64
TVA, SNP, RCI-03, Personnel Monitoring, Rev. 48
TVA, SNP, RCI-11, Bioassay Program, Rev. 15
TVA, SNP, RCI-14, Radiation Work Permit (RWP) Program, Rev. 39
TVA, SNP, RCI-15, Radiological Postings, Rev. 15
TVA, SNP, RCI-16, Radiography, Revisions 10 and 11
TVA, SNP, RCI-21, Control of Radioactive Materials, Rev. 13
TVA, SNP, RCI-22, Contamination Control, Rev. 16
TVA, SNP, RCI-24, Control of Very High Radiation Areas, Rev. 8
TVA, SNP, RCI-28, Control of Locked High Radiation Areas, Rev. 6
TVA, SNP, RCI-29, Control of Radiation Protection Keys, Rev. 7
TVA, SNP, Radiation Protection Management Directive (RMD) FO-02, Radiation and Contamination Surveys, Rev. 19
TVA, SNP, RMD FO-03, Alpha Contamination Monitoring and Controls, Rev. 0
TVA, SNP, RMD FO-08, Radiological Surveys of Equipment and materials leaving the RCA, Rev. 0
TVA, SNP, Technical Instruction 0-TI-NUC-000-002.0, Storing Material in Spent Fuel Pool or New Fuel Vault
TVA Standard Programs and Processes (SPP) - 3.1, Corrective Action Program, Rev. 12
Air Sample Survey Numbers (Nos.) 101307005, 101307023, 101307030, 101307050, 101407021, 101407032, 101507019, 102507009, and 102507010
Internal Dose Calculation Report generated on 10/14/07
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Printout of 20 Individuals with Highest TEDE Dose for 2006 and 2007, printed on 10/02/07
Printout of PER summaries for all personal contaminations from May 2006 through October 2007

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RWP No. 07024240, Rev. 0, Unit 1 Lower Containment, Routine Plant Maintenance
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Sequoya Nuclear Station (SNS) Visual Survey Data System (VSDS) Survey Nos. 080207-5, 100607-5, 100907-1, 100907-2, 100907-5, 100907-25, 101007-7, 101507-3, 101107-2, 101607-6, 102107-6, 102207-15, and 102307-5
07-SQN-30-RAD-RD, Snapshot Self-Assessment Report, February 12 - 13, 2007
Problem Evaluation Report (PER) 112196, ALPHA Monitoring
PER 115630, VHRA Key Control - Procedural Violation
PER 130039, Radiographer dose rate alarm
PER 131962, U1C15 Personal Contaminations
PER 132777, Radcon has posted signs on U1 and U2 West Valve Vault Room doors
PER 132792, Access at 714'
PER 132821, Clothing impact on skin dose assessment
PER 132822, PCM-1B Trouble Light
SQN-RP-06-002, Self-Assessment Report, Dated 09/07/06

<u>2OS2 ALARA Planning and Controls</u>

TVA, SNP, RCI-10, ALARA Program, Rev. 30
TVA, TVAN RCDP - 105, Personnel Inprocessing and Dosimetry Administrative Processes, Rev. 0
TVA, TVAN SPP, SPP - 3.1, Corrective Action Program, Rev. 12
TVA, TVAN SPP, SPP - 5.2, ALARA Program, Rev. 3
ALARA Outage Report for U2C14
ALARA Outage Report for U1C14
Dose Records of all declared pregnant workers (4) during the period 10/05 to 09/07
Fiscal Year 2005 and 2006 Personnel Exposure by Section and Goals
List of Active Hot Spots, Dated 08/229/07
RCI-19, Rev. 10, Appendix A, Temporary Shielding Request Form, U1 C15 Upper Containment Reactor Head Stand
RWP No. 07024240, U-1 Containment – General Walkdowns of Various Systems, Engineering Activities to Include: Installation of /Removal of ECCS Flow Test Equipment, Manipulation of Throttle Valves and Tech Support Temperature and Pressure Walkdowns for Mode 3
RWP No. 07024260, U-1 Lower Containment, Inside Excess Letdown Heat Exchanger Room, Inspections, Tests and Valve Alignments
RWP No. 07034020, U-1 Lower Containment Steam Generators 1-4, Full Jump for Installing and Removing Nozzle Dams
RWP No. 07034080, U-1 Lower Containment – IPCW/)PCW and 734’ Elevation, Aux Building, Radcon, Laborer, Boilermaker and Westinghouse Support to include Equipment Monitoring, Movement of Equipment into and out of Zones, Transferring Trash, Cleaning Stud Bolts (No Entry to Platform)
RWP No. 07044131, U-1 Upper Containment/Reactor Cavity: Tension/Detension Reactor Reactor Head Stud Bolts and Associated Work

SNS VSDS Survey Report, Survey Nos. 100407-04, 1R112.WMF – R112 U1 #2 RCP Seal Platform, Dated 10/04/07; and 100407-16, 1R141.WMF – R141 U1 Inside Polar Crane Wall, Dated 10/04/07
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SQN ALARA Planning Report (APR) 2007-10, U1 C15 RFO, U1 Upper Containment, Aux Building, DAW Building and other areas as necessary
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SQN APR 2007-31, U1 C15 Reactor Head Penetration Volumetric Exams
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0-SO-77-29, Waste Processing, Rev. 11
Process Control Program, Rev. 3
Radioactive Material Shipment Manual, Vols. II & III, Rev. 38
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RWTP-100, Radioactive Material/Waste Shipments, Rev. 4
RWTP-101, 10 CFR 61 Waste Characterization, Rev. 0
RWTP-102, Use of Casks, Rev. 1
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Section 4OA1: Performance Indicator Verification

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Section 4OA2: Identification and Resolution of Problems

SI-266.2.1, Inspection Verification of ITE 7.5HK-500 6900V Breakers and Siemens 6900V Vacuum Breakers, Revision 29
SPP-2.1, Administration of Standard Programs & Processes and Standard Department Procedures, Revision 12
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