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

July 28, 2005

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

SUBJECT: SEQUOYAH NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2005003 AND 05000328/2005003

Dear Mr. Singer:

On June 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Power Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 12, 2005, with Mr. R. Douet 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.

Based on the results of this inspection, the inspectors identified four issues of very low safety significance (Green). Three of these issues were determined to be violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Furthermore, one licensee-identified violation which was determined to be of very low safety significance (Green) is listed in Section 4OA7. If you contest these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report 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 facility.

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

Sincerely,

/RA/

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

Docket No.: 50-327, 50-328
License No.: DPR-77, DPR-79

Enclosure: Inspection Report 05000327/2005003 and 05000328/2005003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

cc w/encl:
Ashok S. Bhatnagar
Senior Vice President
Nuclear Operations
Tennessee Valley Authority
Electronic Mail Distribution

Larry S. Bryant, General Manager
Nuclear Engineering
Tennessee Valley Authority
Electronic Mail Distribution

Randy Douet
Site Vice President
Sequoyah Nuclear Plant
Electronic Mail Distribution

Robert J. Beecken, Vice President
Nuclear Support
Tennessee Valley Authority
Electronic Mail Distribution

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

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

Glenn W. Morris, Manager
Corporate Nuclear Licensing and
Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

Paul L. Pace, Manager
Licensing and Industry Affairs
ATTN: James D. Smith
Sequoyah Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

David A. Kulisek, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

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

Ann Harris
341 Swing Loop
Rockwood, TN 37854

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

Distribution w/encl: (See page 4)

Distribution w/encl:
Bob Pascarelli, NRR
D. Pickett, NRR
L. Slack, RII EICS
RIDSNRRDIPMLIPB
PUBLIC

☐ SISP REVIEW COMPLETE: Initials: ________ ☐ SISP REVIEW PENDING*: Initials: ________ *Non-Public until the review is complete
☐ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☐ NON-SENSITIVE
ADAMS: ☐ Yes ACCESSION NUMBER:________________________

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	MFS per email	MES per email	WCB per email	MSL for	MSL for	DWJ	WTL per email
NAME	SFreeman	MSpeck	WBearden	MMaymi	AVargas	DJones	WLoo
DATE	07/28/2005	07/28/2005	07/28/2005	07/28/2005	07/28/2005	07/27/2005	07/27/2005
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
OFFICE	RII:DRS	RII:DRS					
SIGNATURE	JLK per email	JXD per email	LRM per email				
NAME	JKreh	JDiaz	LMiller				
DATE	07/27/2005	07/27/2005	07/27/2005	July 29, 2005	July 29, 2005	July 29, 2005	July 29, 2005
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report No:	05000327/2005003 and 05000328/2005003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	S. Freeman, Senior Resident Inspector M. Speck, Resident Inspector M. Maymi, Reactor Inspector (Section 1R07) W. Bearden, Senior Resident Inspector, Browns Ferry (Section 1R08) A. Vargas, Reactor Inspector (Section 1R08, 4OA5.5) D. Jones, Senior Fuel Facility Inspector (Sections 2OS1, 2OS2) W. Loo, Senior Radiation Protection Inspector (Sections 2OS1, 2OS2, 2PS2, 4OA1, 4OA5) J. Kreh, Emergency Preparedness Inspector (Sections 2OS1, 2OS2, 4OA1) J. Díaz Vélez, Health Physicist, (Sections 2OS1, 2OS2, 2PS2) L. Miller, Senior Emergency Preparedness Inspector (Sections 1EP2 - 5)
Approved by:	S. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2005003, IR 05000328/2005003; 04/01/2005 - 06/30/2005; Sequoyah Nuclear Power Plant, Units 1 & 2; Flood Protection Measures, Refueling and Outage Activities, Event Followup.

The report covered a three-month period of inspection by resident inspectors and announced inspections by eight region-based inspectors. 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A finding was identified for a self-revealing failure to implement effective corrective actions for oil port misalignment between the main turbine front pedestal and the turbine protective trip block. Improper use of gasket material to correct a previous problem with oil seepage from the main turbine trip block resulted in the loss of auto stop oil pressure and a turbine and reactor trip.

  This finding was more than minor because it affected the design control attribute of the initiating event cornerstone and upset plant stability by causing a reactor trip. This finding was of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss-of-coolant initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or flood. Because the affected equipment was non-safety related, no violation of regulatory requirements was identified. The cause of this finding was associated with the cross-cutting area of Problem Identification and Resolution (Section 4OA3.1).

- Green. A non-cited violation of Technical Specification 6.8.1 was identified for a self-revealing failure to have adequate work procedures for testing molded case circuit breakers associated with 125-volt Vital Battery Board IV. The procedures included provisions for installing threaded rods in place of the panel mounting bolts to maintain positive positional control while removing and reinstalling panel covers, but these provisions applied only to 120-volt Vital Instrument Power Boards. While installing a breaker in Battery Board IV, the panel cover slipped, opened a control power breaker, and tripped the Unit 2 reactor.

  This finding was more than minor because it affected the procedure quality attribute of the initiating event cornerstone and upset plant stability by causing a reactor trip. This finding was of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss-of-coolant initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or flood (Section 4OA3.2).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1 for an inadequate procedure to mitigate the probable maximum flood. Should the postulated flood event have occurred, conflicts between different sections of the procedure, conflicts between steps within one section of the procedure, and a missing step would have lead to a loss of decay heat removal or a loss of reactor coolant system inventory for a unit in a refueling outage.

  This finding was more than minor because if the procedure problems were left uncorrected the result would be a more significant safety concern. This finding was of very low safety significance due to the low frequency of occurrence for the probable maximum flood and because the mitigating equipment for a loss of decay heat removal or reactor coolant inventory during a refueling outage would not be affected (Section 1R06.2).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1 for a self-revealing failure to follow plant procedures prior to and during draining of the fuel transfer canal. Leakage past the spent fuel pit gate seal resulted in inadvertently transferring approximately 10,000 gallons of spent fuel pit inventory to the refueling water storage tank.

  This finding is more than minor because it affected the Barrier Integrity cornerstone, in that operators failed to adhere to procedures while changing plant configurations resulting in a loss of spent fuel pit inventory. Additionally, if left uncorrected, it would become a more significant safety concern. The cause of this finding is related to the cross-cutting area of human performance. This finding is of very low safety significance because it represented only a small degradation of the radiological barrier function provided by the spent fuel pit (Section 1R20).

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100% rated thermal power (RTP) during the inspection period except for an automatic reactor trip on April 9, 2005, when a gasket in the main turbine auto stop oil system failed, resulting in a low auto stop oil pressure main turbine trip and a subsequent reactor trip. Following repairs, the unit was restarted on April 12, 2005 and the unit returned to 100% RTP on April 15, 2005. The unit remained at or near 100% RTP through the end of the inspection period.

Unit 2 began the period at 100% RTP. On April 25, 2005, the unit was shutdown for a scheduled refueling outage. Outage activities were completed and the unit was restarted on May 28, 2005. The unit returned to 100% RTP on June 1, 2005, and remained at or near 100% RTP through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns. The inspectors performed a partial walkdown of the following three systems 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 systems 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. Documents reviewed are listed in the attachment.

- EGTS Train B during Unit 2 ERCW ESF Header Outage Work
- Motor Driven AFW Trains 2A and 2B during Turbine Driven AFW Valve Steam Leak Repairs
- Emergency Diesel Generators 1B, 2A, and 2B during Diesel Generator 1A Outage

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 procedures; 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.

- Control Building Elevation 706 (Spreading Room)
- Control Building Elevation 669 (250 VDC Battery and Battery Board Rooms)
- Auxiliary Building Elevation 734 (Shutdown Board Rooms, Battery Board Rooms, and Auxiliary Control Room)
- Emergency Diesel Generator Building
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Unit 2 Reactor Coolant Pump Oil Collection System
- Auxiliary Building Elevation 714 (Corridor)
- Auxiliary Building Elevation 690 (Corridor)
- Control Building 732 (Mechanical Equipment Room and Relay Room)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed the ECCS room flood detection system to verify that a leak in one train of ECCS equipment would not cause a loss of function in the opposite train due to flooding. The inspectors reviewed ECCS passive failure criteria and subsequent leakage sub-sections of Chapter 6 of the UFSAR, walked down the various ECCS pump rooms to verify the material condition of flood detection equipment, reviewed the periodic test procedures of several detectors to ensure that they were properly calibrated, and observed the RHR pumps during shutdown of Unit 2 to verify leakage through the pump seal was minimal. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 External Flooding

a. Inspection Scope

The inspectors reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to verify that it properly implemented methods described in the UFSAR. This procedure included different sections for different operating modes, however, for this review, the inspectors focused on flood mitigation with one unit in a refueling outage with the head removed while the other unit remained in power operation. The flooding AOP also included provisions for installing spool pieces in different sections of piping throughout the plant. In order to verify that these pieces were properly fabricated and staged, the inspectors walked down the spent fuel pit heat exchangers, the RHR heat exchangers, and the spent fuel pit pumps. Documents reviewed are listed in the attachment.

b. Findings

Introduction: The inspectors identified a green NCV for an inadequate procedure to mitigate the probable maximum flood. Conflicts between different sections of the procedure, conflicts between steps within one section of the procedure, and a missing step would lead to a loss of decay heat removal or a loss of RCS inventory for a unit in a refueling outage.

Description: Appendix 2.4A of the UFSAR described the flood protection plan for the Sequoyah units. With one or both units in a refueling outage, the prescribed method of cooling the fuel in the core and the spent fuel pit would be to fill the refueling cavity with borated water, cross-connect the spent fuel pit cooling system with the RHR system, and supply both the spent fuel pit and RHR heat exchangers with cooling from the ERCW System. The spent fuel cooling pumps would then be aligned to take suction from the spent fuel pit, discharge through the spent fuel pit heat exchangers, through the RHR heat exchangers, into all four cold legs, through the core, and into the refueling cavity. This would result in a water level differential with sufficient driving head to assure flow through the fuel transfer tube back to the spent fuel pit. The necessary pipe connections would be made with prestaged spool pieces.

Procedure AOP-N.03, Flooding, Revision 22, contained the instructions for mitigating a flood. This procedure contained two methods for use, closed mode and open mode, depending on whether or not a unit was in operation or refueling. These methods were further divided based on RCS temperature, for closed mode, or reactor vessel head status, for open mode. In reviewing the section for open mode cooling with the head off the inspectors discovered three problems. First, there was a conflict between the open mode section and the section for closed mode cooling with RCS temperature greater than 340EF. Both of these sections provided instructions to supply ERCW flow to the CCS system, however, the section for closed mode cooling isolated cooling flow to all RHR heat exchangers on both units. This would leave a unit in refuel without any cooling to the RHR heat exchangers. Secondly, the AOP open mode section contained two steps for installing the spool piece between the spent fuel system and the RHR system. The first step was contained in a Stage II flood preparation section and directed spool piece installation via a maintenance procedure. The second step was contained in the open mode section and directed installation via the same maintenance procedure. However, this step provided directions for isolating and draining the RHR piping before removing the blank flange to install the spool piece, which the maintenance procedure did not provide. Therefore, attempting to install the spool piece using the maintenance procedure as directed by the first step would open a path for RCS water to spill out and result in a loss of inventory. Thirdly, the open mode section did not contain any instructions to fill the refueling cavity and open the fuel transfer tube. It would be possible for a unit to be in refuel with the head off and not have the refueling cavity filled, therefore, the inspectors considered this to be a critical step.

Analysis: This finding is more than minor because if the procedure problems were left uncorrected they would result in a loss of RHR cooling or a loss of RCS inventory on a unit in refueling while preparing for flood conditions. Therefore, the finding affected the mitigating system cornerstone. Because the finding related to a loss of RHR or a loss of RCS inventory while shutdown, the inspectors evaluated it against Manual Chapter 0609 Appendix G, Shutdown Operations Significance Determination Process (SDP). For Phase 1 of the SDP the inspectors used Checklists 3 and 4, both of which directed a Phase 2 analysis because of the increased likelihood of a loss of decay heat removal or RCS inventory. For the Phase 2 analysis, the inspectors assumed the early time window (before removing fuel) with RCS water level equal to the reactor vessel flange, gave credit for recovery of RHR (due to ease of valve manipulations), and considered the time to RWST depletion to be greater than 10 hours (<500 gpm needed to maintain inventory). For flood frequency the inspectors used a bounding value of once per 50 years based on discussions with the regional Senior Reactor Analyst and UFSAR Section 2.4.3, which stated that the maximum flood would occur from probable maximum precipitation augmented by a break at the Watts Bar Dam. Additionally, the inspectors assumed that a unit was susceptible to the finding for 13 days of each outage based on the typical time from unit shutdown until the refueling cavity is flooded. From this, the inspectors completed the Phase 2 worksheets for a loss of RHR and loss of inventory with the RCS vented and determined the potential core damage frequency to be on the order of E-8/year. This is considered to be of very low safety significance (Green).

Enforcement: TS 6.8.1a requires that written procedures be established, implemented, and maintained covering activities in Regulatory Guide 1.33, Revision 2, Appendix A. Paragraph 6.w of Appendix A calls for procedures to combat acts of nature, such as a flood. Contrary to this, as of June 16, 2005, Procedure AOP-N.03 was inadequately maintained to mitigate a probable maximum flood. Two step conflicts within the procedure and one missing step would result in a loss of RHR or a loss of RCS inventory for a unit in refuel if the probable maximum flood were to occur. Because this violation was determined to be of very low safety significance (Green), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000327,05000328/2005003-01, Inadequate Procedure for Flood Mitigation. This violation is in the licensee’s corrective action program as PER 84376.

1R07 Biennial Heat Sink Performance

a. Inspection Scope

The inspectors reviewed inspection records, test results, maintenance work orders, and other documentation to ensure that heat exchanger (HX) deficiencies that could mask or degrade performance were identified and corrected. The test procedures and records were also reviewed to verify that these were consistent with Generic Letter (GL) 89-13 licensee commitments, and industry guidelines. Risk significant heat exchangers reviewed included the Component Cooling System (CCS) HXs, Containment Spray (CS) HXs, and the Diesel Generator coolers.

The inspectors reviewed HX inspection and cleaning work instructions, work maintenance history, tube plugging limits, and completed inspection records for all the safety related HXs selected. In addition, the inspectors reviewed Eddy Current Test (ECT) records for the CS HXs. These documents were reviewed to verify inspection methods were consistent with industry standards, to verify HX design margins were being maintained, and to verify performance of the HXs under the current maintenance frequency was adequate.

The inspectors also reviewed the general health of the Essential Raw Water Cooling (ERCW) system via review of design basis documents, system health reports, chlorination treatment trending and performance reports, and discussions with the ERCW system engineer. In addition, component health was verified via review of ERCW pump data trending such as differential pressure and flow rate, and AFW emergency makeup line flushes and valve stroke time test trends. These documents were reviewed to verify design bases were being maintained and to verify adequate ERCW system performance under current preventive maintenance, chemical treatment and inspection frequencies.

The inspectors also verified ERCW system corrosion and degradation were being monitored and addressed via review of corrosion control program procedures, ERCW pipe replacement and material condition action plans, and ERCW intake wells and traveling screen inspections. ERCW header flushes, system flow balance, and flow monitoring during molluscicide injections were also reviewed to verify design flow conditions to components were maintained and verified.

Problem evaluation reports (PERs) were reviewed for potential common cause problems and problems which could affect system performance to confirm that the licensee was entering problems into the corrective action program and initiating appropriate corrective actions. These PERs included actions regarding ERCW system wall thinning issues and component flow degradation issues including clams, and silt found in heat exchangers, and thermal relief valve corrosion nodule blockage. In addition, the inspectors conducted a walk down of all selected HXs and major components for the ERCW system to assess general material condition and to identify any degraded conditions of selected components.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems ISI

a. Inspection Scope

From May 4-6 and 11-13, 2005, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Unit 2. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of inspection procedure 71111.08, Inservice Inspection Activities, based upon the ISI activities available for review during the onsite inspection period.

The inspectors conducted an on-site review of the following types of nondestructive examination activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspectors observed and performed record review of the following examinations;

Ultrasonic Examination (UT):

- Weld # FDF-021A, Pipe to Elbow
- Weld # FDF-022, Pipe to Nozzle

Penetrant Examination (PT):

- Pipe to Valve Weld Number# 2-CM-196G
- Stainless Steel Pipe to Stainless Steel Valve, Weld # 2-CM-196-E & 2-CM-196-F
- 3/4" Pipe to 3/4" Valve, Weld # 2-CM-196-E C1 R0
- Pressurizer Nozzle Weld # RCW-25-SE, Vessel Nozzle to Vessel
- Pressurizer Nozzle Weld # RCF-42, Safe End to Elbow
- Pressurizer Nozzle Weld # RCW-26-SE, Vessel Nozzle to Vessel
- Pressurizer Nozzle Weld # RCF-36, Safe End to Elbow

Visual Examination (VT):

- 3 Pressurizer Safety Nozzles
- 1 Pressurizer Relief Nozzle
- 1 Spray Valve Nozzle

The inspectors also reviewed examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. The inspectors reviewed pressure boundary welds for Code Class 1 or 2 systems which were completed during the previous refueling outage, to verify that the welding acceptance and pre-service examinations. In specific:

- Pipe to Valve Weld Number# 2-CM-196G
- Stainless Steel Pipe to Stainless Steel Valve, Weld # 2-CM-196-E & 2-CM-196-F
- 3/4"Pipe to 3/4" Valve, Weld # 2-CM-196-E C1 R0

The inspectors performed a review of piping system 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. Additionally, 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. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.2 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From May 4-6 and 11-13, the inspectors reviewed the Unit 2 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.

The inspectors conducted an on-site record review and direct observation of the BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. In particular, the inspectors 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 performed observations and record reviews of the visual examinations, reviewed the visual examination procedures and examination records for the BACC examination conducted.

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

- 03-016523-00, lower reactor vessel head incore instrumentation leakage
- 05-774457-000, dried boron found on top of ductwork
- 05-774447-000, dried boron found on top of 2A-A LCCU03-004395-000, Accumulator #3 expansion joint downstream of piping has evidence of reddish brown boron

b. Findings

No findings of significance were identified.

.3 Steam Generator (SG) Tube ISI

a. Inspection Scope

The inspectors reviewed the licensee's scan plan, procedures, and selected inspection records for the eddy current examination (ET) for the Sequoyah Unit 2 Steam Generators (SGs). The records were compared to the Technical Specifications (TS), License Amendments and applicable industry established performance criteria to verify compliance. Qualification and certification records for examiners, equipment and procedures for the eddy current examination activities were reviewed. Available bobbin, rotating coil, and intelligent array inspection ET data was reviewed to evaluate the adequacy of completed data analysis. Additionally, the inspectors reviewed four Problem Evaluation Report (PERs) associated with the SG examinations.

c. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed just-in-time simulator training on April 20, 2005. The training involved real-time practice in the control room while collapsing the pressurizer bubble, establishing solid water plant pressure control, initiating pressurizer cooldown, pressurizer draining, and plant response to the loss of an operating reactor coolant pump followed by loss of an operating RHR pump. Inspectors also observed operators practice RCS level draining to reduced inventory, partial drain down, mid-loop conditions following core reload. The inspectors observed crew performance in terms of appropriate pre-evolution briefing; communications; ability to take timely and proper actions; prioritizing, interpreting and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; 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; and group dynamics involved in crew performance. The inspectors also reviewed simulator fidelity to verify that differences between Unit 1 and the simulator were appropriately addressed. 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 CFR 50.65(a)(1) or (a)(2); 8) appropriateness of performance criteria for 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.

- Replacement of Unit 2 RWST Level Instrument Enclosures
- Problems with Shutdown Board Room and Electric Board Room Chillers

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following five activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. 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 6, and Instruction 0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 8. Documents reviewed are listed in the attachment.

- Removal of Diesel Generator 2B from Service for Battery Work
- Use of SI Pump 2B as an Outage Makeup Path

- Unit 2 ERCW ESF Header Outage work on 2-FCV-67-127 Resulting in EGTS “A” Train Inoperable
- Weld Repair of Turbine Driven AFW Valve Steam Leak
- Diesel Generator 1A Outage

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 76181, Low Flow in RHR Pump 1A Room Cooler
- PER 81669, Cut in U2 Divider Barrier Seal
- PER 81532, U2 Containment Debris
- PER 79182, Snubbers Inappropriately Tied Off to Unit 1 AFW Suction Piping
- PER 80602, Gas in Discharge of SI Pump 2A

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the seven 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.

- WO 05-774214-000, Troubleshoot and Repair Unit 2 Charging Flow Controller
- WO 04-772035-000, Inspect Valve 2-FCV-30-57 for Possible Causes of LLRT Failure
- WO 03-017199-002, Instrument Support to Replace RWST Level Transmitter Enclosures
- WO 05-774432-000, U1 Turbine Driven AFW Stop Valve Steam Leak Repair
- WO 04-78829-002, Replace Diesel Generator 1A 125VDC Breakers and Panel
- WO 05-775646-00, 2-LCV-3-148, Loop 3 AFW Level Control Valve Limit Switch Repairs
- WO 05-777325-000, Investigate Cause of Smoke From Diesel Generator Field Flash Resistors

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

For the Unit 2 refueling outage that began on April 25, 2005, 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 2 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between April 25, 2005, and May 29, 2005, the inspectors performed inspections and reviews of the outage activities below. Documents reviewed during the inspection 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 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 reactor coolant system (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 safety injection pumps and injection valves 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 give accurate information, the inspectors reviewed the installation of the Mansell level monitoring system. Specifically, the inspectors discussed the Mansell with system engineers, 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 drain down, verified that operators properly set level alarms to procedurally required setpoints, and verified that the system consistently tracked while lowering RCS level to reduced inventory conditions. The Inspectors also evaluated the chain of events preceding a loss of approximately 9 inches (about 10,000 gallons) from the spent fuel pit while draining the fuel transfer canal.

- Refueling Activities. The inspectors observed fuel movement from the main control room, at the spent fuel pit, 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. In addition, the inspectors reviewed Engineering Document Change E21902A, Changes to Midloop Design Information Calculation, to verify that the calculated required containment closure times were based on time to core boil and containment environmental conditions, and that any open penetrations could be closed within the required time.

- 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 and observed operators in the main control room 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 reactor criticality.

b. Findings.

Introduction: A Green self-revealing NCV was identified for operators failing to follow plant procedures prior to and during draining of the fuel transfer canal. Leakage past the spent fuel pit gate seal resulted in inadvertently transferring approximately 10,000 gallons of spent fuel pit inventory to the refueling water storage tank.

Description: On May 23, 2005, at 0545, following Unit 2 refueling activities, licensee personnel began pumping down the fuel transfer canal. At 0635, Unit 1 operators received a Spent Fuel Pit High-Low Level alarm, referred to the appropriate annunciator response procedure, and dispatched an operator to determine the level. Only one of the Unit 1 control room operators was aware of the draining operation which had been discussed during watch station turnover and he did not immediately connect the low level alarm with this operation. At 0650, an auxiliary operator noted that spent fuel pit level was slightly below the alarm setpoint and informed Unit 1 control room personnel. He was then directed to report to the control room for a shift turnover brief and to prepare for spent fuel pit makeup. Operators believed the probable cause of the low level alarm was due to evaporative losses based on experience with a recently off-loaded core and concluded rapid response was not critical. Unit 1 personnel initiated makeup to the spent fuel pit at 0800 and at 0807 an operator assigned to monitor pool level informed control room personnel that level was approximately 7 inches below the previous level, well below that anticipated. At 0815, Unit 2 operators were contacted to determine if transfer canal pumpdown was in progress and then were requested to secure it. Pumpdown ceased at 0830. A refueling SRO proceeded to the spent fuel pit along with support personnel and determined that a fitting in the air line to inflate the spent fuel pit gate seal was not properly engaged. Once the fitting was properly engaged and the gate seal inflated at 0856, the loss of spent fuel pit inventory to the transfer canal stopped. Based on the difference between the lowest observed level and the low-level alarm setpoint, pool level dropped about 9 inches, equating to approximately 10,000 gallons.

Subsequent investigation showed that the gate had been reinstalled on May 18 per Procedure MI-1.2.11, Spent Fuel Pool to Transfer Canal Gate Removal and Installation, Revision 3, which included steps to connect air to the spent fuel pit gate seal and a visual check that the seal has inflated. Draining the fuel transfer canal was performed per procedure 0-SO-78-1, Spent Fuel Pit Coolant System, Revision 23. This procedure also included steps to verify that the seal was inflated and that spent fuel pit level was not decreasing after pumpdown was commenced. The annunciator response procedure, 1-AR-M6-D, Auxiliary Systems, Revision 27, identified spent fuel pit gate seal leakage as one of the probable causes of the low level alarm and directed operators to evaluate evolutions in progress that affect spent fuel pit level. Inadequate communications between operators of both units and conducting shift turnover briefings during the alarm condition contributed to the event.

Analysis: This finding was more than minor because it affected the Human Performance attribute of the Barrier Integrity cornerstone, in that, operators failed to adhere to procedures while changing plant configurations and failed to respond appropriately to valid alarms, resulting in a loss of spent fuel pit inventory. Additionally, if left uncorrected, it would become a more significant safety concern. Since the condition only represented a small degradation of the radiological barrier function provided by the spent fuel pit, the finding was determined to be of very low safety significance (Green). The cause of the finding is related to the cross-cutting area of human performance. The issue is in the licensee's corrective action program in PERs 83242 and 83124.

Enforcement: TS 6.8.1a requires that procedures be implemented covering activities in Regulatory Guide 1.33, Revision 2, Appendix A. Paragraph 3.h of Appendix A requires that written procedures be established, implemented, and maintained covering activities which include spent fuel pit operations. Paragraph 5 of Appendix A covers alarm conditions. Licensee Procedure MI-1.2.11, directed reconnecting air to the spent fuel pit gate seal and checking that the seal is inflated. Procedure 0-SO-78-1 required a verification that the inflatable seal was inflated and that pool level was not decreasing during transfer canal dewatering. Annunciator Response Procedure 1-AR-M6-D (D-3) included gate seal leakage as a probable cause of a spent pool low level alarm and required an evaluation of evolutions in progress that affect pool level and to check the air supply to the gate inflatable seal for possibly causing the low level. Contrary to this, on May 18, 2005, maintenance workers failed to properly reconnect the spent fuel pit gate inflatable seal air supply and check that it was inflated. Additionally on May 23, 2005, operators failed to verify that the seal was inflated and to adequately verify that pool level was not decreasing while dewatering the transfer canal. Operators also did not expeditiously carry out the annunciator response actions by failing to evaluate evolutions in progress that affect spent fuel pit level. These resulted in a loss of approximately 10,000 gallons of spent fuel pit inventory over a 3-hour period. Because this violation was determined to be of very low safety significance (Green), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000327,05000328/2005003-02, Failure to Follow Procedures Resulting in an Inadvertent Transfer of 10,000 Gallons of Spent Fuel Pool Inventory. This violation is in the licensee's corrective action program as PERs 83242 and 83124.

1R22 Surveillance Testing

a. Inspection Scope

For the seven surveillance tests identified below, by witnessing testing and/or reviewing the test data, the inspectors verified that the systems, structures, and components(SSCs) involved in these tests satisfied the requirements described in the TS surveillance requirements, the UFSAR, applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions. Documents reviewed are listed in the attachment. Those tests included the following:

- SI-90.82, Reactor Trip Instrumentation Monthly Functional Test (SSPS) - Unit 2, B Train, Revision 40
- 2-SI-SXP-072-201.A, Containment Spray Pump 2A-A Performance Test, Revision 8*
- 0-SI-IFT-099-093.1, Functional Tests of Turbine Auto Stop Oil Dump and Throttle Valves Reactor Trips, Revision 17
- 2-SI-ICC-068--29B.2, Channel Calibration of Loop 2 Reactor Coolant Flow Channel F-68-29B (F-425) Protection Set II Rack 5, Revision 8
- 0-SI-MIN-061-109.0, Ice Condenser Intermediate and Lower Inlet Doors and Vent Curtains, Revision 2**
- 0-SI-MIN-061-106.0, Ice Condenser Flow Passage Inspection, Revision 2**
- 2-SI-ICC-077-410.0/411.0, Channel Calibration of Reactor Building Auxiliary Floor and Equipment Drain Sump Level, Revisions 7***

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

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors evaluated the adequacy of licensee methods for testing the alert and notification system in accordance with NRC Inspection Procedure 71114, Attachment 02, Alert and Notification System (ANS) Testing. The applicable planning standard 10 CFR Part 50.47(b)(5) and its related 10 CFR Part 50, Appendix E, Section IV.D requirements were used as reference criteria.

The criteria contained in NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Revision 2 and Regulatory Guide 1.101 were also used as references.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation

a. Inspection Scope

The inspectors reviewed the ERO augmentation staffing requirements and the process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The results of the February 1, 2005, unannounced off-hours augmentation drill were reviewed. The inspectors conducted a review of the backup notification systems. The qualification records of key position ERO personnel were reviewed to ensure ERO qualifications were current. A sample of problems identified from augmentation drills or system tests performed since the last inspection were reviewed to assess the effectiveness of corrective actions. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, Emergency Response Organization (ERO) Augmentation Testing. The applicable planning standard, 10 CFR 50.47(b)(2) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspectors evaluated the associated 10 CFR 50.54(q) reviews associated with non-administrative emergency plan changes, implementing procedures changes, and EAL changes. The revisions covered the period from June 2004 to June 2005.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 01, Emergency Action Level and Emergency Plan Changes. The applicable planning standard, 10 CFR 50.47(b)(4) and its related 10 CFR 50, Appendix E requirements were used as reference criteria. The criteria contained in

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Revision 2 and Regulatory Guide 1.101 were also used as references.
The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors reviewed the corrective actions identified through the EP program to determine the significance of the issues and to determine if repeat problems were occurring. The facility's self-assessments and audits were reviewed to assess the licensee's ability to be self-critical, thus avoiding complacency and degradation of their EP program. In addition, inspectors review licensee's self-assessments and audits to assess the completeness and effectiveness of all EP-related corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, Correction of Emergency Preparedness Weaknesses and Deficiencies. The applicable planning standard, 10 CFR 50.47(b)(14) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

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

During the inspection, radiological controls for ongoing refueling activities were observed and discussed. Reviewed tasks included steam generator nozzle dam preparation and testing, removal of a motor operated valve in the Unit 2 (U-2) Auxiliary Feedwater Pump Room, and installation of drain lines to the ice melt tanks. In addition, licensee controls for selected tasks scheduled and on-going during the current refueling outage were assessed. The evaluations included, as applicable, Radiation Work Permit (RWP) details; use and placement of dosimetry and air sampling equipment; electronic dosimeter set-points, and monitoring and assessment of worker dose from direct radiation and airborne radioactivity source terms. Effectiveness of established controls was assessed against area radiation and contamination survey results, and occupational doses received. Physical and administrative controls and their implementation for locked high radiation areas (LHRA) and very high radiation areas (VHRA) were evaluated through discussions with cognizant licensee representatives, direct field observations, and record reviews.

Occupational workers' adherence to selected RWPs and Health Physics Technician proficiency in providing job coverage were evaluated through direct observations of staff performance during job coverage and routine surveillance activities, review of selected exposure records, and interviews with cognizant licensee staff. Radiological postings and physical controls for access to designated high radiation (HRA) and LHRA locations within the U-2 Containment, Auxiliary Building, and Refuel Floor areas were evaluated during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for selected Auxiliary Building locations. Occupational exposures associated with direct radiation and potential radioactive material intakes for were reviewed and discussed with cognizant licensee representatives.

RP program activities were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; Updated Final Safety Analysis Report (UFSAR) details in Section 12, Radiation Protection; Technical Specifications (TS) Section 6.11, High Radiation Area; and approved licensee procedures. Licensee procedures, guidance documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report Attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Standard Programs and Processes (SPP) - 3.1, Corrective Action Program, Revision 8. Licensee self-assessments and PER documents related to access control that were reviewed and evaluated in detail during inspection of this program area are identified in Section 2OS1 of the report Attachment.

b. Findings

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 U2 Cycle 13 outage was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included removal of the manway covers and installation of the steam generator primary manway inserts. Also, incorporation of planning, established work controls, expected dose rates and dose expenditure into the ALARA pre-job briefings and RWPs for those activities were reviewed. The inspectors directly observed performance of 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 300 written for the current outage. Electronic dosimeters (EDs) included administrative limits (warning to employee and manager, followed by ED lockout, if designated limits exceeded) on individual worker exposure. The inspectors attended a meeting of the licensee's ALARA Committee on May 19, 2005, which discussed revisions to outage goals for specific jobs and departments.

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. The inspectors reviewed the licensee's source-term control strategy, noting the employment of new techniques for Sequoyah. These included the use of macroporous resin in the CVCS and cleanup of the RCS with hydrogen peroxide addition while the reactor coolant pumps were running. The inspectors discussed the results that were achieved with these methods during the U1C13 outage (overall, a 15% dose reduction compared with the previous outage).

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 April 2003 to May 2005 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 CAP documents listed in Section 2OS2 of the report Attachment that are related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with SPP - 3.1, Corrective Action Program, Rev. 7.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

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

Solid and liquid radwaste equipment was inspected for material condition, configuration compliance with the UFSAR, and consistency with Process Control Program (PCP) requirements. The inspectors reviewed the status of non-operational or abandoned in place radwaste equipment. The inspectors reviewed the licensee's administrative and physical controls of non-operational or abandoned in place radwaste equipment to prevent unmonitored releases, impact to operating systems or contribute to unnecessary personnel exposure. Inspected equipment included liquid radwaste hold-up tanks; resin transfer piping; filters, and elements of the Mobile Demineralization System. The inspectors discussed system changes, component function, and equipment operability with licensee staff. In addition, procedural guidance for resin transfer was evaluated and compared with current equipment configuration. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Licensee radionuclide characterizations for selected waste streams were reviewed and discussed with radwaste staff. For selected resin, charcoal, radwaste filters, and dry active waste (DAW), the inspectors evaluated analyses for hard-to-detect nuclides and appropriate use of scaling factors. Comparison results between licensee waste stream characterization data and outside laboratory data were reviewed for the period May 2003 to December 2004. For selected shipment records, waste classification calculations were performed and the methodology used for resin waste stream mixing and concentration averaging was evaluated.

The inspectors also interviewed cognizant radwaste staff and reviewed procedural guidance to evaluate the licensee's program for monitoring changing operational parameters.

Radwaste processing activities were reviewed for consistency with the licensee's PCP, Rev. 3, dated November 4, 2002; and UFSAR, Chapter 11, Amendment 16, dated May 2001. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position on Waste Classification and Waste Form, 1983.

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

The inspectors directly observed transportation activities including shipment packaging, surveying, blocking and bracing, vehicle placarding, vehicle checks, emergency instructions, preparation of disposal manifest, and the provision of shipping papers and special instructions to drivers. Specifically, inspectors observed two incoming shipments and two outgoing shipments. Incoming shipments contained laundry (containing byproduct materials) and fissile materials (fuel). The laundry shipment was shipped as exclusive use, radioactive-low specific activity (LSA-II), and the fuel shipment was shipped as Yellow-II, fissile. The outgoing shipments contained laundry (containing byproduct materials) and empty fuel containers. The laundry shipment was shipped as exclusive use, radioactive-low specific activity (LSA-II).

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

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

Problem Identification and Resolution The inspectors reviewed the licensee's events reports and self assessment related to radioactive material processing and transportation areas, to determine if problems were identified and entered in the system for resolution. Specifically, the inspectors reviewed PER reports and interviewed cognizant licensee personnel to determine if problems were identified, properly characterized, prioritized, evaluated and corrected. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure SPP - 3.1, Corrective Action Program, Rev. 8. Reviewed documents are listed in Section 2PS2 of the report Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data submitted to the NRC for the performance indicators (PIs) listed below for the period from July 1, 2004, through March 31, 2005. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Rev. 2, were used to verify the basis in report for each data element.

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

Public Radiation Safety Cornerstone The inspectors reviewed and evaluated selected radiological liquid and gaseous effluent release data, abnormal release results, cumulative and projected doses to the public, and selected PER records for the period of July 1, 2004, through March 31, 2005. Documents reviewed are listed in section 4OA1 of the report Attachment.

Emergency Preparedness Cornerstone The inspectors reviewed the licensee’s procedure for developing the data for the EP PIs which are: (1) Drill and Exercise Performance (DEP); (2) ERO Drill Participation; and (3) ANS Reliability. The inspectors examined data reported to the NRC for the period April 2004 to March 2005. Procedural guidance for reporting PI information and records used by the licensee to identify potential PI occurrences were also reviewed. The inspectors verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspectors reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspectors verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee’s records of periodic system tests.

The inspection was conducted in accordance with NRC Inspection Procedure 71151, "Performance Indicator Verification." The applicable planning standard, 10 CFR 50.9 and NEI 99-02, Revision 3, "Regulatory Assessment Performance Indicator Guidelines," were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

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 Gas Intrusion Problems on Valves 1(2)-FCV-63-8

a. Inspection Scope

Following startup from the last two refueling outages on Unit 1 (spring 2003 & fall 2004) and the intervening Unit 2 outage (fall 2003) the licensee discovered gas underneath Valves 1(2)-FCV-63-8. This valve was located in a vertical section of pipe that supplied the charging pump suction from RHR system Train A, i.e. piggyback valve, and was situated such that any gas released from the suction piping to the charging pumps would collect there. The inspectors reviewed licensee actions to resolve this issue because the presence of gas in this piping has the potential to affect the ability of the charging pump to perform its ECCS function. The inspectors reviewed several PERs, interviewed engineering personnel, and observed several of the corrective actions for this issue. Documents reviewed are listed in the attachment.

b. Findings and Observations

No findings or violations of significance were identified; however, the inspectors noted that corrective actions put in place following the fall 2003 Unit 1 outage could have been implemented earlier and that the method for implementing one of them was weak, resulting in an incomplete implementation.

Following the spring 2003 Unit 1 outage the licensee identified 2.8 ft^3 of gas on the charging pump side of Valve 1-FCV-63-8. The licensee determined that this amount did not affect the operability of the charging pump, attributed the cause to elevated VCT pressure during startup, and specified corrective actions. Following the fall 2003 Unit 2 outage, the licensee identified a small amount of gas on the charging pump side of Valve 2-FCV-63-8 on two occasions. The licensee again determined that this amount did not affect charging pump operability, but chose not to perform a cause evaluation or implement any new corrective actions. Following the fall 2004 Unit 1 outage, the licensee again found 2.8 ft^3 of gas underneath Valve 1-FCV-63-8. At this point, the licensee implemented further corrective actions to identify any gas underneath this valve at an earlier time. After reviewing the applicable PERs, the inspectors determined that the licensee could have implemented these corrective actions 12 months earlier. Finding gas under 2-FCV-63-8 following the fall 2003 Unit 2 outage presented an opportunity to implement further corrective actions at that time.

One of the additional corrective actions implemented after the fall 2004 Unit 1 outage was to check for gas underneath valves 1(2)- FCV-63-8 starting 24 hours after entering Mode 3 on startup following an outage and then every other day for seven days, a total of four checks. For the spring 2005, Unit 2 outage, the inspectors noted that this corrective action was implemented by adding four activities to the outage schedule. As a result, while four checks were made, only one was done within the seven days as specified in the PER. Because the outage schedule could be readily changed, the inspectors determined that the licensee was not effective in implementing the corrective actions for this issue.

.3 Annual Sample Review of Problems With Clams Discovered in Various Safety-Related Components

a. Inspection Scope

In September and October of 2004, the licensee experienced several occurrences of reduced ERCW flow in various room coolers and, upon inspection, discovered clams on the ERCW side of several safety-related chillers. The inspectors reviewed licensee actions to resolve this issue because the growth of clams in the service water system has the potential to affect core decay heat removal, ECCS pump performance, and the emergency diesel generators by degrading the ultimate heat sink. The inspectors reviewed PER 68880, Clams in Electric Board Room Chiller A, and associated PERs, interviewed engineering and chemistry personnel, and observed several of the corrective actions for this issue. Documents reviewed are listed in the attachment.

b. Findings and Observations

No findings or violations of significance were identified; however, the inspectors noted that corrective actions only minimally focused on correcting one of the identified causes. The inspectors considered this to be similar to corrective action problems associated with the Siemens breaker problems discussed in IRs 05000327,328/2004-004, 2004-010, 2005-007, and 2005-008.

One of the identified causes for the increased clams was the lack of a formal process to integrate the chemistry treatment program with engineering system alignments and flow requirements. The inspectors noted that the specified corrective actions for this cause were to increase raw water team meetings to once/month and to provide training to the various departments involved in clam control in the ERCW system. The inspectors determined that while these actions served to heighten awareness in the short term, and have been effective up through this inspection, they did not fully establish a formal process to integrate the chemistry and engineering aspects of clam control. The inspectors further concluded that the licensee was predominately relying on good coordination among engineers and technicians in the various departments in order to control clams. This was similar to the problems with Siemens breakers because process and organizational issues were involved there as well and the licensee did not fully address them until the problems began affecting equipment operability.

.4 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 January 2005 through June 2005, 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 October 2004 through December 2004 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 January 2005 through June 2005.

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 licensee had not yet performed a trend review for the period from January 2005 through June 2005; however, the inspectors evaluated the methodology for the trending actually performed by the licensee 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.

Two equipment issues emerged in 2005 that have the potential to be more safety significant and are being closely monitored by the resident inspectors. On January 31, 2005, Unit 1 RCS activity, as measured by dose-equivalent iodine, increased one order of magnitude from approximately 1E-3 micro curies/gram to 1E-2 micro curies/gram.

While this was well below the TS limit of 0.35 micro curies/gram, the licensee determined that a small leak had developed in one rod of one fuel assembly on its first cycle. The licensee immediately increased the RCS sample frequency, organized a fuel integrity team, and began monitoring the concentration of various Iodine and Xenon isotopes in order to detect any changes that would indicate a worsening fuel leak. The trend has remained stable since January at or below approximately 1E-2 micro curies/gram.

After startup following the Unit 1 refueling outage in November 2004, pressure in the RHR discharge headers began increasing. The pressurization rate was enough that it required operators to vent the RHR headers five to six times per shift in order to prevent over pressurization of the piping. The licensee determined that check valve 1-63-634 was leaking in the reverse direction and allowing water from Cold Leg Accumulator 3 into the RHR headers causing them to pressurize. The leak started at approximately 0.5 gallons/hour in November 2004 but has steadily increased to approximately 13 gallons/hour in June 2005. While this leak rate has remained well below the TS limit of 3 gallons/minute, it has presented a large burden to the operators and has required frequent starts of the Unit 1 SI pumps to fill the accumulator. The licensee installed a temporary continuous vent on the RHR headers to reduce the operator burden and has been in the process of developing another temporary modification to keep the accumulator filled and reduce the number of starts on the SI pumps.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000327/2005-001-00, Unit 1 Automatic Reactor Trip Following Loss of Turbine Auto Stop Oil Pressure

a. Inspection Scope

On April 9, 2005, Unit 1 tripped on low auto stop oil pressure. In addition to initially responding to the event and verifying plant and operator response, the inspectors reviewed the LER and PER 80518, Unit 1 Reactor Tripped - Initial Indication Auto Stop Oil System Failure. The PER documented this event in the licensee corrective action program. The inspectors verified that the cause of the reactor trip was identified and that corrective actions were appropriate. Inspectors also verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required. Documents reviewed are listed in the attachment.

b. Findings

Introduction: The inspectors identified a Green finding for a self-revealing failure to implement effective corrective actions for control oil port misalignment between the turbine front pedestal and the turbine protective trip block.

Description: The licensee identified the cause of the event to be improper machining of the turbine front pedestal oil ports during original manufacturing. This resulted in oil port misalignment and oil seepage between the turbine protective trip block and the turbine governor pedestal. In 1998, the licensee requested turbine vendor concurrence to use a gasket seal instead of the original O-ring seal to correct the oil seepage problem. The vendor concurred, stipulating that the gasket material must be compatible with turbine control oil and suggested BUNA-N material. However, in April 2003, the licensee replaced the BUNA-N gasket with 1/16" red rubber sheet gasket material. This change was performed with verbal concurrence of the vendor but without a material compatibility check prior to installation. Red rubber gasket material was found not to be fully compatible with the turbine oil. It therefore degraded and eventually failed. This failure provided an oil flow path between the auto stop oil supply and drain ports, resulting in a loss of auto stop oil system pressure and subsequent automatic turbine and reactor trip.

Analysis: The finding was more than minor because it affected the design control attribute of the initiating event cornerstone and upset plant stability by causing a reactor trip. While the finding resulted in an actual trip, the inspectors determined that it did not contribute to the likelihood of a primary or secondary system LOCA initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. Thus, the finding was considered to be of very low safety significance (Green). This issue is in the licensee corrective action program as PER 80518. The cause of the finding was considered to have aspects of the Problem Identification and Resolution cross-cutting area.

Enforcement. Because the affected equipment was non-safety related, no violation of regulatory requirements occurred. Therefore, this finding is identified as FIN 05000327/2005003-03, Gasket Failure on Turbine Trip Block Resulted in Reactor Trip. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000328/2005-001-00, Unit 2 Reactor Trip Following Closure of Main Feedwater Upon Inadvertent Opening of Control Breakers

a. Inspection Scope

On February 23, 2005, Unit 2 tripped on low-low steam generator levels following a loss of control power to all four feedwater regulating valves. Control power was lost when maintenance personnel inadvertently dropped a breaker board panel cover, opening two breakers. One of the breakers controlled a fuse column providing power to the valves. Inspectors reviewed the LER and PER 77234, Inadvertent Operation of Breaker 2-BKRC-250-KH1/213-B to the Open Position and Subsequent Reactor Trip, which documented this event in the licensee corrective action program, to verify that the cause of the reactor trip was identified and that corrective actions were appropriate. Inspectors also verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required. Documents reviewed are listed in the attachment.

b. Findings:

Introduction. The inspectors identified an NCV for a self-revealing failure to have adequate work procedures for testing molded case circuit breakers associated with the 125VDC battery board supply.

Description. On February 23, 2005, maintenance technicians removed the cover from 125VDC Vital Battery Board IV in order to test Breaker 214. While replacing the panel cover, it slipped from the grasp of the maintenance technician and fell, inadvertently opening two adjacent breakers. One of them, Breaker 213, supplied a fuse panel providing control power to the Unit 2 Main Feedwater Regulating valves.

Upon loss of control power, the valves went shut, resulting in lowering steam generator levels, ultimately resulting in an automatic reactor trip.

Operations and maintenance personnel involved were aware of the proximity and potential for the adjacent breakers to open during the work and had discussed this during the pre-job brief. The Unit Supervisor stressed that great care should be exercised when removing and replacing the panel cover. They were also aware that inadvertently opening Breaker 213 would likely result in a significant plant transient. The procedure in use, 0-MI-EBR-317-010.0, Testing of Molded Case Circuit Breakers, Revision 33, did have provisions for installing threaded rods in place of the panel mounting bolts which would maintain positive positional control while removing and reinstalling panel covers. However, these provisions were noted as applicable for 120 Vac Vital Instrument Power Boards only. The steps were therefore marked as “Not Applicable” for the 125VDC breaker work.

Analysis. The finding was more than minor because it affected the procedure quality attribute of the initiating event cornerstone and upset plant stability by causing a reactor trip. While the finding resulted in an actual trip, the inspectors determined that it did not contribute to the likelihood of a primary or secondary system LOCA initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. Thus, the finding was considered to be of very low safety significance (Green). This issue is in the licensee corrective action program as PER 77234.

Enforcement. TS 6.8.1a requires that procedures be implemented covering the activities in Regulatory Guide 1.33, Revision 2, Appendix A. Paragraph 9.a of Appendix A recommended that maintenance be properly preplanned and performed in accordance with written procedures appropriate for the circumstances. Licensee Procedures SPP-2.2, Administration of Site Technical Procedures, Revision 13, and MMDP-1, Maintenance Management System, Revision 7S1, require that all procedural steps that can result in “Intolerable Consequences” to the plant be identified in the procedure as Critical Steps. “Intolerable Consequences” included turbine or reactor trips. Additionally, SPP-2.2 Procedure Verification Review Checklist, Step 32, directs that appropriate controls to ensure that unit operation is not threatened are to be included. Contrary to the above, on February 23, 2005, the licensee failed to implement these requirements by performing breaker testing per 0-MI-EBR-317-010.0 without identifying breaker panel cover replacement as a critical step, even though operators and maintenance personnel were aware of the trip hazard. Additionally, appropriate controls to maintain positive positional control when removing and replacing the breaker panel (e.g. installing guide pins in the breaker panel) to ensure unit operation was not threatened were not included. Because this violation was determined to be of very low safety significance (Green) it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Manual and is identified as NCV 05000328/2005003-04, Failure to Identify Critical Steps in a Maintenance Procedure Resulted in a Reactor Trip. This violation is in the licensee's corrective action program as PER 77234. This LER is closed.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction (TI) 2515/163, Operational Readiness of Offsite Power

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff, as required by the TI 2515/163. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 17; Plant TS for offsite power systems; 10 CFR Part 50.63; 10 CFR Part 50.65(a)(4); and licensee procedures. Documents reviewed for this TI are listed in the attachment.

b. Findings and Observations

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/163 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis. This completes the Region II inspection requirements for this TI for the Sequoyah site.

.2 (Closed) NRC TI 2515/156, Offsite Power System Operational Readiness

This TI was discussed in IR 05000327,328/2004003 with data reported to the headquarters staff. This completes the Region II inspection requirements for this TI for the Sequoyah site.

.3 (Closed) TI 2515/161, Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

Through interviews with cognizant licensee representatives, the inspectors determined that had Sequoyah undergone refueling/defueling activities during Calendar Year 2002 to the present; however, the licensee had not packaged and shipped irradiated Control Rod Drive mechanisms in DOT Specification Type A packages since that period of time.

b. Findings

No findings of significance were identified.

.4 Independent Spent Fuel Storage Installation (ISFSI) Radiological Controls.

a. Inspection Scope

The inspectors conducted independent gamma and neutron surveys of the ISFSI facility and compared the results to previous surveys. The inspectors also observed and evaluated implementation of radiological controls, including RWPs and postings, and discussed the controls with a HPT and HP supervisory staff. Radiological controls for loading the Holtec ISFSI casks were also reviewed and discussed.

Radiological control activities for ISFSI areas were evaluated against 10 CFR Part 20, 10 CFR Part 72, and applicable licensee procedures. Documents reviewed are listed in section 4OA5 of the report Attachment.

b. Findings

No findings of significance were identified.

.5 (Closed) TI 2515/160 (Unit 2 only), Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

The inspectors reviewed the licensee’s 60-day response to NRC Bulletin 2004-01, dated July 27, 2004. The inspectors verified that the licensee’s examinations conducted during the most recent Unit 2 refueling outage were consistent with the licensee’s response.

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

- RCW-25-SE
- RCW-26-SE
- RCW-27-SE
- RCW-28-SE
- Spray Nozzle

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

   1. Performed by qualified and knowledgeable personnel? The inspectors verified that the examination personnel were VT-1 and VT-2 qualified in accordance with the licensee written practice, and response to Bulletin 2004-01.

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

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

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

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

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

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

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

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

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

h. If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations? In accordance with the licensee's response, a BMV examination was conducted this outage, and a surface examination of Pressurizer Nozzle Weld # RCW-25-SE, Vessel Nozzle to Vessel, Pressurizer Nozzle Weld # RCF-42, Safe End to Elbow, Pressurizer Nozzle Weld # RCW-26-SE, Vessel Nozzle to Vessel, Pressurizer Nozzle Weld # RCF-36, Safe End to Elbow were performed.

I. Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system? Yes, the licensee performed surface examinations of the nozzles that had boron indications, as well as taking chemistry samples to date the boron found. The surface examinations showed no indications.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On July 12, 2005, the resident inspectors presented the inspection results to Mr. R. Douet 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.

.2 Regulatory Performance Meeting

A meeting was held on April 12, 2005 at the Sequoyah site between Mr. Randy Douet, Site Vice President of the Sequoyah Nuclear Plant, and Mr. Stephen Cahill, Chief, Reactor Projects Branch 6 of the NRC Region II office. The purpose of the meeting was to discuss a White Finding regarding the failure to promptly identify and correct binding problems with safety-related 6.9 kilovolt (kV) breakers in July 2004 and the results of the associated NRC Supplemental Inspection that was done March 28 - 30, 2005 at the Sequoyah site (see NRC Supplemental Inspection Report No. 05000327/2005008). The corrective action plan for the issue and the licensee's methodical plan to re-introduce the breakers back into the plant were discussed. The meeting constituted the Regulatory Performance Meeting for the finding. Both the Supplemental Inspection and the Regulatory Performance Meeting are required per the NRC Action Matrix (contained in NRC Manual Chapter 305, Operating Reactor Assessment Program) for a licensee in the Regulatory Response Column. As discussed in the NRC Annual Assessment Follow-up Letter dated March 10, 2005, Sequoyah Unit 1 was in the Regulatory Response Column due to the White Finding. At the conclusion of the meeting, all of the Action Matrix requirements for the 6.9 kV breaker White Finding were completed.

4OA7 <u>Licensee-Identified Violations</u>

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

- TS 3.8.1.1 Action b requires that each of the required independent offsite circuits be determined operable by performing SR 4.8.1.1.1.a within one hour of a diesel generator set becoming inoperable. Contrary to the above, on April 6, 2005, the licensee failed to perform the breaker alignment checks required by SR 4.8.1.1.1.a within one hour of EDG 2B becoming inoperable when they failed to recognize that the surveillance test was not completed satisfactorily. This was identified in the licensee’s corrective action program as PER 80424. This violation is of very low significance because the affected diesel generator remained capable of performing its function.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee personnel:

J. Bajraszewski, Licensing Engineer
R. Bruno, Training Manager
E. Camp, Steam Generator Engineer
T. Cornelius, Emergency Preparedness
H. Cothran, Steam Generator Manager
R. Douet, Site Vice President
R. Ford, Emergency Preparedness Manager
M. Gillman, Operations Manager
K. Jones, System Engineer Manager
Z. Kitts, Licensing Engineer
D. Kulisek, Plant Manager
B. Marks, Manager, Emergency Services
S. McCamy, Radiation Protection Health Physicist
P. Pace, Licensing and Industry Affairs Manager
K. Parker, Maintenance and Modifications Manager
R. Reynolds, Interim Security Manager
R. Richie, Chemical/Environmental Manager
R. Rogers, Engineering Manager
P. Sawyer, Radiation Protection Manager
J. Smith, Site Licensing Supervisor
C. Webber, TVA Eddy Current NDE Level III
J. Whitaker, ISI

NRC personnel:

R. Bernhard, Senior Reactor Analyst, Region II,
D. Pickett, Project Manager, Office of Nuclear Reactor Regulation
S. Shaeffer, Senior Project Engineer, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000327,05000328/2005003-01	NCV	Inadequate Procedure for Flood Mitigation (Section 1R06.2).
05000327,05000328/2005003-02	NCV	Failure to Follow Procedures Resulting in an Inadvertent Transfer of 10,000 Gallons of Spent Fuel Pool Inventory (Section 1R20).
05000327/2005003-03	FIN	Gasket Failure on Turbine Trip Block Resulted in Reactor Trip (Section 4OA3.1)
05000328/2005003-04	NCV	Failure to Identify Critical Steps in a Maintenance Procedure Resulted in a Reactor Trip (Section 4OA3.2).

Closed

05000327/2005-001-00	LER	Unit 1 Automatic Reactor Trip Following Loss of Turbine Auto Stop Oil (ASO) Pressure (Section 4OA3.1).
05000328/2005-001-00	LER	Unit 2 Reactor Trip Following Closure of Main Feedwater Upon Inadvertent Opening of Control Breakers (Section 4OA3.2).
05000327,328/2515/163	TI	Operational Readiness of Offsite Power (Section 4OA5.1).
05000327,328/2515/156	TI	Offsite Power System Operational Readiness (Section 4OA5.2).
05000327,328/2515/161	TI	Transportation of Reactor Control Rod Drives in Type A Packages (Section 4OA5.3).
05000328/2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) (Section 4OA5.5).

LIST OF DOCUMENTS REVIEWED

Section R04: Equipment Alignment

0-SO-65-1 Attachment 1, Emergency Gas Treatment System Air Cleanup and Annulus Vacuum Power Checklist, Change 2
0-SO-65-1 Attachment 2, Emergency Gas Treatment System Air Cleanup and Annulus Vacuum Valve Checklist, Change 3
1,2-47W803-2, Flow Diagram Auxiliary Feedwater, Revision 59
1-SO-3-2 Attachment 1, Auxiliary Feedwater System Power Checklist, Change 10

Section R06: Flood Protection Measures

0-PI-IFT-040-001.0, Functional Test of Auxiliary and Reactor Buildings Flood Alarms, Revision 3
AOP-N.03, Flooding, Revision 22
0-FP-MXX-000-005.0, Flood Preparation - Access to Spent Fuel Pit Pumps and Thermal Barrier Booster Pumps, Revision 2
0-FP-MXX-000-011.0, SFPC Pump Enclosure Caps, SFPCS Heat Exchangers, RCP Thermal Barrier Booster Pumps, and RHR Heat Exchanger Spool Pieces, Revision 6
0-FP-MXX-000-012.0, Flood Preparation Open Mode Reactor Cooling Spool Pieces and Flow Orifices, Revision 4
1,2-47W610-40-1, Mechanical Control Diagram, Station Drainage System, Revision 4
1,2-47W859-1, Mechanical Flow Diagram, Component Cooling System, Revision 50
-47W859-2, Mechanical Flow Diagram, Component Cooling System, Revision 30
2-47W859-3, Mechanical Flow Diagram, Component Cooling System, Revision 30
1,2-47W859-4, Mechanical Flow Diagram, Component Cooling System, Revision 20
1,2-47W855-1, Mechanical Flow Diagram, Fuel Pool Cooling and Cleanup System, Revision 41
PER 73053, Problems With AOP-N.03
WO 04-772050-000, Level Switch 1-LS-40-29 Failed to Actuate Alarm

Section R07 : Biennial Heat Sink Performance

Procedures

0-PI-CEM-000-460.4, Raw Water Quaternary Amine Treatment Monitoring, Rev. 10
0-TI-SXX-000-146.0, Program for Implementing NRC Generic Letter 89-13, Rev. 0
SPP-9.7, Corrosion Control Program, Rev. 9

Preventive Maintenance Work Instructions

030330000, ERCW Pump Well & Traveling Screen Inspections, Rev. 7
031651000, Incore Instrument Room Chiller Inspections, Rev. 3
041432002, Component Cooling System Heat Exchanger Inspections, Rev. 9

Completed Work Orders

01-003385-000, ERCW Pipe to the Turbine Driven AFW Pump Flush, completed 11/01
02-001000-000, CS 1B Heat Exchanger Eddy Current Test, completed 06/03
02-008347-000, EDG 1B2 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-008348-000, EDG 1B1 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-008685-000, EDG 2B1 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-008686-000, EDG 2B2 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-009155-000, CCS 0B1 HX Clam, MIC and Degradation Inspection, completed 05/03
02-009156-000, CCS 0B2 HX Clam, MIC and Degradation Inspection, completed 05/03
02-009318-000, EDG 2A2 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-009381-000, EDG 2A1 Cooler Clam, MIC and Degradation Inspection, completed 02/03
02-009839-000, EDG 1A2 Cooler Clam, MIC and Degradation Inspection, completed 01/03
02-009840-000, EDG 1A1 Cooler Clam, MIC and Degradation Inspection, completed 01/03
02-010453-000, CCS 1A2 HX Clam, MIC and Degradation Inspection, completed 05/03
02-011449-000, CCS 2A2 HX Clam, MIC and Degradation Inspection, completed 05/03
02-011450-000, CCS 2A1 HX Clam, MIC and Degradation Inspection, completed 05/03
02-011506-000, CS 2A HX Clam, MIC and Degradation Inspection, completed 09/03
02-013695-000, CS 2B Heat Exchanger Eddy Current Test, completed 12/03
03-015645-000, CCS 1A2 HX Clam, MIC and Degradation Inspection, completed 03/04
04-770767-000, CS 1A Heat Exchanger Eddy Current Test, completed 11/04
04-775330-000, EDG 2B1 Cooler Clam, MIC and Degradation Inspection, completed 02/05
04-775331-000, EDG 2B2 Cooler Clam, MIC and Degradation Inspection, completed 02/05
04-778030-000, CCS 1A2 HX Clam, MIC and Degradation Inspection, completed 02/05

Completed Procedures

0-PI-SFT-067-001.0, ERCW System Monitoring for Chemical Cleaning, completed 04/04
0-PI-SFT-067-001.0, ERCW System Monitoring for Chemical Cleaning, completed 05/04
0-PI-SFT-067-004.A, ERCW ‘A’ Train System Flush, completed 02/05
0-PI-SFT-067-004.B, ERCW ‘B’ Train System Flush, completed 01/05
0-PI-SFT-067-005.A, ERCW ‘A’ Train System Flow Balance Using Hydraulic Modeling, completed 04/02
0-PI-SFT-067-005.B, ERCW ‘B’ Train System Flow Balance Using Hydraulic Modeling, completed 03/02

Problem Evaluation Reports (PERs)

18772, Blockage Found in Unit 2 Thermal Relief Valve, 11/03
20333, Evaluation of IN 2004-01 Point Beach AFW Issue to Determine Applicability, 02/04
25348, Blockage Observed in Unit 1 Thermal Relief Valves, 05/03
26908, Low Flow Condition Observed on the 2A 690 Penetration Room Cooler, 10/03
26920, Low Flow Condition Observed on the 1B 690 Penetration Room Cooler, 10/03
27529, Blockage of Pipe Leading to Unit 2 Thermal Relief Valve, 11/03
28319, ERCW System Wall Thinning Issues, 03/04
68880, Clams Discovered in the Electrical Board Room Chiller A and Associated Temperature Control Valve, 09/04
72778, ERCW Strainer Inspections, 11/04

Miscellaneous

SQN-DC-V-7.4, Essential Raw Cooling Water System (67), Rev. 25
Traveling Water Screen Inspection, Unit 1-A, 07/03
Traveling Water Screen Inspection, Unit 1-B-B ERCW Pits, 07/03
Raw Water Treatment Program Reports, 2004-2005
Maintenance Work History, Component Cooling System Heat Exchangers 0-HEX-070-0012A/B, 1-HEX-070-0008A/B, and 2-HEX-070-0015A/B, 1991-2005
Maintenance Work History, Containment Spray Heat Exchangers 1/2-HEX-072-0007, and 1/2-HEX-072-0030, 1991-2005
Maintenance Work History, Diesel Generator Coolers 1/2-CLR-082-0280/1, 2, and 1/2-CLR-082-0290/1, 2, 1991-2005
Action Plan for PER 28319, ERCW System Piping Replacement, Rev. 0
System Health Report Card, U0-SYS 067 Essential Raw Cooling Water System, FY2005-P1
Pump Discharge Pressure, Differential Pressure, and Flow Rate Data Trending, ERCW Pumps J-A, K-A, L-B, M-B, N-B, P-B, Q-A, and R-A, 1996-2005
ERCW Chemical Treatment Trending, 2004-2005
Molluscicide Injections Heat Exchanger Flow Trending, 2002-2004
Valve Stroke Time Trending, 1/2-FCV-3-116A/B-A, 126A/B-B, 136A/B-A, 179A/B-B, 2000-2005

Section R08: Inservice Inspection Activities

1R08 Inservice Inspection Activities (IP 71111.08)

Corrective Action Program Documents

03-016523-00, lower reactor vessel head incore instrumentation leakage
05-774457-000, dried boron found on top of ductwork
05-774447-000, dried boron found on top of 2A-A LCCU03-004395-000, Accumulator #3 expansion joint downstream of piping has evidence of reddish brown boron
81632, Leakage observed on Pressurizer Safe End Welds

Nondestructive Examination Procedures

Procedure N-VT-19, Visual Inspection of Alloy 600/82/182 Pressure Boundary Components, Revision 1
Nondestructive Examination Procedure N-UT-64, Rev. 6, General Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
Nondestructive Examination Procedure N-PT-9, Rev. 25, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds
Reactor Building Post Shutdown Leakage Examination, 0-PI-SLT-068-200.0, Revision 0

Steam Generator Procedures

Sequoyah Nuclear Plant - Unit 2 Steam Generator Eddy Current Examination Guidelines, Revision 7

Other Documents

Sequoyah Nuclear Plant - Unit 2 Cycle 12 (U2C12) Steam Generator Tube Pluging Report, December 12, 2003
Tube plug list for U2C13 refueling outage
Sequoyah Nuclear Plant - Unit 2 Cycle 13 (U2C13) Steam Generator Tubing Examination Scan Plan, Rev 1
Sequoyah Nuclear Plant - Unit 2 Cycle 13 Degradation Assessment, Rev 2
Sequoyah Nuclear Plant - Unit 2 Steam Generator Eddy Current Examination Data
Problem Evaluation Report (PER) 82556, SG eddy current probe life shorter than anticipated
PER 82651, During foreign object search and retrieval (FOSAR) various pieces of material found in sludge lane filter and/or removed form SGs
PER 81784, Evaluation of Watts Bar primary to secondary leakage
PER 80588, Westinghouse SG Alternate Repair Criteria (ARC) 90 day report methodology did not precisely follow GL 95-01 guidance

Self Assessments

Self Assessment Report SQN-ENG-03-007, Boric Acid Effectiveness
Self Assessment Report CRP-ISO-04-001, ISO Industrial Program
SPP-9.7, Revision 9, Corrosion Control Program

Section R11: Licensed Operator Requalification

EPM-4, Emergency Procedure User's Guide, Revision 13

Section R12: Maintenance Rule Implementation

WO 04-781083-000, Clam Inspection of Electric Board Room Chiller B-B Condenser
WO 05-772817-000, Replace Oil Pump and Motor on Electric Board Room Chiller B-B
WO 05-771099-000, Linkage in Main Control Room B-B Air Handling Unit Binding
CDEs for System 030-K, Shutdown Board Room Chillers
WO 03-017199-000, Replace Unit 2 RWST Level Instrument Enclosures
WO 03-017199-001, Electrical Support for Replacing Unit 2 RWST Level Enclosures
WO 03-017199-002, Instrument Support for Replacing Unit 2 RWST Level Enclosures

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

Sentinel Run for 9 May through 29 May, 2005
WO 03-018004-000, Troubleshoot/Repair Valve Indication of 2-FCV-067-127-A

Sentinel Run for 23 May through 5 June, 2005
WO 05-77432-0000, Install and Seal Weld New Pipe Plug to 1-FCV-1-51
SQN Daily Work Schedule for 30 May through 5 June, 2005

Sentinel Run for 30 May through 19 June, 2005
1A-A Diesel Outage Schedule
SQN Daily Work Schedule for 6 June through 12 June, 2005

Section R15: Operability Evaluations

GENSTP3-001, Upper Boundary Temperature for Mild Environments Related to Environmental Qualification of Electrical Equipment, Revision 0

30D53EPMASD01031187, HVAC Cooling Load Calculation: Auxiliary Building RHR Pump Rooms, Floor Elevation 653-0, Revision 2

Engineering Justification for Repair of Divider Barrier Seal during U2C14
FSAR 6.2.1.3.5
0-SI-MIN-302-239.0, Testing of the Divider Barrier Seal, Revision 4
2-44W290-1,2, Seals Between Ice Condenser and Containment Vessel Details - Revision 4

0-TI-DXX-000-013.0, Temporary Equipment Control, Revision 1
SPP-10.7, Housekeeping/Temporary Equipment Control, Revision 1
1,2-47W803-2, Flow Diagram Auxiliary Feedwater, Revision 60

Section R19: Post Maintenance Testing

0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 4
0-SI-SXV-001-266.0, ASME Section XI Valve Testing, Revision 24
0-SI-SXV-000-206.0, Testing of Category A and B Valves After Work Activities, Upon Release From a Hold Order, or When Transferred from Other Documents, Revision 4
0-SI-SXV-003-266.0, ASME Section XI Valve Testing, Appendix O, Revision 26
WO 05-77325-001, Replace Panel Wiring for Diesel Generator 1B Field Resistors

Section R20: Refueling and Outage Activities

0-GO-7, Unit Shutdown From Hot Standby To Cold Shutdown, Revision 38

0-PI-ICC-068-005.0, Calibration of the Mansell Level Monitoring System Transducers, Revision 2

0-MI-MXX-088-001.0, Opening Penetrations X-54, X-88, X-108, X-109, X-117, X-118, MK-72, and Additional Equipment Building Sleeves for Maintenance Activities, Revision 17

0-MI-MXX-410-616.0, Removal and Installation of Equipment Access Hatch, Biological Shield Blocks, Doors, Bridge, and Curbs, Revision 17

0-GO-15, Containment Closure Control, Revision 19
SQS20133, Midloop Design Information Calculation, Revision 6
TI-45, Physical Verification of Core Load Prior to Vessel Closure, Revision 24
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 49
0-GO-9, Refueling Procedure, Revision 28
SQN U2C13 Mode 4 and Mode 3 Mode Change Checklists
2-SI-OPS-088-014.0, Verification of Containment Integrity, Revision 18
0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 18

Section R22: Surveillance Testing

FSAR Sections 7.2.1.1, 7.3.1.1

Section 1EP2: Alert and Notification System Testing

Records and Data

Weekly and Monthly Siren testing data June 2004 through March 2005
Maintenance records for sirens, June 2004 through March 2005

Section 1EP3: Emergency Response Organization (ERO) Augmentation

Records and Data

Sequoyah Nuclear Plant 2004 Blue Team Drill Report, 8/30/2004
L63 050412 001, Sequoyah Nuclear Plant 2005 Off Year Exercise (OYE) Report, 4/12/2005
2004 Annual Miscellaneous Drills
2005 Sequoyah Nuclear Plant Post Accident Sampling System Drill, 1/12/2005

Section 1EP4: Emergency Action Level (EAL) and Emergency Plan Changes

Records and Data

Generic Radiological Emergency Plan (REP), Revision (Rev.) 73, 74, 75, 79
Radiological Emergency Plan Appendix B Rev. 76
Plan Effectiveness Determination for Generic REP Rev. 73
Plan Effectiveness Determination for Generic REP Rev. 74
Plan Effectiveness Determination for Generic REP Rev. 75
Plan Effectiveness Determination for REP Appendix B Rev. 76
Plan Effectiveness Determination for Generic REP Rev. 79

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

Procedures

SPP-1, Corrective Action Program, Rev. 8
SPP-1.6, TVAN Self-Assessment Program, Rev. 11

Records and Data

SSA0404, REP Program Nuclear Assurance Audit, September 13, 2004
CRP-EP-04-002, Training, Drills, Maintenance of the Emergency Response Organization (ERO), 11/6/2004
INPO Emergency Preparedness Review, March 18, 2005
NA-CH-05-001, Assessment of Emergency Preparedness Performance, June 10, 2005
L63 040716 001, Sequoyah Nuclear Plant 2004 Graded Exercise (GE) Report, 7/16/2004

Problem Evaluation Reports

33676, 50.54(x) use in the TSC, 03/25/2004
83754, Problems with REP Dose Assessment, 06/06/2005
78907, OSC Team Boards, 03/17/2005
66933, Paged to Incorrect number during REP Pager Test, 08/11/2004
64464, EPIP-1 EAL Containment Barrier Procedure Error, 07/01/2004
63990, TSC communication with OSC, 06/25/04
75692, EPIP-12 and Dosimetry Packages for Security, 01/28/2005
62361, Declaration of Event, 05/28/2004
67376, REP Drill Nextel Problems (Loss of ERCW), 8/19/2004

Section 2OS1 Access Control To Radiologically Significant Areas

Procedures, Instructions, Guidance Documents, and Operating Manuals

Tennessee Valley Authority (TVA), TVA Nuclear (TVAN), Standard Programs and Processes (SPP) - 3.1, Corrective Action Program Revision (Rev.) 8
TVA, TVAN, SPP-5.1, Radiological Controls, Rev. 5
TVA, TVAN, Radiation Control Departmental Procedure (RCDP) - 1, Conduct of Radiological Controls, Rev.2
TVA, Sequoyah Nuclear Plant (SNP), Radiological Control Instruction (RCI)-01, Radiation Protection Program, Rev. 63
TVA, SNP, RCI-14, Radiation Work Permit (RWP) Program, Rev.3
TVA, SNP, RCI-15, Radiological Postings, Rev. 15
TVA, SNP, RCI-24, Control of Very High Radiation Areas, Rev. 4
TVA, SNP, RCI-28, Control of Locked High Radiation Areas, Rev. 2
TVA, SNP, RCI-29, Control of Radiation Protection Keys, Rev. 3
Technical Instruction (TI) 0-TI-NUC-000-002.0, Storing Material in Spent Fuel Pool or New Fuel Vault, Rev. 9

Records and Data Reviewed

High Radiation Area Access Control/Posting Inspection Log (Form 5.29, Dated 4/24/05)
Radiological Survey (RS) Number (No.) 020305-8, 1B-B Residual Heat Removal (RHR) and Containment Spray (CS) Heat Exchanger (HX) Room
RS No. 042505-1, U-2 Steam Generator (SG) 2&3 Handhole Platforms
RS No. 042505-2, U-2 SG 1&4 Handhole Platforms
RS No. 042505-11, U-2 SG #4 Primary Platform
RS No. 042505-12, U-2 SG #2 Primary Platform
RS No. 042505-14, U-2 SG #3 Primary Platform
RS No. 042505-16, U-2 SG #1 Primary Platform
RS No. 042505-30, U-2 Keyway
RS No. 042505-45, U-2 Inside Polar Crane Wall
RS No. 042605-9, U-2 Pressurizer Platform
RS No. 042705-1, U-2 Raceway
RS No. 092701-4, Hold Up Tank (HUT) Room
RS No. 111904-2, 1A-A RHR/CS HX Room
RWP No. 05000032, Spent Fuel Pool (SFP) Inspection, Valve Alignment, Surveillance, Surveys
RWP No. 05000069, SFP Area - Crane PMs

RWP No. 05000156, SFP - Top Side Support Work
RWP No. 05017110, Aux Bldg - Perform MOVATS Inspection & Testing
RWP No. 05027040, U-2 Lower Ctmt. - Build, Remove, and Inspect Scaffolding
RWP No. 05027081, U-2 Keyway - Under Vessel Inspections
RWP No. 05037010, U-2 Primary SG Set Up and Prep Activities
RWP No. 05037011, U-2 Secondary SG Set Up and Prep Activities
RWP No. 05037020, U-2 Installing and Removing SG Nozzle Dams
RWP No. 05047112, U-2 Reactor Cavity/Equipment Pit Decon Activities
RWP No. 05047135, U-2 Equipment Pit - Remove/Install Vortex Suppressors, Drain Plugs, and Fuel Transfer Tube Blind Flange
RWP No. 05047181, U2 Lower Ice Condenser - Ice Condenser Maintenance
RWP No. 05057003, U2 Annulus: OPs Inspections and Valve Alignment
Sequoyah Nuclear Station (SNS) VSDS Survey Report No. 042605-5, 2R161.WMF - R261 U2 Reactor Cavity
SNS VSDS Survey Report No. 050405-6, 2R100.WMF - R200 U2 Annulus
SNS VSDS Survey Report No. 051806-7, 2R100d.wmf - Unit 2 Annulus Transfer Tube

<u>Corrective Action Program (CAP) Documents</u>

Nuclear Assurance (NA) - TVAN-Wide - Audit Report No. SSA0302 - Radiological Protection and Control Audit, dated December 31, 2003
Problem Evaluation Report (PER) 68859, Ladder connecting Raceway and #1 Fan Room was locked and posted LHRA on only one side.
PER 69654 QA Containment Entry/Exit Logs were not properly completed by plant workers.
PER 70777 Plant worker lost TLD.
PER 70800 Inadequate pre-job briefing for work in U-1 Keyway.
PER 70822 Individual exceeded the RWP specified rate limit.
PER 70899 Individual exceeded the RWP specified rate limit.
PER 73002 Stanchions marking rad material storage area boundary in Turbine Bldg. Railroad bay were slightly moved causing some rad material to be partially outside the boundary.
PER 73198 LHRA keys not marked as either “tumbler key” or “security lock”.
PER 73861 Perform review to determine whether total posted area on refuel could be reduced.
PER 74749 Worker received dose rate alarm on electronic dosimeter.

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

<u>Procedures, Instructions, Guidance Documents, and Operating Manuals</u>

ALARA Pre-Planning Report (APR) 2005-12, U2C13 Refuel Outage: Steam Generator Primary Side Inspection and Maintenance
APR 2005-26, U2C13 Temporary Shielding
APR 2005-35, U2C13 Reactor Head Inspection (10 Year ISI)
APR 2005-36, U2C13 Refuel Outage: RCP 4 Rotating Element (Impeller) Change Out
APR 2005-38, U2C13 Remove/Replace Excess Letdown Heat Exchanger Head to Include Gasket Replacement, Stud Inspections, and Installation of Super Nuts
Sequoyah Nuclear Plant Site Dose Reduction Strategy, 04/16/2004
TVA, SNP, RCI-03, Prenatal Radiation Exposure Program, 02/24/2005
TVA, SNP, RCI-10, ALARA Program, Rev. 28
TVA, TVAN RCDP - 105, Personnel Inprocessing and Dosimetry Administrative Processes, Rev. 0

TVA, TVAN SPP, SPP - 3.1, Corrective Action Program, Rev. 7
TVA, TVAN SPP, SPP - 5.2, ALARA Program, Rev. 2
Work Director APR/RWP Briefing Template

Records and Data

ALARA Outage Report for U2C12
ALARA Outage Report for U1C13
Dose Records of all declared pregnant workers (4) during the period 04/01/2003 to 05/15/2005
RWP No. 05027040, U-2 Lower Containment, Build, Remove and Inspect Scaffolding
RWP No. 05027081, U-2 Keyway - Perform Under Vessel Inspections and All Support Activities
RWP No. 05027125, U-2 Lower Containment, Install, Inspect and Remove Temporary Shielding
RWP No. 05027441, Remove/Replace RCP #4 Impeller and Gasket to Include Cleaning Flange Surface
RWP No. 05037020, U-2 Lower Containment Steam Generators 1-4, Full Jump for Installing and Removing Nozzle Dams
RWP No. 05037040, U-2 Lower Containment/Steam Generators 1-4, Detorqueing and removal of Manway Covers
RWP No. 05037041, U-2 Lower Containment/Steam Generators 1-4, Installing S/G Primary Manway Inserts
RWP No. 05037100, U-2 Lower Containment. Cleaning and Inspecting of Old S/G Inserts, Insert Boxes and Nozzle Dams in the Laydown Areas.
RWP No. 05047253, U2 Reactor Cavity: 10 Year ISI - Remove/Replace Nozzle Covers, Gaskets, and Insulation in HL1, CL2, HL3, CL4; Inspect Vessel and Nozzle Supports
RWP Total, Person-REM Person-Hours and Dose Rate for APR 2005-36, Dated 05/17/05
Sequoyah Nuclear Plant (SQN) - ALARA Committee Meeting Minutes dated 05/21/04, 05/28/04, 06/30/04, 08/13/04, 10/14/04, 10/17/04, 11/07/04, 11/11/04, 01/25/05, 04/06/05, 04/14/05, 04/18/05, 04/19/05, and 04/22/05
SQN ALARA Planning Report 2005-24, U2C13 Scaffolding Installation and Removal
U2C13 ALARA Summaries, Dated 04/26/05, 05/16/05, 05/20/05

CAP Documents

PER 20725, Westinghouse employee received an unexpected dose rate alarm, 11/18/2003
PER 21310, Respiratory Protection Self-Assessment area for improvement, 09/04/2003
PER 71172, Head Inspection ALARA Improvement, 10/31/2004
PER 72087, CRDM Shroud Extra Dose, 11/15/2004
Quarterly Integrated Analysis of RADCON Performance, July-September 2003
Quarterly Integrated Analysis of RADCON Performance, 1st Quarter 2004
Quarterly Integrated Analysis of RP Performance, 2nd Quarter 2004
Quarterly Integrated Analysis of RP Performance, October-December 2004
Radiation Protection - Department Level Integrated Quarterly Review, April-June 2004
Radiation Protection - Department Level Integrated Quarterly Review, July-September 2004
Self-Assessment Report, Assessment No. SQN-RP-05-001, High Radiation Area Controls
Self-Assessment Report, Assessment No. SQN-RP-05-002, Contamination Control
TVA, TVAN SPP, SPP-3.1, Corrective Action Program, Rev. 8

Section 2PS2 Radioactive Material Processing and Transportation

Procedures, Guidance Documents and Manuals

DM-OP-044-49200, Operating Procedure for Duratek Modular Fluidized Transfer Demineralization System at TVA-Sequoyah, Dated 06/26/2002
0-VI-RCI-077-001.0, Operating Procedure for Duratek Modular Fluidized Transfer Demineralizer System (MFTDS), Rev. 0, Dated 07/17/2002
Radioactive Materials Shipment Manual, TVA, Rev. 37 and 37A (Volumes I and II)
TVA, SQN, Radwaste Handling and Shipping Instruction (RHSI) - 1.1, Packaging Filters and Items of High Levels of Radiation, Quality Related, Rev. 4, Dated 03/17/1999.
TVA, SQN, RHSI - 6, Bead Resin/Activated Carbon Dewatering Procedure for CNS 14-215 or Smaller Liners, Rev. 6, Quality Related, Dated 05/21/2001
TVA, SQN, RHSI - 1, Packaging Dry Active Waste for Shipment to a Waste Processor/Broker or a Commercial Radwaste Burial Facility Rev. 8, Quality Related, Dated 08/25/2000.
TVA, SNP System Operating Instruction 0-SO-77-29, Waste Processing, Rev. 10., Quality Related
TVA, TVAN Common Technical Procedure (CTP), RWTP - 100, Radioactive Material/Waste Shipments, Rev. 2, Dated 01/03/2005
TVA, TVAN CTP, RWTP - 101, 10 CFR Waste Characterization, Dated 04/29/2002
TVA, TVAN, CTP, RWTP-102, Use of Casks, Rev. 1, Dated 01/09/2004
TVA, TVAN SPP, SPP - 3.1, Corrective Action Program, Rev. 8

Records and Data

Auxiliary Building Units 1 & 2 - Flow Diagrams CVCS, CCD Nos.: 1, 2-47W809-4 to CCD No.: 1, 2-47W809-6 - Chemical Control
Letter Dated August 20, 2003, to Superintendent, Radwaste/Env. TVA/Sequoyah Nuclear Plant, RE: June 30, 2003, Inspection Infraction Identified by the South Carolina Department of Health and Environmental Control at Barnwell, SC
Powerhouse Auxiliary Building Units 1 & 2 - Mechanical Flow Diagrams, CCD Nos.: 1, 2-47W830-1 to CCD No: 1, 2-47W830-7 - Waste Disposal System
RW IEB79-19 CERT 2001 Memo, Designation of Individuals Responsible for the Safe Packaging, Transfer and Transport of Radioactive Material, Dated 02/06/2001
Shipping Documents for Shipments: SNP 05-0411, 05SEQ0004, 151N-08, SNP 05-0412, SNP 04-0302, SNP RW-1211, SNP 04-0308, SNP 03-1205, SNP 04-0603, SNP 05-0403, SNP 05-0303, and SNP 05-0405
TVA Letter Dated September 18, 2003, to the Assistant Director of the State of the State of South Carolina, Department of Health and Environmental Control, Division of Waste Management, RE: Inspection Fraction as a Result of a June 30, 2003 Inspection at Chem-Nuclear Site in Barnwell, SC.

CAP Documents

PER 33719, RadCon dry cask procedures
PER 34095, Radwaste shipping
PER 60953, Contamination found in Cask
PER 66683, Dose reduction - Radwaste
PER 68286, Elevated Dose Rates - Unit 1
PER 69142, RadWaste LSA box issue
PER 72691, RCS filter extra dose
PER 73169, HIC damage
PER 73406, Rad Material DOT requirements
PER 76816, Temperature Inaccuracy
PER 78311, New Fuel truck Incoming Notification and Survey
PER 79898, Improper use of MK-1 resulted in extra dose to workers

Section 4OA1: Performance Indicator Verification

Procedures, Guidance Documents and Manuals

SNP Desktop Guide for Identification and Reporting of NEI 99-02 Performance Indicators
TVA, TVAN SPP, SSP - 3.1, Corrective Action Program, Rev. 7
TVA, TVAN SPP, SPP-3.4, Performance Indicator and MOR Submittal Using INPO Consolidated Data Entry, Rev. 2

Records and Data

Monthly Liquid and Gaseous Effluent Dose Reports: July-October 2004 and January-March 2005
Monthly Assessments for Occupational Exposure PI: January-March 2005
Individual RWP Access Records for exit doses exceeding 100 mrem: July 2004-March 2005

Annual Examination EPIP Classification Documentation for 9/04
Selected training records of drill/exercise participation by ERO personnel during 2004-2005
Siren weekly and monthly tests 6/04 thru 3/05

Section 4OA2: Identification and Resolution of Problems

Assessment SQN-ENG-04-004, ECCS Venting, Revision 0
PER 61224, Enhancements From ECCS Gas Venting Self-Assessment
PER 31911, Piping Below Valve 1-FCV-63-8 Found Voided
PER 21834, Piping Below Valve 1-FCV-63-8 Found Voided
PER 73203, A Gas Void of Approximately 2.8 Cubic Feet Below Valve 1-FCV-63-8
PER 20658, 0.6 Cubic Feet of Gas Found Below Valve 2-FCV-63-8
PER 20142, Piping Below Valve 2-FCV-63-8 Found Voided
PER 68880, Clams Discovered in Electric Board Room Chiller A
PER 68650, Low Flow in RHR Pump 2A Room Cooler
PER 68693, Flow Blockage in Charging Pump 1A Room Cooler and Elevation 714 Penetration Room Cooler
PER 68862, Clam Shell Observed in 0-FI-090-0134
PER 68947, Low Flow in RHR Pump 2B Room Cooler

PER 69364, Clams Discovered in Main Control Room Chiller B
PER 70250, Clams Discovered in Main Control Room Chiller B Temperature Control Valve
1-PI-OPS-000-020.1, Operator At The Controls Duty Station Checklists Modes 1-4, Revisions 27 & 29
2-PI-OPS-000-022.1, Operator At The Controls Duty Station Checklists Modes 1-4, Revision 26
0-PI-ISO-000-001.0, Periodic Check for Presence of Water in Various ECCS Piping Locations, Revision 6
0-TI-CEM-260-011.21, Chemical Analytical Methods Gas Analysis, Revision 15

Section 4OA3: Event Followup

LER 50-328/2005-001-00, Unit 2 Reactor Trip Following Closure of Main Feedwater Upon Inadvertent Opening of Control Breakers
0-MI-EBR-317-010.0, Testing of Molded Case Circuit Breakers, Revision 33
Reactor Trip Report and Root Cause Analysis for PER 77234
SPP-2.2 Administration of Site Technical Procedures, Revision 13
MMDP-1, Maintenance Management System, Revision 7S1
LER 50-327/2005-001-00, Automatic Reactor Trip Following Loss of Turbine Auto Stop Oil Pressure
Reactor Trip Report and Root Cause Analysis for PER 80518

Section 4OA5: Other Activities

SPP-7.1, On Line Work Management, Revision 6
GOI-6, Apparatus Operations, Revision 103
OPDP-9, Emergent Issue Response, Revision 2
PER 60496, Corporate PER on Grid Reliability Issues
PER 61350, Sequoyah PER on Grid Reliability Issues

ISFSI Documents Reviewed

Sequoyah Nuclear Station VSDS Survey Report Nos. 010705-4, 020405-1, 030705-2, 040705-6, 071304-11, 081004-6, 090604-4, 100704-4, 110604-7, and 120504-1
TVA, SNP, Surveillance Instruction (SI) No. 0-SI-DCS-079-001.0, Hi-Trac Average Surface Dose Rates, Rev. 2, Dated 05/13/04
TVA, SNP, SI No. 0-SI-DCS-079-002.0, Hi-Trac Contamination Surveys, Rev. 2, Dated 05/13/04
TVA, SNP, SI No. 0-SI-DCS-079-003.0, Hi-Storm Average Surface Dose Rates, Rev. 2, Dated 05/13/04

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
CAP	Corrective Action Program
CFR	Code of Federal Regulations
DAW	Dry Active Waste
DOT	Department of Transportation
EAL	Emergency Action Level
EDs	Electronic Dosimeters
EP	Emergency Plan
ERO	Emergency Response Organization
HRA	High Radiation Area
ISFSI	Independent Spent Fuel Storage Installation
LHRA	Locked High Radiation Area
No.	Number
OA	Other Activities
PCP	Process Control Program
PI	Performance Indicator
PER	Problem Evaluation Report
radwaste	Radioactive Waste
Rev.	Revision
RP	Radiation Protection
RWP	Radiation Work Permit
SPP	Standard Programs and Processes
TS	Technical Specification
U2	Unit 2
UFSAR	Updated Final Safety Analysis Report
VHRA	Very High Radiation Areas
OSC	Operations Support Center
RSPS	Risk Significant Planning Standard
TSC	Technical Support Center