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

July 25, 2003

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

SUBJECT: SEQUOYAH NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2003004 AND 05000328/2003004

Dear Mr. Scalice:

On June 28, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Sequoyah Nuclear Power Plant, Units 1 and 2, including the steam generator replacement project on Unit 1. The enclosed report presents the results of the integrated inspection which were discussed on July 8, 2003, with Mr. Mike Lorek 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.

On the basis of the results of this inspection, no findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) components 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: (See page 2)

Enclosure: Inspection Report 05000327/2003004, 05000328/2003004
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos: 50-327, 50-328

License Nos: DPR-77, DPR-79

Report No: 05000327/2003004, 05000328/2003004

Licensee: Tennessee Valley Authority (TVA)

Facility: Sequoyah Nuclear Plant, Units 1 & 2

Location: Sequoyah Access Road
Soddy-Daisy, TN 37379

Dates: 4/06/2003 - 6/28/2003

Inspectors:
S. Freeman, Senior Resident Inspector
R. Telson, Resident Inspector
T. Kolb, Project Engineer
R. Carrion, Project Engineer (Sections 1R06, 1R12, 1R16)
S. Vias, Senior Reactor Inspector (Sections 1R08, 4OA5.1, 4OA5.3.1-10)
E. Testa, Senior Health Physicist (Sections 2OS1, 2OS2, 2PS2, 4OA1.2, 4OA1.3)
D. Jones, Senior Health Physicist (Sections 2OS1, 2OS2, 2PS2, 4OA1.2, 4OA1.3)
J. Kreh, Health Physicist (Sections 2OS1, 2OS2, 2PS2, 4OA1.2, 4OA1.3)
J. Lenahan, Senior Reactor Inspector (Sections 4OA5.4.5, 4OA5.4.6)
J. Ennis, Physical Security Inspector (Section 4OA5.3.11)
D. Holman, Physical Security Inspector, (Section 4OA5.3.11)
W. Sartor, Senior Emergency Preparedness Inspector (Sections 1EP2-5, 4OA1.4)

Approved by: S. Cahill, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000327/2003-004, IR 05000328/2003-004, 4/6/2003 - 6/28/2003, Sequoyah Nuclear Power Plant, Units 1 & 2**,** resident inspector integrated report.

The report covered a three-month period of inspection by resident inspectors and region based Project Engineers and announced inspections by eight region based inspectors. It includes inspection of the Unit 1 steam generator replacement project. 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. <u>NRC-Inspector Identified and Self-Revealing Findings</u>

None

B. <u>Licensee Identified Violations</u>

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period in a scheduled refueling and steam generator replacement outage. Outage activities were completed and the unit was taken critical on June 15, 2003. The unit returned to 100 % rated thermal power on June 23, 2003.

Unit 2 operated at or near 100 % rated thermal power during the inspection period except for a reactor trip on April 12, 2003, due to problems in the turbine supervisory circuits. The problems were repaired and the unit returned to 100 % rated thermal power on April 15, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors observed the licensee response to a tornado watch on May 5, 2003. The inspectors reviewed licensee Procedure AOP-N.02, Tornado Watch/Warning, Revision 11, for its effectiveness to limit the risk of tornado-related initiating events and to adequately protect mitigating systems from the effects of a tornado. In addition, the inspectors verified the securing of large outside cranes in accordance with guidance in Topical Report 24370-TR-C-002, Rigging and Heavy Load Handling.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial system walkdowns of the following three systems to verify the capability of redundant or diverse systems and components and that defense-in-depth was maintained during periods when safety equipment was inoperable. The inspectors reviewed applicable operating procedures, walked down control systems components and verified that identified problems were entered into the corrective action program.

- A-train and Turbine-Driven Train of Auxiliary Feedwater (AFW) during unavailability of Motor Driven AFW Pump 2B-B
- Cross-train Emergency Core Cooling Systems (ECCS) equipment during unavailability of 1A Essential Raw Cooling Water (ERCW) header
- A-train Containment Spray System during unavailability of Containment Spray (CS) Pump 2B

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted a tour of the ten areas listed below to assess the material condition and operational status of fire protection. The inspectors verified that transient combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures, that fire protection equipment was available for use, and that fire barriers were intact. Documents reviewed are listed in the attachment.

- Control Building Elevation 706 (Cable Spreading Room)
- Essential Raw Cooling Water Building
- Control Building Elevation 685 (Reactor Protection Room)
- Auxiliary Building Elevation 714 (Component Cooling System (CCS) Heat Exchangers, Safety-related Chillers, and AFW level control valves)
- Auxiliary Building Elevation 653 (Unit 2 Residual Heat Removal (RHR) Pump Rooms)
- Emergency Diesel Generator Building
- Unit 1 Lower Containment Area
- Turbine Building Elevation 708 (Unit 1 Turbine Oil, Main Feedwater Pump (MFP) Turbines, Condenser Area)
- Auxiliary Building Elevation 690 (Penetration Room & Pipe Gallery)
- Switchyard (Common Station Service Transformers)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed selected risk-important internal flood protection barriers to evaluate the adequacy of protecting risk-important equipment. The inspectors walked down the basement (Elevation 662') of the Unit 2 Turbine Building to assess the effectiveness of the licensee's internal flooding mitigation program to protect susceptible systems and equipment. The inspectors evaluated the operability and material condition of internal flood protection-related components in the risk-significant areas to verify that the floor drain system was operable, including turbine building sump pumps A (0-PMP-40-1A) and B (0-PMP-40-1B), respectively, and the emergency portable sump pump (0-PMP-40-514).

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's execution and on-line monitoring of bio-fouling controls and cleanliness of the CCS Heat Exchanger 1A-2 plates during a routine inspection and cleaning. The inspectors directly examined the as-found and as-left condition of the opened heat exchanger for adverse conditions. The inspectors also reviewed corrective action documentation and interviewed personnel involved in the cleaning process to verify that the licensee was inspecting and cleaning the heat exchanger with sufficient frequency to detect conditions that could adversely affect its ability to remove heat.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

The inspectors observed Unit 1, Cycle 12 in-process ISI work activities during the 2nd interval, 3rd ISI period, 1st outage and reviewed selected ISI records. The observations and records were reviewed for compliance to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with 1995 Addenda). The following Unit 1 ISI examinations were observed/reviewed:

Penetrant	(PT)	1-SIH-022-IA	B-K	B10.20	10" Cold Leg #4
		CRDM #3			RVH CRDM #3 nozzle
Visual	(VT-3)	1-RHRH-007	F-A	F1.10B	14"
		1-CVCH-361	F-A	F1.10B	3"
Radiographic	(RT)	1-RC-11-1			S/G 2 Safe end to elbow
		1-RC-03-1			S/G 1 Safe end to elbow
		1-RC-13-1			S/G 2 RCS CL
		1-RC-18-1			S/G 3 RCS HL
		1-RC-19-1			S/G 3 RCS CL
Ultrasonic	(UT)	1-MS-021-1			Main Steam to S/G 3 Nozzle/Elbow
		1-RC-02			S/G 1 RCS HL
		1-FW-131			S/G 3 to Feedwater

Qualification and certification records for examiners, equipment and consumables, and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. In addition, a sample of ISI issues in the licensee's corrective action program were reviewed for adequacy. The following records/documents were reviewed:

NDE Examiner/QC Inspector Qualification Certification and Visual Acuity Records Examined

Examiner	Method-Level
SW, TG	VT-II, PT-II
HD	UT-III, PT-III
SD	PT-III
PV, MP, PB, JP	UT-II
MH, DM, WT, JE, MG	RT-II
RT, DW, RP	PT-II

NDE Equipment and Consumables

Probes:	KBA- Model: Comp-G, SN:00W3MY KBA- Model: SWS-Comp, SN:00XJ58/00XJ59
Thermometer:	T-26787 ID:573205; E27489; E26104; 558273
Scope:	Krautkramer-Model USN-52L S/N:E30218 Krautkramer-Model USN-52L S/N:E21664
Cal Std.	6011137 A & B
Cal Block:	SQ-5, SQ-112, SQ-64
Couplant:	01225
Cleaner:	SKC-S: 01B06K, 02B01K
Penetrant:	SKL-SP: 98M02K, SKL-SP1, SKL-WP
Developer:	SKD-S2: 96L0SK, 01G06K

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training on June 25, 2003. The scenario involved a recoverable loss of condenser vacuum followed by a Loss of Coolant Accident (LOCA) with several component malfunctions. This placed the simulated unit in a functional response procedure for high containment pressure. The inspectors observed crew performance in terms of communications; ability to take timely and proper actions; prioritizing, interpreting and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; 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 Technical Specification (TS) actions; and group dynamics involved in crew performance. The inspectors also reviewed simulator fidelity to verify that it closely paralleled the recent steam generator replacement on Unit 1 or accounted for the differences between units.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following two maintenance activities, which were identified because either they indicated equipment performance issues or affected equipment that performed high safety significant functions, to verify the effectiveness of the activities. Documents reviewed are listed in the attachment.

- Emergency Diesel Generator (EDG) 1A-A and 2A-A unavailability in excess of maintenance rule performance criteria
- Maintenance of 6.9 kV Breakers

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed six activities to verify that the appropriate risk assessments were performed prior to removing equipment from service. When emergent work was performed, the inspectors verified that the risk for the work was assessed and required equipment was protected. The inspectors referenced Procedure SPP-7.1, Work Control Process, Revision 4, and Instruction, 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Revision 7, during these inspection activities.

- Removal of Motor Driven AFW Pump 2B-B from service for maintenance
- Removal of ERCW Header 1B coincident with CCS Pump C-S maintenance
- Removal of C station service and control air compressor for emergent maintenance
- Removal of Auxiliary Building Gas Treatment System (ABGTS) Train A from service for duct inspection
- Unavailability of CS System Train 2B due to recirculation lineup of Refueling Water Storage Tank (RWST)
- Unavailability of EDG 1B-B during required surveillance testing

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

During the seismic disturbance of April 29, 2003, the inspectors reviewed the operator logs, plant computer information, and associated Problem Evaluation Reports (PERs). The inspectors also conducted interviews with operators to determine what occurred and how the operators responded, and to determine if the response was in accordance with the associated procedures. Documents reviewed are listed in the attachment.

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 operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred. Documents reviewed are listed in the attachment.

- PER 02-005083-000, Shutdown Board 2B-B logic relay K609X
- PER 03-005768-000, Units 1, 2 Turbine Driven AFW pumps - Installation of incorrect low pressure steam traps
- PER 03-007064-000, ERCW Drain Valve broke off from header
- PER 03-007360-000, Unit 1 Thermal Relief Valve blockage
- PER 03-008296-000, Siemens 6.9 kV Breaker Failed Receipt Inspection

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWAs)

a. Inspection Scope

The inspectors reviewed the cumulative effects of deficiencies that constituted operator workarounds to determine whether or not they could affect the reliability, availability, and potential for misoperation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also assessed whether OWAs were being identified and entered into the licensee's corrective action program at an appropriate threshold. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the eight 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). Additional documents reviewed are listed in the attachment.

- WO 03-004376-000, Rework/Replace parts and/or controller and verify calibration of Unit 2 Steam Generator (SG) 1 Power-Operated Relief Valve (PORV) which opened spuriously following Unit 2 trip
- WO 03-002824-000, Replace erratic 6900V logic relay K609X which had erratic readings during surveillance testing
- 0-SI-SFT-072-138.0, Containment Spray-Spray Nozzle Test, Revision 3
- 1-STI-088-156.0, Primary Containment Vessel Post-Modification Pressure/New Weld Leakage Inspection Test, Revision 0
- 1-SI-SFT-065-001.B, Emergency Gas Treatment System Annulus Vacuum Draw Down Test Unit 1 Train B, Revision 5
- WO 01-03501-000, Remove and Replace Emergency Gas Treatment System (EGTS) ductwork
- WO 03-008636-000, Repair/Clean Pressure Control Valve 1-PCV-65-81
- 1-RT-MOD-520-002.0, 3 % Load Step Change, Revision 0

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

Between April 6, 2003 and June 17, 2003, the inspectors observed portions of the refueling, heatup, and startup activities associated with the Unit 1 steam generator replacement outage to verify that the licensee maintained defense-in-depth commensurate with the outage risk plan and applicable TS. The inspectors monitored licensee controls over the outage activities listed below and reviewed the licensee's response to debris found in the reactor vessel during refueling activities. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including daily outage reports, to evaluate defense-in-depth commensurate with the outage safety plan and compliance with the applicable TS when taking equipment out of service.
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met.
- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling.
- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after-core offload.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Reactivity controls to verify compliance with TS and that activities which could affect reactivity were reviewed for proper control within the outage risk plan.
- Containment closure for control of containment penetrations in accordance with refueling TS, to ensure that containment closure could be achieved during selected configurations, and to verify maintenance of secondary containment in accordance with TS.
- Refueling activities for compliance with TS, to verify proper tracking of fuel assemblies from the spent fuel pool to the core, and to verify foreign material exclusion was maintained.
- Removal activities associated with debris found during coreplate inspection on May 25, 2003, to verify that the evaluation of nineteen objects removed from and the nine objects left in the reactor pressure vessel after refueling would not impact operation of the unit.
- Reduced inventory and mid-loop conditions for commitments to Generic Letter 88-17 to verify that these commitments were in place, that plant configuration was in accordance with those commitments, and that distractions from unexpected conditions or emergent work did not affect operator ability to maintain the required reactor vessel level.
- Heatup and startup activities to verify that appropriate prerequisites for mode changes were met prior to changing modes, that containment integrity was established, that debris was not left that could affect the containment sump, and that core operating limit parameters were consistent with core design.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the six surveillance tests identified below, by witnessing testing and/or reviewing the test data, the inspectors verified that the systems, structures, and components involved in these tests satisfied the requirements described in the TS, the Updated Final Safety Analysis Report (UFSAR), and applicable licensee procedures, and that the tests demonstrated that the Systems, Structures, and Components (SSCs) were capable of performing their intended safety functions.

Documents reviewed are listed in the attachment. Those tests included the following:

- 1-SI-OPS-082-007.A, AC Electrical Power Source Diesel Generator 1A-A, Revision 28
- 0-SI-MIN-061-109.0, Ice Condenser Intermediate and Lower Inlet Doors and Vent Curtains, Revision 1
- 1-SI-ICC-003-042.4, Channel Calibration of Steam Generator 1 Level Channel IV, Rack 12, Loop L-3-42, Revision 6
- 1-SI-OPS-082-026.B, Loss of Offsite Power With Safety Injection - D/G 1B-B Test, Revision 26
- 1-SI-ICC-068-320.3, Channel Calibration of Pressurizer Level Channel III Rack 9, Loop L-68-320, Revision 11
- 2-SI-SXP-070-201.B, CCS Pump 2B-B Performance Test *

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the two temporary modifications described below, to verify that the design was adequate, the modification was properly installed, the modification did not affect system operability, drawings and procedures were appropriately updated, and post-modification testing was satisfactorily performed. Documents reviewed are listed in the attachment. Those modifications included the following:

- DCN EP-001, 002, WP&IR C-MHX-026, Unit 2 temporary airlock door
- TACF 1-03-0029-057, Inhibit Stator Ground Relay trip function

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert Notification System Testing

a. Inspection Scope

The inspector reviewed the alert (siren) and notification system (ANS) designed to meet the acceptance criteria of Section B of Appendix 3, NUREG-0654, and described in Section B.8 of Appendix B to the TVA NP (Nuclear Power) Radiological Emergency Plan (REP). The monthly complete cycle tests, bi-weekly silent tests, and annual growl tests were reviewed against the test frequencies commitments listed in paragraph B.8.1 of the REP.

The inspector reviewed testing results, assessed the failure rate of individual sirens and the effectiveness of repairs, and reviewed any changes related to the siren system.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing

a. Inspection Scope

The inspector reviewed Figure B-2 and Figure B-1 of the REP to determine the licensee's commitment for on-shift and augmentation staffing respectively. The results of the off-hours augmentation drill, most recently conducted on November 2, 2002, were evaluated against the requirements listed in the notes to Figure B-1 of Appendix B to the REP.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed the REP to identify any changes for review in accordance with the requirements of 10 CFR 50.54(q). There had been no changes made to the Sequoyah Appendix B since the last inspection.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. Items reviewed included exercise and drill critique reports, emergency preparedness assessment reports done by the Nuclear Quality Assurance, and the licensee's Corrective Action Program. The review was conducted against the requirements listed in Section 14.4 and 16.4 of the NP-REP.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

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

2OS1 Access Control To Radiologically Significant Areas

.1 Access Control

a. Inspection Scope

The inspectors evaluated licensee activities for monitoring and controlling worker access to radiologically significant areas. The inspection included direct observation of administrative and physical controls, appraisal of the knowledge and proficiency of radiation workers and health physics technicians (HPTs) in implementing radiological controls, and review of the adequacy of procedural guidance and its implementation.

The inspectors observed implementation of radiological controls for selected Radiation Area (RA), Radioactive Material Area (RMA), and High Radiation Areas (HRA) locations within Radiologically Controlled Areas (RCAs). Area radiological posting and material labeling of materials at these locations were evaluated for consistency with procedural guidance and compliance with regulations. The inspectors directly observed the posting and locking status of selected Locked High Radiation Area (LHRA) locations in the Unit 1 (U1) and Unit 2 (U2) auxiliary buildings and the U2 lower containment. Independent dose-rate measurements were taken in the U2 lower containment, radioactive waste (radwaste) processing and storage areas, and 1B-B containment spray heat exchanger room. Results of the independent measurements were compared to current licensee surveys.

The inspectors evaluated licensee performance during containment spray heat exchanger 1B-B maintenance and transfer of a used steam generator (S/G) from containment to its interim storage facility. The inspectors attended the heat exchanger maintenance pre-job briefing, evaluated the implementation of radiological controls, observed the HPTs' and radiation workers' performance, evaluated Radiation Work Permit (RWP) requirements and electronic dosimeter (ED) alarm setpoints, and discussed the task evolutions with selected personnel. During general observations of outage work, the inspectors queried radiation workers on RWP requirements associated with their tasks in progress.

The inspectors reviewed administrative guidance documents and procedures for control of material stored in spent fuel pools and LHRA locations, including access controls, posting, surveys, and RWP use. The inspectors reviewed selected RWPs and surveys of such areas to evaluate the adequacy of RA, HRA, and airborne area radiological controls. Health Physics supervision was interviewed regarding administrative control of LHRA keys, as well as changes to procedural guidance for access controls.

Radiation protection program activities and their implementation were evaluated against Title 10 Code of Federal Regulations (10 CFR) 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. Licensee procedures, records, and other documents reviewed within this inspection area are listed in Section 2OS1 of the report attachment.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

Selected Problem Evaluation Reports (PERs) related to access control issues were reviewed in detail and discussed with Radiation Control (RadCon) supervision. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure TVAN Standard Programs and Processes, SPP-3.1, Corrective Action Program, Revision (Rev.) 4. Licensee documents evaluated during the inspection of this program area are listed in Section 2OS1 of the report attachment.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

.1 As Low As Reasonably Achievable (ALARA)

a. Inspection Scope

Implementation of the licensee's ALARA program during the Unit 1 Steam Generator Replacement Refueling Cycle 12 was observed and evaluated by the inspectors. The inspection included evaluation of ALARA activities for high person-rem tasks, assessment of licensee source-term reduction efforts, and review of historical dose data. The reviewed high dose expenditure tasks included:

- Inspection of the reactor head
- Upper internals removal
- Lower internals replacement
- Steam Generator replacement
- Steam Generator Scaffold Removal

The jobs and implementation of ALARA principles were observed via closed-circuit television and during tours of work sites. Projected dose expenditure was compared to actual dose expenditure values and noted differences were discussed with the ALARA staff. Changes to dose budgets relative to any changes in job scope were also discussed. The inspectors reviewed ALARA committee meeting minutes and evaluated ALARA initiatives for these and other outage jobs. The Steam Generator Replacement

Project (SGRP) RadCon Handbook was also reviewed by the inspectors. The inspectors attended pre-job briefings and evaluated communication of ALARA goals, RWP requirements, and industry lessons-learned. Maintenance department understanding of dose budgets and ALARA concepts was assessed through discussions with radiation workers and job sponsors. Management support for ALARA was evaluated through interviews with ALARA staff and the RadChem Manager. The inspectors reviewed applicable procedural guidance and five exposure reduction plans to assess procedural and administrative guidance for ALARA activities.

The licensee's source term reduction program, including cobalt reduction, was evaluated through discussions with the chemistry supervisor and review of dose rate trends for primary side piping. Selected parts of a temporary shielding procedure and the outage shielding plan were assessed. The inspectors interviewed the RadChem Manager and Chemistry Manager.

Historical dose data for collective exposure was reviewed from Fiscal Year (FY) 1999 through FY 2002. Dosimetry procedure guidance, as applicable, was reviewed to assess licensee controls for declared pregnant workers. Documents reviewed are listed in the attachment to this report.

The inspectors toured the onsite mock-up facility to evaluate the training of radiation workers engaged in specialized work activities associated with used steam generator removal and new generator installation. The inspectors also reviewed the Health Physics Emergency action plans for a "Dropped SG Immediate Response Inside the Containment" and "Dropped SG Immediate Response Outside Containment."

The inspectors observed the rig-out of the U1 steam generator. The observation included independent surveys of RCA boundaries, installation of lay-down fixtures, rolling RCA dose control, operation of site exit portal monitors during transit, and dose surveys of exterior and inside mid-plane areas of the used steam generator storage facility. The inspectors independently verified evacuation and control of offices that were located inside the RCA during generator transit to the old steam generator storage facility. Surveys of the removal and transport of the other three old steam generators were evaluated for RCA dose control and worker doses received as measured by ED monitoring.

The licensee's ALARA program was evaluated against the requirements of 10 CFR Part 20 and TS Section 5.4.1, Regulatory Guide (R.G) 1.33, Quality Assurance Program Requirements, as well as the guidance contained in R.G 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable, and R.G 8.13, Instruction Concerning Prenatal Radiation Exposure.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

Five licensee PERs and one self-assessment associated with ALARA activities were reviewed and assessed. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure TVAN Standard Programs and Processes, SPP-3.1, Corrective Action Program, Revision (Rev.) 4. Licensee documents evaluated during the inspection of this program area are listed in Section 2OS2 of the report attachment.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

.1 Waste Processing and Characterization

a. Inspection Scope

The inspectors evaluated the operational status of selected liquid and solid radwaste processing systems and equipment. Inspection activities included document and record review, interviews with responsible plant personnel, and direct inspection of selected processing equipment and associated piping.

The inspectors observed the material condition and configuration of the systems and equipment used for spent resin and liquid radwaste processing. Systems and equipment abandoned in place were examined for isolation from systems in use and material condition. Operational radwaste processing system material condition and configurations were examined during facility tours. Procedural guidance for monitoring changes in waste streams; classifying radioactive wastes; and for transferring, sampling, and preparation of spent resins for temporary storage and final disposal was assessed. Current records were reviewed to evaluate application of radwaste scaling factors and their supporting analyses for hard-to-detect radionuclides. The inspectors assessed the licensee's procedure for transferring, sampling, and preparation of spent resins for temporary storage and final disposal.

Operational radwaste processing system equipment and configurations were inspected against system descriptions and process flow diagrams depicted in the UFSAR Section 11, Radioactive Waste Management. Radwaste processing and storage guidance was reviewed against UFSAR, Section 11.5, Solid Waste Management System, Process Control Program, Rev. 3, 10 CFR 61.55, and NRC Branch Technical Position documents on Radioactive Waste Classification dated May 1983 and January 1995. Reviewed procedures and records are listed in Section 2PS2 of the report attachment.

b. Findings

No findings of significance were identified.

.2 Transportation

a. Inspection Scope

The inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included review of procedures and completed shipping records, assessment of worker knowledge and proficiency in shipping activities, and direct observation of shipment preparation activities.

The inspectors observed shipment preparations for a package containing contaminated laundry. Generation of shipping papers, package contact dose rates measurements, vehicle inspection, and driver instructions were observed by the inspectors. Responsible staff were interviewed regarding package requirements, and radiation dose rate, and contamination limits. Current training records for individuals involved in transportation activities were reviewed.

Transportation program guidance and implementation were reviewed against regulations detailed in 10 CFR 71 and 49 CFR 170-189, and applicable licensee procedures. In addition, training activities were assessed against 49 CFR 172, Subpart H, and the guidance documented in NRC Bulletin 79-19. The reviewed procedures and records are listed in Section 2PS2 of the report attachment.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

Issues identified through self-assessments and PERs associated with radwaste processing and radioactive material transportation activities were reviewed. 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. 4. Specific assessments, audits, and PERs reviewed and evaluated in detail for this inspection area are listed in Section 2PS2 of the report attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors sampled licensee submittals for the two PIs listed below for the period from April 1, 2002 through March 31, 2003. To determine the accuracy of the PI data reported for that period, guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, was used to verify the basis for reporting each indicator.

- Reactor Coolant System Activity
- Reactor Coolant System Leakage

The inspectors reviewed portions of the operator and chemistry logs to verify that the licensee had accurately determined the Reactor Coolant System (RCS) activity and leakage during the previous four quarters for both units. The inspectors also observed the performance of Procedure 0-SI-OPS-068-137.0, RCS Water Inventory, which determines the amount of RCS leakage. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's records and data generated during the fourth quarter of 2002 and the first quarter of 2003 for the Occupational Exposure Control Effectiveness PI. The information reviewed included data reported to the NRC, pertinent corrective action program issues, and selected radiological control program records. Selected PERs initiated during the review period were reviewed and assessed for potential PI occurrences. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and PERs reviewed and evaluated for this PI verification are listed in Section 4OA1 of the attachment to this report.

b. Findings

No findings of significance were identified.

.3 Public Radiation Safety Cornerstone

a. Inspection Scope

The licensee's records and data generated during the period October 2002 through March 2003 for the Radiological Effluent Control Performance Indicator (PI) were reviewed. The information reviewed included data reported to the NRC, pertinent corrective action program issues, and selected Radiological Effluent Control program records. The inspectors assessed the licensee's monthly reviews for PI occurrences which were performed pursuant to procedure SPP-3.4, Performance Indicator for NRC Reactor Oversite Process, Rev. 0. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and PERs reviewed and evaluated for this PI are listed in Section 4OA1 of the report attachment.

b. Findings

No findings of significance were identified.

.4 Emergency Preparedness Cornerstone

a. Inspection Scope

On June 24-27, 2003, licensee records for Emergency Response Organization (ERO)Drill/Exercise Performance, ERO Drill Participation, and Alert and Notification System Reliability were reviewed to determine whether the submitted PI values through the first quarter of 2003 were calculated in accordance with the guidance contained in Section 2.4 (Emergency Preparedness Cornerstone) of NEI 99-02, Revision 2, "Regulatory Assessment Performance Indicator Guideline."

The inspector assessed the accuracy of the PI for ERO drill and exercise performance (DEP) over the past eight quarters through review of a sample of drill and exercise records. The inspector assessed the accuracy of the PI for ERO drill participation during the previous eight quarters for personnel assigned to key positions in the ERO. The inspector assessed the accuracy of the PI for the alert and notification system reliability through review of a sample of the licensee' s records of the monthly full cycle and bi-weekly silent tests.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up Unit 2 Reactor Trip

a. Inspection Scope

On April 12, 2003, following a Unit 2 reactor trip due to a spurious operation of the turbine high bearing vibration relay and turbine trip, the inspectors evaluated plant status, mitigating actions, and the licensee's classification of the event, to enable the NRC to determine an appropriate NRC response. Pinched wiring in the turbine supervisory cabinet actuated the high vibration relay as operators were attempting to reset a status light. The event was reported to the NRC as event notification (EN) 39753 and documented in the licensee corrective action program as PER 03-004354-000.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 NRC Temporary Instruction (TI) 2515/150, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Bulletin 2002-02)

a. Inspection Scope (TI 2515/150 and Order 03-009)

The inspectors observed activities relative to inspection of the Unit 1 reactor pressure vessel (RPV) head penetrations in response to NRC Bulletin 2002-02 and Order 03-009. The guidelines and criteria for the inspection were provided in NRC temporary inspection (TI) procedure TI 2515/150, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Bulletin 2002-02). The inspectors had previously reviewed the Unit 1 bare metal visual examination and documented their observations in Section 4OA5.1 of Inspection Report 50-327, 328/03-03. The inspection during the current inspection period included review of nondestructive examination (NDE) procedures, assessment of NDE personnel training and qualification, and observation and assessment of subsequent Ultrasonic (UT) and Liquid Penetrant (PT) examinations. Discussions were also held with contractor representatives and licensee personnel. The inspectors reviewed licensee-identified boron deposits around CRD nozzle 3 and boron deposits on six CRD nozzles on the periphery of the head. The activities were examined to verify licensee compliance with regulatory requirements and gather information to help the NRC staff identify possible further regulatory positions and generic communications. Specifically, the inspectors reviewed or observed: (1) the in-process acquisition and analysis system of "Accusonex" UT circumferential blade probe examination data of reactor vessel head control rod drive (CRD) nozzles in the annulus between the outside surface of the thermal sleeve and the inner diameter (ID) of the nozzle; and (2) the resultant PT inspection of the J-groove weld and RPV head penetration nozzle base material of any area in question.

b. Findings and Observations

No findings of significance were identified, but the following documents the results of the inspectors' verification activities for each section of the TI:

1) Verification that the examinations were performed by qualified and knowledgeable personnel.

The inspectors verified that NDE (UT & PT) inspections were being performed in accordance with approved and demonstrated procedures with trained and qualified inspection personnel. The inspectors reviewed the training and qualification records and determined that all the examiners had significant experience, including experience inspecting RPVs.

2) Verification that the examinations were performed in accordance with demonstrated procedures.

The inspectors verified by observation and discussions with the examination personnel that the licensee's UT examinations for six nozzles (3, 53, 64, 65, 72, and 73) were examined in accordance with Framatome-ANP examination procedure - Remote Ultrasonic Examination of Reactor Head Penetrations, 54-ISI-100-09, Rev. September 9, 2002. These examinations were performed beneath the vessel head using the "U.S. Blade Probe Tool" in conjunction with the Framatome-ANP "ACCUSONEX" automated data acquisition and analysis system. The ultrasonic probe used for performing the examinations, "Circumferential Blade Probe," was inserted into the annulus between the outside surface of the thermal sleeve and the inner surface of the nozzle. After positioning the probe onto the nozzle inside surface, subsequent scanning was performed in the axial direction. Indexing of the probe was performed in the circumferential direction using a maximum two-degree increment for nozzle 3 and three-degree increment for the other 5 nozzles. The circumferential scan distance covered from minus five (-5) degrees to positive three hundred and sixty five (+365) degrees, yielding a ten degree overlap. De-mineralized water was used as the coupling agent between the transducer face and the nozzle surface.

The blade probe uses two 5.0 MHz elements, configured for forward scatter Time of Flight Diffraction (TOFD), to produce a 50° longitudinal wave looking in the axial direction of the nozzle. The TOFD technique provides detection and sizing information for inside surface and outside surface connected axial and circumferential flaws contained within the nozzle walls. Flaw detection is identified by signal loss from the lateral wave or back-wall response, as well as detection of crack tip responses.

In addition to the UT examination, the inspectors verified by observation by way of video and discussions with the examination personnel that a Liquid Penetrant (PT) examination was performed on CRD nozzle 3 from the outside surface, which included the weld and adjacent base material in accordance with Framatome ANP Nondestructive Examination Procedures - Liquid Penetrant Examination of Reactor Vessel Head Penetration J-Groove Welds, 54-ISI-250-00, Rev. February 8, 2002 and Visible Solvent Removable Liquid Penetrant Examination Procedure, 54-PT-6-07, Rev. August 3, 2000.

The inspectors reviewed the Framatome ANP procedures and the licensee's inspection plan, which were approved by Sequoyah management, for use for the RPV inspection. The inspectors noted that the approved acceptance criteria and/or critical parameters for RPV leakage were applied in accordance with the procedures.

3) Verification that the licensee was able to identify, disposition, and resolve deficiencies.

The inspectors reviewed the vendor procedures controlling the PT and UT examination techniques and determined that they provided adequate guidance to ensure that they would be able to identify, disposition, and resolve relevant deficiencies in the RV head penetration materials. The inspectors verified that the examination results for each nozzle were individually documented.

4) Verification that the licensee was capable of identifying the primary water stress corrosion cracking (PWSCC) phenomenon described in the bulletin.

The inspectors reviewed, via remote camera/video, the licensee's 6 UT & 1 PT examinations from below the RPV during the U1C12 outage; observed the licensee conduct the examination; discussed the examination with the licensee examiners prior to, during, and following the examination; and verified the qualifications of the licensee examination personnel. The UT NDE inspection techniques had been previously demonstrated capable of detecting PWSCC type manufactured cracks as well as cracks as documented in MRP 2002-107, Overview of RPV Head Penetration Inspection Demonstration Program, November 1, 2002. The inspectors concluded that the licensee's visual examination was adequate to identify potential leakage resulting from PWSCC cracking of reactor head penetrations.

5) Evaluate condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions).

See Paragraph 4OA5.1 of Inspection Report 05000327,328/2003003 for additional discussion of this area.

6) Evaluate ability for small boron deposits, as described in Bulletin 2001-01, to be identified and characterized.

See Paragraph 4OA5.1 of Inspection Report 05000327,328/2003003 for additional discussion of this area.

7) Determine extent of material deficiencies (associated with the concerns identified in the bulletins) which were identified that required repair.

All UT and PT NDE results indicated that there were no material deficiencies in any material reviewed. The PT on nozzle 3 was also independently reviewed by an outside consulting firm which concluded that the results were negative for any deficiencies that may have required any repair. No examples of RPV leakage or material deficiencies were identified during any NDE examinations.

8) Determine any significant items that could impede effective examinations.

No significant items that could impede the examination process were noted during observation of the NDE examinations.

9) Determine the basis for the temperatures used in the susceptibility ranking calculation.

See Paragraph 4OA5.1 of Inspection Report 05000327,328/2003003 for additional discussion of this area.

.2 (Closed) Unresolved Item (URI) 50-327, 328/02-02-05: Corrective Actions Related to the Apparent Failure of Rod Cluster Control Assembly (RCCA) L-11 to Properly Insert.

This URI was opened following a reactor trip of Unit 2 on May 31, 2002, in which the RCCA at location L-11 apparently failed to properly insert into the core. The inspectors reviewed licensee PER 02-008088-000 and plant records of RCCA L-11 performance during six reactor shutdowns since the URI was opened.

The inspectors determined that PER 02-008088-000, provided an analysis that bounded the safety significance of the event. In this PER, the licensee addressed the following items:

- The impact on shutdown margin if all RCCAs stuck at 12, 20, and 30 steps and adequate margin for 12 and 20 steps. For 30 steps, subcriticality was demonstrated.
- Fuel manufacturing records showing that L-11 fuel assembly was not manufactured with excessive bowing or twist.
- Vendor information showing that RCCA finger tip swelling was not the likely cause of the occurrence, given drag testing results, age of RCCA, and trace characteristics for three RCCA drops.
- Mechanical interference or debris in guide tube as a potential likely cause.

The inspectors concluded that the PER adequately addressed the issue. No findings of significance were identified; therefore, this URI is closed.

.3.0 Unit 1 Steam Generator Replacement Project (SGRP)

.3.1 Engineering Preparation and Implementation for the SGRP

a. Inspection Scope

The inspectors reviewed engineering preparations including: selected Engineering Packages (EPs), calculations, analyses, drawings, Work Package and Inspection Reports (WP&IRs) and Design Change Notice (DCNs) for the Steam Generator Replacement Project (SGRP) in order to assess adequacy and completeness. The inspectors also held discussions with SGRP management to obtain a greater understanding of the entire project scope.

b. Findings

No findings of significance were identified.

.3.2 Project Management Organization and Staffing

a. Inspection Scope

The inspectors reviewed the SGRP organization including: controls for contractor oversight and interface, plans for identifying and resolving non-conforming conditions, and plans for implementing quality assurance requirements in order to assess adequacy. To evaluate the SGRP project management and organization, the inspectors reviewed various documents, including staffing reports, forecasts, and administrative procedures; and conducted interviews with various personnel in differing organizations.

b. Findings

No findings of significance were identified.

.3.3 SGRP Procedures and Documentation

a. Inspection Scope

To verify adequacy of SGRP procedures, the inspectors reviewed the “Special Processes Manual (SPM) for Sequoyah Nuclear Plant Unit 1, Steam Generator Replacement Project,” which contained procedures for welding and NDE matrices, procurement and control of welding filler materials, welder performance qualification standards, general welding standards, nondestructive examination standards, post weld heat treatment standards, weld documentation requirements, and welding procedure specifications.

Other procedures reviewed and compared with regulatory requirements and codes that were utilized during the SGRP are listed in the attachment.

b. Findings

No findings of significance were identified.

.3.4 Pre-Service Baseline Examination, Eddy Current (ET) of Replacement Steam Generators

a. Inspection Scope

The inspectors reviewed the baseline eddy current data as issued in the “Report on Steam Generator Eddy Current Examination and Repair, TVA Sequoyah Nuclear Power Plant Unit 1, Pre-Service Inspection, MRS-FSR-1216-TVA, Rev. 0, January 2003."

The inspectors reviewed aspects of the examination program for the Westinghouse Model 57AG steam generators, which included multifrequency bobbin testing for indications of degradation, loose parts, dents, etc., and motorized pancake/plus point testing for detection of anomalies in the tube sheet region, further evaluation of detected bobbin indications, and a plus point examination of the U-bend region for the first three rows. The inspection was to determine if all examinations were in compliance with NRC Regulatory Guide 1.83, TVA Sequoyah TS, and Sections V and XI of the 1995 ASME Code through the 1996 Addenda. The inspectors reviewed results of the examinations of the four steam generators, which required a total of 20 tubes to be plugged (4 in SG A, 6 in SG B, 5 in SG C, and 5 in SG D).

b. Findings

No findings of significance were identified.

.3.5 Review of SGRP Lifting and Transportation Program

a. Inspection Scope

The inspectors reviewed the adequacy of the SGRP lifting programs as described in 24370-EP-004, Rev.2, assuring that it was prepared in accordance with regulatory requirements, appropriate industrial codes, and standards; and verified that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors examined the SGRP lifting equipment necessary to perform steam generator rigging and transport, design evaluation/erection/use and disassembly of the Outside Lift System (OLS), removal of the shield building dome concrete/containment vessel steel, and steam generator compartment concrete and load drop protection. The procedures and lifting documents that controlled the SGRP which were reviewed by the inspectors are listed in the attachment.

b. Findings

No findings of significance were identified.

.3.6 Haul Route Load Test and Evaluation

a. Inspection Scope

The inspectors reviewed the adequacy of the haul route evaluation, placement of temporary protection for plant commodities and haul route upgrades required to prepare the haul route for load testing and transport of the steam generators. The inspectors reviewed to verify that they had been prepared and tested in accordance with regulatory requirements, appropriate industrial codes, and standards.

The inspectors discussed the results of the transport path load testing with SGRP engineering personnel in order to determine that, where minor discrepancies occurred, these areas had been properly corrected with appropriate material.

b. Findings

No findings of significance were identified.

.3.7 Observation of SG Lifting and Movement

a. Inspection Scope

The inspectors observed various portions of the old steam generators (OSG) being lifted from the steam generator (S/G) cubicle through the temporary penetrations in the reactor building to the hydraulic trailer transporter. The inspectors also observed various portions of the replacement steam generators (RSG) being lifted from the hydraulic trailer transporter into containment. During these observations the inspectors performed visual inspections of the Outside Lift System and the hydraulic trailer transporter. For the task of rigging and movement of the SGs, the inspectors reviewed the WP&IRs for content, technical adequacy and to verify that appropriate line items had been signed off and that required pre-lift equipment inspections had been performed and documented in the enclosures provided. This review was also to verify that Industry Experience was utilized and reflected in the procedures.

b. Findings

No findings of significance were identified.

.3.8 Review and Walkdown on Engineering Preparation

a. Inspection Scope

The inspectors reviewed the installation of temporary pipe restraints; modification of the existing restraints; removal of snubbers and beams; and pipe cuts in order to verify that the engineering preparation for the removal of OSGs was in accordance with the engineering packages and drawings for the SGRP.

The inspectors performed a walk-through inspection of the Containment Building, to observe the cut reactor coolant piping from the S/G nozzles and observe housekeeping conditions around the work area. The inspectors looked at corrective actions to verify that problems identified early in the outage, associated with cleanliness, housekeeping, and control of materials and tools around the work area, were resolved.

b. Findings

No findings of significance were identified.

.3.9 Interference Removal and Restoration

a. Inspection Scope

The inspectors observed the vicinity of all S/G cavities before the lifting operation began to assure that the licensee had removed all of the interferences and restraints. The inspectors reviewed procedures which controlled the removal and re-installation of interferences. Provisions for the temporary storage of removed interference items were also reviewed. In addition, the inspectors observed portions of the removal of interferences including piping, steam generator restraints, snubbers, and lateral supports.

After the installation of the RSGs, the inspectors observed various portions of the re-installation of various items including (but not limited to); piping, steam generator restraints, snubbers, lateral supports, and instrumentation tubing to assure that they had been installed per the engineering drawings and procedures.

b. Findings

No findings of significance were identified.

.3.10 Special Procedures for Welding and Nondestructive Examination

a. Inspection Scope

1. Reactor Coolant System (RCS), Feedwater (FW), Main Steam (MS) and Steel Containment Vessel (SCV) Fit-Up and Welding

The inspectors conducted inspections of the fit-up and welding activities involving primary RCS, MS and FW piping, and the SCV. Activities were compared with appropriate Codes and Standards as listed in the attachment of this report and the Bechtel Special Process Manual (SPM).

The inspectors verified the as-built configuration and held discussions with cognizant engineering personnel. This inspection verified that the amount of movement and as-built "gap" associated with the cutting and fit-up of the RCS piping, for all four SGs, was within specification allowable tolerance requirements and applicable codes.

The inspectors observed the automatic welding of RCS hot-leg and cold-leg piping connections to all four steam generators RCS nozzles, and the connections for the MS and FW systems via video monitors on the control panels located outside containment. The inspectors verified that the operator at the weld site and the operator at the control panel were in constant communication through head sets. The inspections during welding operations were to verify that the welding machine settings were being maintained within the qualified welding parameters listed in the welding procedure specification.

2. Training and Qualification

The inspectors observed work, examined selected records, and reviewed procedures to evaluate the licensee's training and qualification efforts for personnel performing cutting, machining, welding, and NDE. The inspectors also reviewed the programs and compared them with the regulatory requirements and codes that were utilized during the SGRP, as discussed previously in this report.

NDE Examiner/QC Inspector Qualification Certification and Visual Acuity Records Examined

Examiner	Method-Level
MH, DM, WT, JE, MG	RT-II
MP, PB, JP, PV	UT-II

Welder/Welding Operator Qualification Records

Welder/Welding Operator Symbol
M-71, M-55, M-50, M-35, M-29, M-19, M-63, M-59, M-33

3. Nondestructive Examination (NDE) & Post-Weld Heat Treatment

The inspectors evaluated the licensee's welding, Nondestructive Examination (NDE) and Post-Weld Heat Treatment (PWHT) activities related to the SG replacement, by conducting an inspection of the records for calibration, examination results, fit-up, welding, certifications of personnel and materials, and NDE (including review of radiographs and UT data).

To verify that the radiographs showed the welds to be free of rejectable indications, the inspectors reviewed the radiographs of intermediate (root, hot pass, 1/3 and 2/3) and completed Reactor Coolant System (RCS), Feedwater (FW) and Main Steam (MS) welds to verify proper penetrameter type, size, placement, and sensitivity as well as film density, identification, quality, and weld coverage. The welds reviewed for this work effort was as follows:

Weld No.	Size	NDE Type
1-RC-11-1	S/G 2 Safe end to elbow	RT
1-RC-03-1	S/G 1 Safe end to elbow	RT
1-RC-13-1	S/G 2 RCS CL	RT
1-RC-18-1	S/G 3 RCS HL	RT
1-RC-19-1	S/G 3 RCS CL	RT
1-MS-021-1	Main Steam to S/G 3 Nozzle/elbow	UT
1-FW-131	S/G 3 to Feedwater	UT
1-RC-02	S/G 1 to RCS HL (elbow to safe-end)	UT

Records reviewed included WP&IRs, Field Welding Check Lists, Filler Material Withdrawal Authorizations, welding filler material Certified Material Test Reports, NDE

Reports (UT and RT), UT consumables certifications, QC inspectors and NDE examiner certification and visual acuity documentation, and certification of visual acuity examiner's qualification. Records were reviewed for completeness, accuracy and technical adequacy. The radiographs were examined for both film quality and acceptability.

b. Findings

No findings of significance were identified.

.3.11 Effect of Steam Generator Replacement Project (SGRP) on Site Security Program

a. Inspection Scope

During the period April 21-23, 2003, the inspectors evaluated the impact of the SGRP on the site physical security program. The inspectors observed protected area and vital area security boundaries affected by SGRP activities and evaluated compensatory measures implemented to control access through breaches in the barriers. The inspectors observed security posts which had been added for SGRP support and to meet the objectives of the site defensive strategy under outage conditions, evaluated security staffing for the outage, interviewed security officers to ascertain their knowledge and understanding of assigned duties, and evaluated the licensee's assessment of the impact of SGRP activities on the site defensive strategy.

The licensee's actions were evaluated against applicable portions of the site Physical Security Plan, security procedures, and the licensee's commitments in response to the NRC's February 25, 2002, Order requiring the implementation of certain interim security compensatory measures.

Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.4 Steam Generator Replacement (SGR) Inspection Overview

This inspection report documents completion of inspections required by IP 50001, Steam Generator Replacement Inspection, some of which were fulfilled by completion of baseline inspection procedures. The table below identifies and correlates specific IP 50001 inspection requirements examined during this inspection period with the corresponding sections of this report.

IP 50001 Section	Inspection Scope	Section of This Report
02.02.a.1.	Design changes and modifications to SSCs described in the UFSAR reviewed for compliance with 10 CFR 50.59	4OA5.4.1
02.02.a.2.	Key design aspects and modifications for the replacement SGs and other Mods. associated with SG replacement	4OA5.4.1

02.02.b.	Engineering design, modification, and analysis associated with SG lifting and rigging	4OA5.3.5
02.02.c.	Radiation protection program controls, planning, and preparation	2OS1, 2OS2
02.02.d.1.	Security considerations associated with vital and protected area barriers that may be affected during replacement activities	4OA5.3.11
02.02.d.2.	Controls and plans to minimize any adverse impact on the operating unit and common systems	1R20, 4OA5.4.2
02.03.a.1.	Special procedures for welding and Non-Destructive Examination (NDE)	4OA5.4.6, 4OA5.3.10
02.03.a.2.	Training and qualifications for welding and NDE personnel	4OA5.3.10
02.03.a.3.	NDE results and work packages for selected welds	4OA5.3.10
02.03.a.4.	Completion of pre-service NDE requirements for welds and baseline eddy current examination of new SG tubes	4OA5.3.10
02.03.b.	Activities associated with lifting and rigging	4OA5.3.5, 4OA5.3.7
02.03.c.	Major structural modifications to facilitate SG replacement	4OA5.4.6
02.03.d.	Restoration of temporary containment opening and containment leakage testing	4OA5.4.5
02.03.e.1.	Establishment of operating conditions including defueling, RCS draindown, system isolation and safety tagging	1R20, 4OA5.4.3
02.03.e.2.	Implementation of radiation protection controls	4OA5.4.3, 2OS1, 2OS2
02.03.e.3.	Controls for excluding foreign materials in the primary and secondary side of the SGs and in related RCS openings	1R20, 4OA5.4.3
02.03.e.4.	Installation, use, and removal of temporary services	1R20, 4OA5.4.3
02.03.f.	Radiological safety plans for temporary storage or disposal of retired SGs and components	2OS1, 2OS2, 2PS2
02.04.1.	Post-installation containment testing	1R19, 1R20 4OA5.4.4
02.04.2.	Post-installation inspections and verifications	4OA5.4.4
02.04.3.	RCS leakage testing	4OA5.4.4
02.04.4.	SG secondary side leakage testing	4OA5.4.4
02.04.5.	Calibration and testing of instrumentation	1R22, 4OA5.4.4
02.04.6.	Procedures for equipment performance testing	1R19, 4OA.4.4
02.04.7.	Preservice inspection of new welds	4OA5.3.10

.4.1 SGR Design Changes and Modifications - 10 CFR 50.59 Review

a. Inspection Scope

As required by IP 50001, Sections 02.02.a.1 and 02.02.a.2, the inspectors reviewed design change notice, D-20672, SG Vessel Replacement/RCS Support Modifications, Revision A, for key design aspects and modifications of the replacement steam generators and verified that changes to the facility as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59. To complete these reviews, the inspectors used IP 71111.02 as guidance.

b. Findings

No findings of significance were identified.

.4.2 SGR Controls to Minimize Adverse Impact on Operating Unit

a. Inspection Scope

As required by IP 50001, Section 02.02.d.2, throughout this inspection period, the inspectors reviewed and monitored licensee controls and plans to minimize any adverse impact on the operating unit and common systems. Specific areas reviewed included:

- Modifications to the auxiliary building secondary containment envelope (ABSCE)

- Abnormal Operating Procedure AOP-M.07, RSG Heavy Load Drop, Rev. 0

b. Findings

No findings of significance were identified.

.4.3 SGR Operating Conditions, Radiation Protection Controls, Foreign Material Exclusion, and Temporary Services

a. Inspection Scope

As required by IP 50001, Section 02.03.e, throughout this inspection period, the inspectors routinely inspected the following activities as they occurred:

- Establishment of operating conditions including defueling, RCS draindown, and system isolation and safety tagging/blocking.
- Implementation of radiation protection controls.
- Implementation of controls for excluding foreign materials in the primary and secondary side of the SGs and in the related RCS openings.
- Installation, use, and removal of temporary services directly related to steam generator replacement activities.

b. Findings

No findings of significance were identified.

.4.4 Post-Installation and Verification and Testing Inspections

a. Inspection Scope

As required by IP 50001, Section 02.04, the inspectors reviewed and monitored the following nine post-installation and activities. Some of these tests were reviewed as part of the baseline inspection program.

- 1-STI-088-156.0, Primary Containment Vessel Post-Modification Pressure/New Weld Leakage Inspection Test, Revision 0
- 1-SI-SFT-065-001.B, Emergency Gas Treatment System Annulus Vacuum Draw Down Test Unit 1 Train B, Revision 5
- 0-SI-OPS-000-187.0, Containment Inspection, Revision 21
- 0-SI-SFT-072-138.0, Containment Spray-Spray Nozzle Test, Revision 3
- 1-RT-MOD-520-001.0, Steam Generator Replacement Post - Modification Test Sequence, Revision 0
- 1-SI-SXI-068-201.0, Leakage Test of the Reactor Coolant Pressure Boundary, Revision 4
- 0-SI-SXI-000-201.0, American Society of Mechanical Engineers (ASME) Section XI Inservice Pressure Test, Revision 11
- 1-SI-ICC-003-042.4, Channel Calibration of Steam Generator 1 Level Channel IV, Rack 12, Loop L-3-42, Revision 6
- 1-RT-MOD-520-002.0, 3 % Load Step Change, Revision 0

b. Findings

No findings of significance were identified.

.4.5 Containment Restoration Activities

a. Inspection Scope

The inspectors reviewed containment restoration activities associated with two temporary construction openings in the containment shield building dome and steel containment vessel, which were approximately 20 feet by 40 feet, for the Unit 1 steam generator replacement project.

The inspectors reviewed procedures for installation of concrete reinforcing steel and Bar-Lock splices and procedures for control of concrete placement activities. The inspectors observed installation of concrete reinforcing steel and installation of Bar-Lock splices to determine if the work was completed in accordance with requirements shown on design drawings. The inspectors reviewed results of quality control acceptance testing performed on materials (cement, fine and coarse aggregate, water, and admixtures) selected for batching the concrete and results of qualification testing for the Bar-Lock splices. The inspectors also reviewed the concrete mix data to ensure that selected trial mix met concrete design strength requirements, and that Quality Control (QC) acceptance criteria specified in the procedures for the plastic concrete were based on the trial mixes.

The inspectors reviewed the restoration of the steel containment vessel welding procedures and weld process sheets. The inspectors observed in-process welding activities, reviewed the licensee's program for control of weld filler materials, and observed preliminary preparations for performance of radiographs of the completed welds on the steel containment vessel. Welding acceptance criteria were specified in Bechtel General Welding Standard GWS-1 and American Welding Society AWS D1.1, Structural Welding Code.

b. Findings

No findings of significance were identified.

.4.6 Steam Generator Enclosure Roof Restoration

a. Inspection Scope

The inspectors reviewed drawings and observed work activities for restoration of the steam generator enclosure roofs after replacement of the Unit 1 steam generators. The inspectors examined the structural steel support frames installed to support the plugs removed from the enclosure roofs for access to replace the steam generators. Acceptance criteria for the restoration of the enclosure roofs were specified on the design drawings.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

On July 8, 2003, the resident inspectors presented the inspection results to Mr. Mike Lorek and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided nor retained during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

J. Bajraszewski, Licensing Engineer
T. Carson, Maintenance and Modifications Manager
H. Cothran, Steam Generator Manager
E. Freeman, Operations Manager
J. Gates, Business & Work Performance Manager
M. Gillman, Operations Manager
C. Kent, Radcon/Chemistry Manager
D. Koehl, Plant Manager
M. Lorek, Assistant Plant Manager
D. Lundy, Site Engineering Manager
R. Purcell, Site Vice President
R. Rogers, Design Manager
P. Salas, Licensing and Industry Affairs Manager
J. Smith, Site Licensing Supervisor
K. Stephens, Security Manager

NRC personnel:

S. Cahill, Chief, Reactor Projects Branch 6
R. Bernard, Region II, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

None

Closed

50-327,328/02-02-05	URI	Corrective Actions Related to the Apparent Failure of RCCA L-11 to Properly Insert (Section 4OA5.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection

Fire Impairment NFOR 2003A0118
SPP-10.9, Control of Fire Protection Impairments

Section 1R08: Inservice Inspection

Visual Examination Procedure for ASME Section XI Preservice and Inservice, N-VT-1, Rev. 33
Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, N-PT-9, Rev. 25
Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, N-UT-76, Rev. 4
TVA Procdure N-UT-33, Manual Ultrasonic Examination of Static and Centrifugally Cast Stainless Steel Piping Welds, Rev. 9
Ultrasonic Examination of Socket Welds to Detect Cracks Initiating at the Pipe I.D. Beneath the Socket Weld heel to Toe Area, N-UT-60, Rev. 1
ASME Section XI IWE/IWL Containment Inservice Inspection Program (CISI) Unit 1 and 2, 0-PI-DXI-000-116.1 Rev. 3
Snubber Visual Inspection (Hydraulic and Mechanical), 1-SI-MIN-000-001.0, Rev. 17
Snubber Program, 0-TI-DXX-000-009.0. Rev. 2
ASME Section XI ISI/NDE Program Unit 1 and 2, 0-SI-DXI-000-114.2, Rev. 17
Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, N-UT-64, Rev. 5.
Sequoyah Nuclear Plant Unit 1 Cycle 12 Inservice Inspection Scan Plan, Revision 0
WO:PER 03-002841- Missing portion of weld on pipe support, 1-CVCH-361
Self-Assessment Report CRP-ISO-02-001, Evaluation of TVAN Augmented Examination Program
Nuclear Assurance Audit Report SSA0204, ASME Section XI, Feburary 2003
Nuclear Assurance Audit Report SSA0006, ASME Section XI, Feburary 2001

Section 1R12: Maintenance Effectiveness

<u>Emergency Diesel Generators</u>

Performance Data (including unavailability and unreliability data) for Unit 1 Emergency Diesel Generator Trains “A” and “B,” from March 2001 through January 2003
Performance Data (including unavailability and unreliability data) for Unit 2 Emergency Diesel Generator Trains “A” and “B,” from March 2001 through January 2003
Operations Log from January 2001 through April 2003 with respect to DG entries
Performance Indicator packages for DGs for March and April 2002
CDEF 1439, Starting Air System for DG 2B-B
CDEF 1517, Exhaust fan 1A tagged with the discharge damper open
CDEF 1519, DG 1A-A exceeded unavailability criteria
Maintenance Rule Expert Panel Meeting Minutes for: March 5, 2002; March 21, 2002; May 30, 2002; September 4, 2002; and October 17, 2002

System Status Report for Unit 1,2 Standby Diesel Generators for second-fourth quarters of FY 2002 and first quarter of FY 2003
PER 01-006160-000, EDG 2B2 engine lube oil analysis 1 ppm silver in the engine oil
PER 01-007184-000, Uncontrolled Blowdown of the DG Starting Air System
PER 01-010452-000, 2AA DG failed to start from local idle start signal
PER 01-011284-000, During the performance of 1-SI-OPS-082-024 for DG 1B-B, the A/D converter was installed with the incorrect computer
PER 01-017257-000, Evidence of deterioration of the silencer/piping of the DG exhaust system
PER 02-011438-000, During the performance of 0-SO-82-3, the 2A-A DG LRX2A relay would not reset initially
PER 02-012966-000, 1BB DG engine #2 south PVC blew down for approximately ten seconds during the idle start for 1-SI-OPS-082-007B
PER 03-001442-000, initiated to perform common cause analysis for the issues identified during the 2-year DG outages and to capture lessons learned
PER 03-004028-000, Engine 2B2 cylinder temperature exceeded the operating limit
PER 03-004029-000, Temperature reading for engine 2B2 thermocouple #7 reads 200 degrees higher than adjacent thermocouples and fluctuates approximately 50 degrees

6.9 kV Breakers

PER 01-009568-000, Revised to consolidate vendor material condition problems
PER 03-005396-000, While installing 6.9 kV Siemens vacuum breaker into the 6.9 kV shutdown board 1A-A spare compartment 19 for testing, the charging motor failed to operate in test position
System Status Report for Unit 1,2 6.9 kV Electrical Boards for FY 2002 and the first quarter of FY 2003
Operations Log from September 2001 through April 2003 with respect to DG entries
CDE 1559, Loss of refrigeration and glycol flow due to tripping normal supply feeder breaker 1414 to bus 2B and the loss of 6.9 kV common board B
CDE 1338, During the release of clearance 2-30-263, the SIP 2B-B breaker closed and tripped back open in the test position
CDE 1442, The arm on the breaker was over-traveling causing damage to peripheral equipment in the breaker cabinet
CDE 1444, The R-A ERCW pump was inoperable due to breaker problems
CDE 1549, Due to issues with Siemens 6.9 kV breakers, the decision was made to swap the ERCW 1B MCC breaker with a Siemens so that the original breaker could be used in an application that experienced frequent cycles
CDE 1550, Following the swap of the 1B ERCW MCC breaker, while OPS was attempting to rack the breaker in, the breaker tripped free
CDE 1544, A new Siemens breaker was being installed into compartment 22 of 6.9 kV shutdown board 1B-B and tripped during PMT
CDE 1537, Operators were restoring to CSST B, racking breaker 1412 into operate position when the charging spring was heard to energize. When the spring was energized, the breaker closed, sending trip signal to 1414 normal supply breaker
CDE 1533, Power was lost to 6.9 kV unit boards 2B and 2D on Unit 2, resulting in a Unit 2 trip
Maintenance Rule Expert Panel Meeting Minutes for: January 22, 2001; March 22, 2001; April 26, 2001; July 26, 2001; July 26, 2001; August 17 & 21, 2001; and July 10 & 11, 2002

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

0-TI-DSM-000-007.1, Equipment To Plant Risk Matrix
Unit 2 Reactor Operator logs dated 6/5/2003 (Dayshift)
Unit 1 Reactor Operator logs dated 6/18/2003 (Nightshift)

Section 1R14: Personnel Performance During Non-routine Plant Evolutions

AOP-N.05, Earthquake, Revision 8

Section 1R15: Operability Evaluations

UFSAR Section 9.2.7, Raw Cooling Water System
Tech Spec Section 3.7.4, Essential Raw Cooling Water System
PER 03-007360-000
PER 03-007064-000, Functional Evaluation dated 05/23/2003

Section 1R16: Operator Workarounds (OWAs)

AUO Round Deficiency (ARD) 1, Unit 1 Auxiliary Building
ARD 2, Unit 2 Auxiliary Building
ARD 3, Unit 1 Turbine Building
ARD 4, Unit 2 Turbine Building
ARD 5, Control Building
ARD 6, Radwaste
ARD 7, Outside
ARD 8, Con DI

Section 1R19: Post-Maintenance Testing

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
SPP-6.3, Pre-Post-Maintenance Testing, Revision 1
0-TI-PMT-000.000.0, Pre-Post Maintenance Testing Matrices, Revision 10
WO 01-03501-000, Complete package review of work and testing
Unit 2 Reactor Operator logs dated June 8-9, 2003
WO 03-008636-000, Repair/Clean 1-PCV-65-86

Section 1R20: Refueling and Other Outage Activities

0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 17
PER 03-007973-000, Debris in Unit 1 Reactor Vessel

Section 1R22: Surveillance Testing

0-MI-MXX-001-000.0, Ice Condenser Maintenance Inspections, Revision 7
PER 03-007499-000, NRC-Identified use of slightly greater-than-allowed lifting force during ice basket weighing activities
0-MI-MXX-061-003.0, Ice Condenser Maintenance Inspections
1-SI-ICC-003-174.0, Channel Calibration of Steam Generator 1 Turbine Driven Auxiliary Feedwater Level Loop 1-L-3-174, Revision 13
1-SI-ICC-068-339.1, Channel Calibration of Pressurizer Level Channel I, Rack 1, Loop L-68-339 (L-459), Revision 11

Section 1R23: Temporary Plant Modifications

WP&IR C-MHX-026, Material Handling System & Temporary Airlock Door, Revision 0
DCN-20673, Steam Generator Replacement Reactor Building Structural Modifications, Revision A
0-VI-SGR-000-002.0, Bechtel Engineering Package for Steam Generator Replacement, Revision 1
24370-EP-002, Temporary Structures and Commodities for the Removal and Restoration of the Shield Building Concrete Dome and Containment Vessel
Drawing 24370-C-031 SH1, Equipment Hatch Temporary Doors & Fill Structure, Revision 2
TI-50, Air Flow Measurement and Balancing Methods, Revision 17
PER 03-006016-000, NRC-identified inconsistency between DCN D-20673 (Containment Structural Modifications) 50.59 Evaluation and Engineering Package EP-002 (Temporary Structures and Commodities...) Screening Review
DCN-T-13024-A, Determination of the ABSCE Boundary Through Equivalent Length of Pipe to Negate the ABSCE Differential Pressure

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Radiological Control Instructions (RCIs), Guidance Documents

Tennessee Valley Authority Nuclear (TVAN) RadCon Department Procedure (RDCP) RDCP-1, Conduct of Radiological Controls, Revision (Rev.) 2
Radiological Control Instruction (RCI)-1, Radiological Control Program, Revision (Rev.) 62
RCI-14, Radiation Work Permit (RWP) Program, Rev. 28
RCI-15, Establishing and Updating Radiological Signpostings, Rev. 13
RCI-24, Control of Very High Radiation Areas, Rev. 4
RCI-28, Control of Locked High Radiation Areas, Rev. 2
RCI-29, Control of RADCON Keys, Rev. 2
Technical Instruction (TI) 0-TI-NUC-000-002.0, Storing Material in Spent Fuel Pool or New Fuel Vault, Rev. 5
U1C12 Radiological Protection Plan

Radiation Work Permit (RWP) Documents

RWP 03044267, Upper Containment All Areas
RWP 03044273, Under the Reactor Vessel Head
RWP 03000805, Outside Polar Crane Wall -- Fan Rooms, Acc. Rooms, and Raceway
RWP 03014130, Residual Heat Removal (RHR) and Containment Spray Heat Exchanger 1B-B

Problem Evaluation Reports (PERs)

PER 02-014140-000, Workers Improperly Entered an Unsurveyed Area, 11/13/2002
PER 02-014226-000, RADCON Technician Reached Across a Contaminated Area Boundary, 11/15/2002
PER 02-014424-000, Some Radiation Workers Were Observed Not Placing Hand-carried Items in the Tool Monitor, 11/22/2002
PER 03-002129-000, A Contractor Employee Exited Unit 1 Upper Containment Without Bagging Items Taken into the Contaminated Area, 03/04/2003
PER 03-002563-000, An Individual Removed a Test Gauge from Inside the Resin Hopper on Elevation 714, Contaminating the Surrounding Area and Himself, 03/14/2003

Section 2OS2: As Low As Reasonably Achievable

Procedures, Instructions, Lesson Plans, and Manuals

TVAN RadCon Department Procedure (RCDP-4), Personnel Inprocessing and Dosimetry Administrative Processes, Rev. 4
TVAN RCDP-6, Special Dosimetry Operations, Rev. 2
SQN RCI -5, Calibration and Operation of the Thermo Eberline Gamma Tool Monitor, Effective Date 6/18/02
SQN RCI-15, Establishing and Updating Radiological Signposts, Rev. 13
RadCon Management Directive Field Operations (RMD) FO-02 Radiation and Contamination Surveys, Rev. 13
TVAN RCDP 3, Administration of Radiation Work Permits (RWPs), Rev.2

Records and Radiation Surveys

ALARA Planning Report (APR) 2003-47 Site Outage Dose Goal, 02/03/03
APR 2003-38 Under Vessel Head Inspection, 04/10/03
APR 2003-46 Steam Generator Secondary Side Work
APR 2003-44 Scaffold
APR 2003-26 Shielding
Radiation Survey Numbers (Nos.) 04150327, 04150328, 04110310, 0413037, 0413034, 0413035, 04130316, and 04170310

Radiation Work Permits

RWP-2140 Rig Out OSG 1, 2, 3,
RWP-2090 Transport and Off load OSGs from Downending area to OSGSF
RWP-2091 Install RCS Shield Plates

PERs and Self-Assessments

PER 03-003851 Access to Excess Letdown Heat Exchanger (HX)
PER 03-003852 Individual Signed in on Wrong RWP
PER 03-003856 Clarification of Work Area in Excess Letdown/Regenerative HX
PER 03-003860 Excess Personnel Dose Associated with Locating Valves
PER 03-005410 Handling of Dry Active Waste Bag Dose Concern
Nuclear Assurance - TVAN - Audit Report NO. SSA0102- Plant Support Functional Area Audit 07/27/01
TVA SQN Annual ALARA Report Fiscal Year 2002

Section 2PS2: Radioactive Material Processing and Transportation

Procedures, Manuals, and Guides

Common -System Operating (0-SO)-77-29, Waste Processing, Rev. 9
RadWaste Technical Procedure (RWTP)-100, Radioactive Material / Waste Shipments, Rev. 0
RWTP-101, 10 CFR 61 Waste Characterization, Rev. 0
Standard Program and Processes (SPP)-3.1, Corrective Action Program, Rev. 4

Shipping Records and Radwaste Data

SNP 03-103 Surface Contaminated Object (SCO), Contaminated Equipment, 01/08/03
SNP 03-308 Low Specific Activity (LSA), Laundry, 03/22/03
SNP 03-403 SCO, Valves, 04/01/03
SNP 03-413 SCO, Reactor Head Inspection Equipment, 04/09/03
SNP 03-424 Dry Active Waste, 04/24/03
SNP 03-508 LSA, Laundry, 05/14/03
Waste Stream Analytical Results, dated 03/19/03
Training Certificates, dated 9/26/02, for Authorized Radwaste Shippers

PERs and Audits

Audit Report SSA0102 - Plant Support Functional Area Audit, 07/21/01
PER 02-008251, Incorrect Transport Index Recorded on Shipping Papers, 7/10/02
PER 02-008567, Process Control Program not Reviewed by Plant Operations Review Committee 07/16/02
PER 02-013675, Drum for Hazardous Waste (Paint) Missing from Sandblast Shop, 10/30/02
PER 03-002093, High Integrity Container (HIC) Lifting Device Would Not Release HIC by Remote Control, 03/03/03
PER 03-006944, New Waste Stream Analysis Required for Filter Material, 05/08/03

PER 03-006935, Training Provided to Three Works for Packing and Loading Radioactive Materials Did Not Include Security Awareness, 05/08/03

Section 4OA1: Performance Indicator Verification

Section 4OA1.1: Barrier Integrity Cornerstone

0-SI-OPS-068-137.0, RCS Water Inventory
SPP-3.4, Performance Indicators for NRC Reactor Oversight Process

Sections 4OA1.2 and 4OA1.3: Occupational Radiation and Public Radiation Safety Cornerstones

Procedures

SPP-3.4, Performance Indicator for NRC Reactor Oversight Process, Rev. 0, 04/03/2002
SNP Desktop Guideline for Identification and Reporting of NEI 99-02 Performance Indicators for Occupational Exposure Control Effectiveness
Common Technical Instruction Chemistry (0-TI-CEM)-000-001.3, Primary Chemistry Specifications, Rev. 13

PERs

02-013539-000, Individual Entered RCA Without TLD badge, 10/27/2002
02-014509-000, Emergent Activities Are Not Being Reviewed and Appropriately Reported/Communicated to the RADCON Staff, 11/26/2002
03-001633-000, Valid ED Dose Alarms and Dose Rate Alarms Not Being Reported via PER Initiation, 02/18/2003
PER 02-013073, Effluent Monitor 0-RM-90-134/141 Inoperable, 10/11/02
PER 02-013472, High Radiation Alarm on Effluent Monitor 0-RM-90-212, 10/25/03
PER 02-014224, Increase in Gaseous Effluent during October 2002 due to Unit 2 Fuel Leak, 11/19/02
PER 02-015201, Instrument Malfunction on Monitor 1-RM-90-120/121, 12/17/02
PER 03-002082, Incorrect Value for Instrument Background Count on Effluent Monitor 0-RM-90-122 Used in Liquid Effluent Batch Release Permit, 03/04/03

Plant Records

Individual RCA exit doses exceeding 100 mrem between 10/01/2002 and 04/16/2003
2002 Annual Radioactive Effluent Release Report
Monthly 10 CFR 50 Appendix I Dose Calculations for Liquid and Gaseous Effluents for the Months of October 2002 through March 2003

Section 4OA1.4: Emergency Preparedness Cornerstone

Radiological Emergency Preparedness Audit SSA0206, dated 8/06/2002.

Section 4OA5: Other Activities

Section 4OA5.1: NRC Temporary Instruction (TI) 2515/150

Visual Examination for Leakage of PWR Reactor Head Penetrations, N-VT-17, Rev. 2
Framatome ANP Nondestructive Examination Procedure - Remote Ultrasonic Examination of Reactor Head Penetrations, 54-ISI-100-09, Rev. September 9, 2002
- Change Authorization (CA) No. FRA-03-010
- CA No. FRA-03-004

Framatome ANP Nondestructive Examination Procedure - Liquid Penetrant Examination of Reactor Vessel Head Penetration J-Groove Welds, 54-ISI-250-00, Rev. February 8, 2002
- CA No. SQ-03-002
- CA No. FRA-02-004

Framatome ANP Nondestructive Examination Procedure - Visible Solvent Removable Liquid Penetrant Examination Procedure, 54-PT-6-07, Rev. August 3, 2000
- CA No. SQ-03-001
- CA No. FRA-01-007
- CA No. FRA-03-008
- CA No. FRA-02-017
- CA No. FRA-03-001
- CA No. 54-CA-03-FRA-001

Framatome ANP NDE Administrative Procedure - Written Practice for Personnel Qualification Ultrasonic Method, 54-ISI-21-31, Rev. July 29, 2002
- CA No. FRA-03-007

Framatome ANP Nondestructive Examination Procedure - Remote Ultrasonic Examination of Reactor Vessel Head Vent Line Penetrations, 54-ISI-137-02, Rev. March 19, 2003
- CA No. FRA-03-009

MRP 2002-107, Overview of RPV Head Penetration Inspection Demonstration Program, November 1, 2002
U1C12 Reactor Bare Head Visual Inspection Disposition of the Boron Deposits found on Penetration 03 and Near the Head Vent

Section 4OA5.3: Unit 1 Steam Generator Replacement Project

ASME Section III (Div 1)	1989 Edition with 1995 Addenda for RSG
ASME Section III (Div 2)	1989 Edition with no Addenda
ASME Section V	1989 Edition
ASME Section IX	Latest edition in effect at time of welding procedure qualification
ASME Section XI	1998 Edition and code case N-416-1
ANSI B31.7	1969 with 1970 Addenda
AWS D1.1	Structural Welding Code-Steel (1972)
AWS D1.4	Structural Welding Code-Reinforcing Steel (1998)

AISC	1969, 7th Edition for supplementary steel design & section properties
ASCE 7-95	Minimum Design Loads for Buildings and Other Structures
USAS B31.7	Nuclear Power Piping Code, 1969 Edition with 1970 Addenda
NCIG-01	Rev. 2 for Structural Weldments designed in accordance with AISC/AWS
NQA-1	Quality Assurance Requirements for Nuclear Facility Applications, Subpart 2.15, Quality Assurance Requirements for Hoisting, Rigging, and Transporting of Items for Nuclear Plants, Section 5.3.1(a)
SNT-TC-1A	Personnel Qualification and Certification in Nondestructive Testing
NUREG 0612	Control of Heavy Loads at Nuclear Power Plants, 1980
NRC Bulletin 96-02	Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment, April 1996

Section 4OA5.4: Steam Generator Replacement (SGR) Inspection Overview

Specifications and Procedures

Specification No. 24841-120-C-321, Technical Specification for Purchase of Ready-Mix Concrete Qualified as Safety Related, Revision 7
TVA General Engineering Specification G-2 Plain and Reinforced Concrete, Revision 7
Specification No. 24370-C-101, Technical Specification for Material Testing Services, Revision 2
Specification No. 24370-C-311, Technical Specification for Purchase of Bar-Lock Rebar Couplers, Revision 1
Specification No. 24370-C-312, Technical Specification for Installation of Bar-Lock Rebar Splices, Revision 0
Specification No. 24370-C-321, Technical Specification for Purchase of Safety Related Ready-Mixed Concrete, Revision 1
Specification No. 24370-C-303, Technical Specification for Purchase of Reinforcing Steel, Revision 0
Specification No. 24370-C-601, Technical Specification for Qualification of Bar-Lock Coupler System for Use in Nuclear Safety Related Applications, Revision 0
Specification No. 24370-C-602, Technical Specification for Qualification Testing of Bar-Lock Mechanical Rebar Splices, Revision 2
Specification No. 24370-C-321, Technical Specification for Purchase of Safety Related Ready-Mixed Concrete, Revision 1
Construction Procedure CP-C-1, Concrete Operations, Revision 2
Construction Procedure CP-C-13, Bar-Lock Rebar Splices, Revision 0
Drawing Number 24370-C-037, Shield Building Dome Concrete Restoration, Revision 1
DCN D-20673-A to Drawing Number 1-41N729-2, Steam Generator Enclosure Roof Repair, Revision 3
DCN D-20673-A to Drawing Number 48N405, Welding Details for Restoration of Steel Containment Vessel Dome, Revision 1
Bechtel General Welding Standard GWS-1, Revision 9, Amendment 2
American Welding Society AWS D1.1, Structural Welding Code, 1998 Edition

Quality Records

Concrete Mixer Uniformity (ASTM C-94) tests
Concrete Mix Design data
Result of testing performed on concrete materials: cement (ASTM C-150), concrete admixtures, AEA-92 Air Entraining lot number 246776 and Eucon WR Water Reducer lot number 272269, fine aggregate (ASTM C-33), and number 57 coarse aggregate (ASTM C-33)
Betchel nonconformance report numbers NCR N1- 068, Inadequate concrete cover over rebar on the concrete dome
Betchel nonconformance report numbers NCR N1- 068, Inadequate concrete cover over rebar on the concrete dome
INEEL Report titled Qualification of the Bar-Lock Coupler for use in Nuclear Safety-Related Applications: Mechanical Testing Program and Performance Analysis dated December, 2003
Bechtel Welding Procedure Specification P1-AT-Lh, Revision 5
PER 02-008088-000, Potential degraded/nonconforming (GL 91-18) condition of control rod system following apparent failure of RCCA L-11 to properly insert
PEDS plots of RCCA L-11 position (Point 2C0030A) during six reactor shutdown events
Evaluation of A(1) Status for Rod Control System 085 RCCA L11 Indications Following Rx Trip On 5/31/2002
PER 01-011654-000, Operating Experience/Implementation: Westinghouse Letter 01TV-G-065
PER 02-006086-000, Unit 2 Reactor Trip - Loss of Stator Cooling Water, Revision 1
Rigging International (RI) Procedure P-2523-20, Haul Route Load Test, Rev. 1
RI Procedure P-2523-30, Dome Plug Removal, Rev. 5
RI Procedure P-2523-40, Offload and Transport to Temporary Storage Four (4) Replacement Steam Generators, Rev. 4
RI Procedure P-2523-42, Lift & Downend OSG's, Rev. 4
RI Procedure P-2523-44, Load OSG on Transport, Transport to OSGSF & Offload, Rev. 4
RI Procedure P-2523-46, Load RSG on Transporter, Transport to Containment & Position to Upend, Rev. 3
RI Procedure P-2523-48, Upend, Lift & set RSG's, Rev. 5
RI Procedure P-2523-48, Upend, Lift & Set RSGs, Rev. 4
RI Drawing No. 2523-212 sheets 1/2 & 2/2, Haul Route Load Test
RI Drawing No. 2523-213, 12-Lane Transporter Haul Route Load Test
RI Drawing No. 2523-214 sheet 1/2 & 2/2, OLS Test Load Assembly and Rigging Details General Arrangement
RI Load Test Certification for Spreader Beams and Spreaders, 10/23/2002
RI Drawing No. 2523-230 sheet 1/4, Handling SGs General Arrangement
RI Drawing No. 2523-232 sheets 1/2 & 2/2, S/G Load Path
RI Drawing No. 2523-235 sheets 1/2 & 2/2, Handling OSGs Up/Down Ending General Arrangement
RI Drawing No. 2523-240 sheets 1/3, 2/3 & 3/3 Handling OSG Trailer Transport OLS to OSGSF
Specification C-312, Technical Specification for Installation of Bar-Lock Rebar Splices
Specification C-004, Technical Specification for Rigging - Transportation Equipment and Services
Drawing FSK-C-166, Haul Route Load Test Repair and Identification

EP-001, Temporary Structures and Commodities for Reactor Building Modifications
EP-002, Temporary Structures and Commodities for the Removal and Restoration of the Shield Building Concrete Dome and Containment Vessel
EP-004, Screening Review/50.59, Revs. 1 & 2
EP-004, Rigging and Transport
EP-006, Haul Route
EP-007, Temporary Structural Supports and Restraints for Steam Generator Replacement
EP-008, 4100W Tower Crane and Liebherr LR 1400/1 Crane
Calc. No. C-001, Rev. 3, Weight of Steam Generators, Upper Lateral Restraint Ring, Lower Lateral Bumpers
Calc. No. C-021, Haul Route Test Weight Wheel Loads
Calc. No. C-024, Rev. 1, Evaluation of Shield Building Dome for Interim Configuration with Construction Openings during SGRO
Calc. No. C-029, Evaluation of Underground Commodities along Haul Route/Temporary Protection
Calc. No. C-045, Rev. 0, Shield Building Dome Construction Platform
Calc. No. C-046, Rev. 2, Design of Shield Building Construction Platform Anchorage Detail
Calc. No. C-201, Haul Route
TR-C-001, “Alternate Rebar Splice - Bar-Lock Mechanical Splices,”Topical Report (TR), letter to NRC from TVA, 3/18/2002
TR-C-001, “Alternate Rebar Splice - Bar-Lock Mechanical Splices,” Response to NRC Request for Additional Information, 12/10/2002
TR-C-002, “Rigging and Heavy Load Handling Topical Report,” Response to NRC Request for Additional Information, 1/15/2003
TR-C-002, “Rigging and Heavy Load Handling Topical Report,” Issuance of Safety Evaluation of Topical Report, 3/25/2003
TR-C-002, “Rigging and Heavy Load Handling Topical Report,” Issuance of Amendment, 3/26/2003
TR-C-002, “Rigging and Heavy Load Handling Topical Report,” Additional Information, 2/4/2003
TR-C-002, “Rigging and Heavy Load Handling Topical Report,” Commitment Information, 2/19/2003
TR-C-003, “Steam Generator Compartment Roof Modification,” Topical Report, 2/14/2003
Mammoet: Daily Inspection Checklist for OLS Crane
Huisman B.V. - Itrec B.V. Rotterdam 010-245 22 22, Manual Document
Letter from Lloyd’s Register, 7/1/1998, Certificate Number: 9757249, 4800 Twin Ring & Platform Ring Crane Heavy Duty
Bechtel Special Processes Standard, SPS-1
Bechtel General Welding Specification GWS-1, Rev. 9
Bechtel Nondestructive Examination Standard, Ultrasonic Thickness Measurement UT-TM, Rev. 0
Bechtel Drawing No.24370-C-002, sheets 1/2 & 2/2, OLS Layout and S/G Heavy Lift Load Path
Bechtel Drawing No.24370-C-003, Haul Route Load Drop Protection
Bechtel Drawing No.24370-C-035, sheet 4, Shield Building Dome, Steel Trusses & Platform Foundation Pier Plan, Sections & Details
Bechtel Technical Specification for Welding Filler Material WM-ER308L, Rev. 0

Bechtel Welding Procedure Specification P8-T (RA), Rev. 1
Bechtel Nondestructive Examination Standard, Radiographic Examination RT-ASME-III, Rev. 0
Bechtel General Welding Standard, PHT-1, Rev. 0
Report on Steam Generator Eddy Current Examination and Repair, TVA Sequoyah Nuclear Power Plant Unit 1, Pre-Service Inspection, January 2003, MRS-FSR-1216-TVA, Rev. 00
Sequoyah Unit 1, Steam Generator Replacement Project, Project Manual, Rev. 2
DCN-D-20672, SG Vessel Replacement / RCS Support Modifications
DCN-D-20673, Containment Structural Modifications
DCN-D-20674, Steam Generator Replacement Large Bore Piping Modifications
DCN-D-20676, Steam Generator Replacement Small Bore Piping Modifications
DCN-D-20849, Temporary Interference Removal for SGR
PER 03-005784-000, Annular gap around the steam generator compartment plug.
PER 03-005640-000, Feedwater Loop #1, weld deficiency at pipe to pipe end.

Work Package and Inspection Reports (WP&IRs):

C-OLS-007, PTC Crane Foundation
C-HRX-011, Haul Route Modifications (Inside and Outside the Protected Area) Steam Generator Haul Route Load Test
C-CD14-017, Dome Concrete “Opening” 1 / 4
C-CD23-018, Dome Concrete “Opening” 2 / 3
C-ACX-020, Auxiliary Crane Installation and Removal
C-PLUG14-021, Remove and Reinstall S.G. Enclosure Plug #1 & #4
C-S/1-031, SG 1 Structural Interferences
C-SL14-035, Steel Containment Vessel Opening 1 & 4
M-UL1-072, SG 1 Upper Lateral Support Removal and Replacement
M-VS1-076, SG Temp Supports
P-FW1-085, Feedwater Piping for SG#1
P-MS1-097, Main Steam Piping for SG #1
R-MTC-037, Installation, Use and Removal of the Tower Crane
R-RSGX-038, Rig In RSG 1, 2, 3, & 4
R-RSGX-016, RSG Offload and Transport to Preparation Area
R-OSGX-039, Rig Out OSG 1, 2, 3, & 4
R-TRSG-046, Transport RSG’s from Preparation Area to the U/D Area
R-TOSG-047, Transport OSG’s From U/D Area to the OSGSF
R-LIB-147, Installation, Use and Removal of Liebherr Crane
R-SUPX-148, Superstructure Installation and Removal
R-OLS–015, Installation, Use and Removal of the Outside Lift System

Security

Site Security Instruction 1, Revision 6, “Security Instructions for Members of the Security Force,” Appendices C, “Search and Final Access Control Functions,” F, “Contingency Plans and Reporting Requirements,” and J, “Instructions for Major Loss of Security Systems”
Security force weekly overtime rates, September 2002 and December 13, 2002 through April 11, 2003
Desktop Instruction Letter 17, “Instructions for Officers Assigned Duties During Unit 1 Cycle 12 Steam Generator Outage
Desktop Instruction Letter 26, “PA Gate(s) and VBS Operation”
Problem Evaluation Report (PER) 03-001879-000
PER 03-002134-000