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

October 27, 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 05000327/2003005, 05000328/2003005, and 07200034/2003001

Dear Mr. Scalice:

On September 27, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Power Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 2, 2003, with Mr. Rick Purcell and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS).

ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket Nos.: 50-327, 50-328, 72-34
License Nos.: DPR-77, DPR-79

Enclosure: Inspection Report 05000327/2003005, 05000328/2003005, and 07200034/2003001
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328, 72-34
License Nos:	DPR-77, DPR-79
Report No:	05000327/2003005, 05000328/2003005, and 07200034/2003001
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	June 29, 2003 - September 27, 2003
Inspectors:	S. Freeman, Senior Resident Inspector R. Telson, Resident Inspector R. Aiello, Senior Operations Engineer (1R02 and 1R17) D. Mas-Penaranda, Reactor Inspector (1R02 and 1R17) R. Taylor, Reactor Inspector (1R02 and 1R17) P. VanDorn, Senior Reactor Inspector (1R07) J. Lenahan, Senior Reactor Inspector (1R02 and 1R17) R. Chou, Reactor Inspector (4OA5) R. Carrion, Project Engineer (4OA1)
Approved by:	S. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2003-005, IR 05000328/2003-005, IR 07200034/2003001; 06/29/2003 - 09/27/2003; Sequoyah Nuclear Power Plant, Units 1 & 2 and ISFSI; resident inspector integrated report.

The report covered a three-month period of inspection by resident inspectors and an announced inspection by seven region-based operations and reactor inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

None.

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100% rated thermal power during the inspection period, except for a power reduction to 49% rated thermal power (RTP) on July 11, 2003, to work on two heater drain control valves and to perform load rejection testing. The unit returned to 100% RTP on July 12, 2003. The unit was manually tripped on August 28, 2003, when a closed instrument isolation valve caused a generator 100% load reject during turbine testing. The unit returned to 100% RTP on September 2, 2003.

Unit 2 operated at or near 100% rated thermal power during the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility, Updated Final Safety Analysis Report (UFSAR), or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for seven design changes and seven operation procedure changes. Additional information, such as calculations, supporting analyses, the UFSAR, drawings, and procedures were included as part of the review process to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The 14 evaluations reviewed are listed in the attachment.

The inspectors also reviewed samples of changes such as design changes, UFSAR changes, procedure changes, and licensee commitment changes for which the licensee had determined that regulatory evaluations were not required. The review was to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10CFR50.59. In addition, the inspectors reviewed a sampling of the facility's training records to verify that persons signing the evaluation as preparers and reviewers were qualified. The inspectors also reviewed a sampling of the facility's training records to verify that training was conducted on procedures that were changed. The sixteen "screened out" changes, reviewed by the inspectors, are listed in the attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

.1 Partial System Walkdowns.

The inspectors performed a partial walkdown of the following three systems to verify the operability of redundant or diverse systems and components and to identify any discrepancies that impact the function of the system 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. Additional documents reviewed are listed in the attachment.

- A-train of auxiliary feed water during unavailability of ERCW to Motor Driven Auxiliary Feedwater Pump 2B-B

- A-train of safety injection during unavailability of Safety Injection Pump 1B-B

- B-train of containment spray during maintenance of Containment Spray Train 2A-A

.2 Complete System Walkdown

The inspectors performed a complete system walkdown of the Essential Raw Cooling Water (ERCW) System to verify proper equipment alignment and identify any discrepancies that could impact the function of the system and increase risk.

The inspectors reviewed the UFSAR, system procedures, system drawings, and system design documents to determine the correct lineup and then examined system components and their configuration to identify any discrepancies between the existing lineup and the correct lineup. In addition, the inspectors reviewed the effects of ERCW discharge temperature on the NPSH of the motor driven AFW pumps and reviewed a synopsis of water hammer issues from Generic Letter 96-06 and evaluated for their impact on ERCW. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

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

- Essential Raw Cooling Water Building
- Control Building Elevation 734 (Vital Battery Board Rooms)
- Emergency Diesel Generator Building
- Auxiliary Building Elevation 690 (Unit 2 Pipe Gallery)
- Auxiliary Building Elevation 734 (6.9-kV Shutdown Board Rooms A and B)
- Control Bldg Elevation 734 (Mechanical Equipment Room)
- Auxiliary Building Elevation 759 (Unit 2 Pressurizer Heater Transformer and Control Rod Drive Equipment Rooms)
- Auxiliary Building Elevation 734 (Emergency Gas Treatment Filter Room)

The inspectors observed the performance of the site fire brigade during an unannounced drill on September 22, 2003, to evaluate the readiness of the fire brigade to fight fires. The observed drill simulated a fire in Elevation 669 of the Unit 2 Auxiliary Building.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed inspection records, test records, Engineering and Maintenance procedures, and other documentation to ensure that heat exchanger (HX) deficiencies that could mask or degrade performance were identified. Inspection records for risk significant HXs were reviewed which included the Component Cooling and Diesel Generator jacket water HXs. The inspectors also reviewed the general health of the ERCW system via review of inspection/test results, review of chemistry activities, review of ERCW corrective maintenance history, review of the ERCW health reports, review of a self-assessment, review of ultrasonic inspection data, and discussions with the ERCW system engineer and HX engineer. Selected Problem Evaluation Reports (PERs) were reviewed for potential common cause problems and problems which could affect system performance to confirm that the licensee was entering problems into the corrective action program and initiating appropriate corrective actions. In addition, the inspectors conducted a walkdown of most of the ERCW system and the major components.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed annual requalification testing on September 15, 2003. The operators were tested on two scenarios. These involved a simulated Adverse Transient Without Scram with a feedwater line break and a steam generator tube rupture. 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 observed the evaluators' critique and reviewed simulator fidelity to verify that it closely paralleled recent modifications.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following six activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65 (a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee's risk assessment tool and risk categories in accordance with Procedure SPP-7.1, Work Control Process, Revision 4, and Instruction, 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Revision 7. Additional documents reviewed are listed in the attachment.

- WO 03-009434-001, Repair Fire Suppression Piping in Switchyard
- Removal of the ERCW suction path from Motor Driven AFW Pump 2B-B during MOVATs testing
- Removal of ERCW Traveling Screen B-B from Service for Inspection by Divers
- Reinstatement of Unit 1 Main Transformer Sudden Pressure Relays
- Removal of Emergency Diesel Generator (EDG) 2A-A for corrective maintenance
- Unit 1 480-VAC Switchboard Alternate Supply Breaker Test

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

During the Unit 1 manual reactor trip following a 100% generator load rejection on August 28, 2003, the inspectors reviewed the operator logs, plant computer information, and associated 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.

The NRC performed a special inspection of this event. The results of that inspection are documented in IR 05000327/2003010.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the six 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. The inspectors reviewed the UFSAR to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed implemented compensatory measures to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PERs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the attachment.

- PER 03-010415-000, Through-Wall Leak in EDG 2A ERCW Piping
- PER 03-008701-000, Cannot Maintain Annulus Vacuum of 5 Inches
- PER 03-011298-000, Wrong Oil Found in 2A-S TDAFW Pump Bearings
- PER 03-009567-000, Silver and Low Viscosity in EDG 1A2 Engine Oil
- PER 03-011780-000, Rust and Corrosion on Containment Spray Pumps
- PER 03-012671-000, Diesel Generator Inoperable During SSPS Testing

b. Findings

Introduction: A URI was identified involving the use of the alternate source term in an operability evaluation associated with problems establishing vacuum in the Unit 1 Annulus. This item remains unresolved until the NRC determines the acceptability of this practice.

Description: On June 9, 2003, following the Unit 1 Steam Generator Replacement outage and associated containment repairs, the licensee identified that neither the 1A nor 1B Annulus Vacuum Fans could maintain the differential pressure between the Unit

1 Annulus and Auxiliary Building at greater than five inches of water vacuum. As the five inches of vacuum is an initial condition of the design analysis for the EGTS as contained in Chapter 6.2.3.3.2 of the UFSAR, the licensee performed a functional evaluation of the condition. The licensee concluded that, with initial annulus pressure between zero and five inches of water vacuum, the EGTS was able to perform its function based on the assumption that no fission products would be released to the containment atmosphere for 10 minutes following an accident. Therefore, the EGTS would be able to establish a vacuum in the annulus before any fission products were released to the atmosphere, even with annulus vacuum lower than originally assumed in the UFSAR. The licensee justified the 10-minute assumption using Regulatory Guide (RG) 1.183, Alternate Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors. The RG described the use of a 10-minute release time for facilities that were licensed to use the alternate source term.

NRC guidance (i.e., Generic Letter 91-18), allows the use of conservatism in design margin and the use of engineering judgement to justify that degraded equipment is operable; however, that judgement must be within the current licensing basis. The generic letter also states that the definition of operability assumes that degraded equipment can perform its specified function given that the event for which it is designed occurs. The inspectors determined that this would require any operability evaluation on EGTS to assume that an accident occurred and that containment leakage assumed in the UFSAR, including the timing, was present. Chapter 15 of the UFSAR assumed an immediate release.

Therefore, the inspectors challenged the acceptability of the 10-minute assumption for two reasons. First, the 10-minute delay was outside the current license basis. It was part of the alternate source term which, in accordance with 10 CFR 50.67, required a license amendment. The alternate source term had not been requested or approved for Sequoyah. Secondly, the 10-minute delay added margin to the design instead of using the conservatism built into the current analysis. In response to inspector questions, the licensee recalculated exclusion area boundary dose rates assuming the containment leakage listed in the UFSAR and determined that dose rates were below the guidelines of 10 CFR 100.

Assessment: If the use of an assumption based on the unanalyzed site specific alternate source term were determined to be inappropriate and remained uncorrected, it could result in calculated dose rates exceeding the guidelines of 10 CFR 100, a more significant safety concern. This would be more than minor.

Enforcement: 10 CFR 50.67, Paragraph (b) requires licensees who seek to revise the current accident source term to obtain approval from the NRC via a license amendment. Furthermore, the definition of source term contained in 10 CFR 50.2 includes the timing of the release. An inappropriate use of the 10-minute delay assumption based on the alternate source term would be considered a change to the current accident source term and therefore a violation of 10 CFR 50.67, Paragraph (b). However, the licensee has indicated that use of the 10-minute assumption in operability evaluations was not a revision to the licensed accident source term in a design basis radiological consequence analysis and therefore was acceptable. Pending resolution of the acceptability of using the alternate source term as engineering judgement in operability evaluations this item remains unresolved and is identified as URI 50-327/03-05-01, Use of Alternate Source Term in Operability Evaluations.

1R17 Permanent Plant Modifications

.1 Biennial Review

a. Inspection Scope

The inspectors evaluated design change packages (DCP) for 14 modifications, in the Initiating Events and Mitigating Systems cornerstone areas, to evaluate the modifications for adverse affects on system availability, reliability, and functional capability. The modifications included 12 design change notices (DCNs), one temporary alteration/modification (TACF), and one post-issuance change (PIC). The modifications and the associated attributes reviewed are as follows and additional documents reviewed are listed in the attachment:

DCN D20258, Modify Supports to Fit Replacement Safety Valves (1-VLV-68-564, -565). Valves have Increased Flange Diameter which Requires a Change in the Support Attachment (Clamp).

- Licensing Basis
- Materials/Replacement components compatibility
- Structural
- Code requirements, and seismic requirements
- Design Basis Review

DCN D20421A, CCS-Surge Tank Vent Valve and Level Instrumentation

- Control signals appropriate under accident conditions
- Replacement component properties serve functional requirements
- Operation procedure and training
- Replacement component properties serve functional requirements under accident/event conditions.

DCN D20664A, Replace Cold Leg Accumulator Level Transmitters, Rev.0

- Control signals appropriate under accident conditions
- Replacement component properties serve functional requirements under accident/event conditions.
- Added flowpaths have not introduced new failure
- Verify post maintenance change

DCN D20733A, Check Valve Conversion to Globe Valve

- Physical inspection
- Operating procedures
- Material evaluation
- Process and Instrumentation Drawings

DCN D20744, Abandon Part of Isokinetic Sample Panel for Aux Building Vent Monitor 0-RE-90-101, Replace Obsolete Equipment, Change to a Fixed Flow Rate System
- Functional test results
- Design basis review
- Supporting vendor analyses
- Plant procedure and critical drawing updating

DCN D20843A, Change Unit 1 Generator Protective Relay Types 160, 121GB, and 151G, Rev. 0
- Control signals appropriate under accident conditions
- Affected operation procedures identified and necessary changes made

DCN D20938A, Adjust the Reactor Control System Coolant Average Temperature Lead-lag Constants
- Instrumentation and Controls
- Engineering analysis
- Post modification testing

DCN D 21180A, Revise Air Start Solenoid Control Circuit, Rev.0
- Response time sufficient to serve functional requirements
- Control signals appropriate under accident conditions
- Failure modes bounded by the existing analysis
- Affected operation procedures identified and necessary changes made.
- Physical inspection
- Verify post-maintenance procedure changes

DCN D21362A, Modify Reactor Vent Line Supports as Detailed
- Licensing Basis
- Materials/Replacement components compatibility
- Structural
- Code requirements, and seismic requirements
- Design Basis Review

DCN D21388, Inside Containment, Protect ACA Lines from Damage Due to Pipe Breaks
- Licensing Basis
- Materials/Replacement components compatibility
- Structural
- Code requirements, and seismic requirements
- Design Basis Review

DCN D21470A, Replace U1 Steam Traps 003-0975 and 0976
- Physical inspection
- Safety Assessment
- Procurement information
- Process and Instrumentation Drawings

DCN M12804A, Resolve Civil Issues as Described in SQ 961347 PER for Unit 2 Pipe Support Modifications Inside Containment

- Licensing Basis
- Materials/Replacement components compatibility
- Structural
- Code requirements, and seismic requirements
- Design basis review

TACF 1-01-003-062, Reach Rod Remote Operator for 1-VLV-062-0538

- Physical inspection
- Operating procedures

PIC 21364, Repair Leaking CRDM Seal Welds on the Reactor Vessel Head at Location J11

- Licensing Basis
- Materials/Replacement components compatibility, Code requirements, and seismic requirements
- Functional test results
- Design basis review
- Supporting vendor analyses
- Plant procedure and critical drawing updating

b. Findings

No findings of significance were identified.

.2 Current Review of Ongoing Modifications

a. Inspection Scope

The inspectors reviewed DCN 21358 Stage 3, Add Redundant Manual Start Contacts to Containment Spray Pump 2A-A Hand-switch in Main Control Room, Revision A, and interviewed engineering personnel regarding the modification and associated post-modification testing to verify that (1) the design bases, licensing bases, and performance capability had not been degraded through this modification, and (2) the modification was not performed during increased risk-significant configurations that placed the plant in an unsafe condition. The inspectors also reviewed applicable sections of the UFSAR, plant modification procedures, system drawings, supporting analyses, technical specifications, and related PERs.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the six 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.

- 1-RT-MOD-520-003.0, 20% Load Step Change, Revision 1
- 0-SI-SXV-003-266.0, ASME Section XI Valve Testing, Revision 19 (Valve 2-FCV-67-126A, ERCW to MDAFW Pump 2B)
- 0-SI-IFT-099-093.1, Functional Tests of Turbine Auto Stop Oil Dump and Throttle Valves Reactor Trips, Revision 9
- WO 02-009679-000, Replace the 2A-A Containment Spray Pump W-2 Main Control Room Hand-switch and implement Stage 3 of DCN D21358A
- WO 03-004453-000, Inspect and Clean EDG 2B-1 Air Start Compressor Discharge Check Valves
- WO 03-004454-000, Inspect and Clean EDG Air Start System Air Tank Check Valves

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

Following the load rejection and manual trip of Unit 1 on August 28, 2003, the unit remained in Mode 3 and the licensee entered a forced outage to replace a RCS flow transmitter in lower containment and perform other activities on plant secondary equipment. The inspectors observed containment entry controls and reviewed Procedure 0-SI-OPS-000-011.0, Containment Access Control During Modes 1-4, Revision 14, to ensure that all items which entered containment were removed so that nothing would be left which could affect performance of the containment sump.

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 surveillance requirements, the UFSAR, and applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions. Documents reviewed are listed in the attachment. Those tests included the following:

- 1-SI-SXP-074-201.B, Residual Heat Removal Pump 1B-B Performance Test, Revision 7*
- SI-90.82, Reactor Trip Instrumentation Monthly Functional Test (SSPS), Revision 34 (Unit 2 Train A)
- 0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 0, (Unit 1)**
- 0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 0, (Unit 2)**
- 0-SI-EBT-250-100.5, Performance Testing of 125-VDC Vital Batteries and 125-VDC Vital Battery Charger Test, Charger UNID: 0-CHGB-250-QG-E (125-VDC Vital Battery II)
- 2-SI-SXV-000-201.0, Full Stroking of Category A and B Valves During Operation, Revision 6*

*This procedure included inservice testing requirements.
**This procedure included testing of a large containment isolation valve.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors evaluated the conduct of a licensee emergency drills on July 31, 2003, and September 9, 2003, to identify any weaknesses and deficiencies in classification, notification, and Protective Action Recommendations (PARs) development activities. The inspectors also attended the licensee critique of these drills to compare any inspector observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying failures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the PIs listed below for the period from January 1, 2002, through March 31, 2003; and July 1, 2002, through March 31, 2003, respectively. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Cornerstone: Mitigating Systems

- Safety System Unavailability: Unit 1 Heat Removal System AFW
- Safety System Unavailability: Unit 2 Heat Removal System AFW
- Safety System Unavailability: Unit 1 Residual Heat Removal System
- Safety System Unavailability: Unit 2 Residual Heat Removal System

The inspectors reviewed portions of the operations logs and raw PI data developed from monthly operating reports and discussed the methods for compiling and reporting the PIs with cognizant engineering personnel. The inspectors also independently calculated selected reported values to verify their accuracy. The inspectors compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

Unit 1 Generator Load Rejection and Manual Reactor Trip With Loss of Normal Heat Sink

On August 28, 2003 the operators manually tripped Unit 1 when, during turbine thrust bearing wear testing, the generator output breakers opened resulting in a 100% generator load rejection. The operators declared an Alert based on their assessment that a turbine trip occurred without an automatic reactor 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. Indication of flow on all four steam lines and of a failed open steam dump valve following the trip resulted in the operators closing the main steam isolation valves, which resulted in a loss of the normal heat sink. The event was reported to the NRC as Event Notification (EN) 40113 and documented in the licensee corrective action program as PER 03-011940-000. Evaluation of personnel performance is addressed in Section 1R14.

The NRC performed a special inspection of this event. The results of that inspection are documented in IR 05000327/2003010.

4OA5 Review of Quality Assurance and Plant Modification for Independent Spent Fuel Storage Installation (ISFSI)

a. Inspection Scope (60851, 60853, and 60854)

The inspectors reviewed the quality assurance program, plant modification, and construction activities associated with the ISFSI construction project. The inspectors reviewed the TVA submittal to the NRC dated July 6, 1999, with respect to its intention to apply the previously approved 10 CFR 50, Appendix B, Quality Assurance Program to activities at the Sequoyah ISFSI, and self-assessment report SQN-PROJ-03-002, in order to determine the adequacy and effectiveness of recent and on-going ISFSI activities at Sequoyah and Corporate Nuclear Fuel. The inspectors reviewed Design Change Notices (DCNs), design calculations, records, an Auxiliary Building crane modification and load test, and the crane operator qualification and medical records in order to determine the adequacy and compliance with the procedures. The inspectors reviewed the safety reviews under 10 CFR 50.59 and calculations to ensure that the Auxiliary Building floors were sufficient to support cask loads and transfer associated weight during the spent fuel cask operations. The inspectors reviewed the Auxiliary Building crane to ensure that it had been modified to a single-failure-proof crane and reviewed load tests in order to verify the crane load capacity for the spent fuel casks. The inspectors walked down the cask transport route, a concrete overpack construction pad, and the cask concrete storage pads, including fence and lighting, to verify completion. The inspectors measured the concrete pad size, the distances between the pads and the fence, and weld sizes and member sizes for the half of the work platform to be installed in the spent fuel pool cask pit area. The results of the measurements were compared to the design drawings. The inspectors also walked down the spent fuel pool area to review the Auxiliary Building crane modification and cask pit stands and portions of the work platform already installed at the pool pit area for the preparation of cask loading operation and compared results to the design specification, procedures, drawings, and Holtec HI-STORM 100 FSAR.

The inspectors reviewed corrective action including the violation response, revised procedure, and records of air content and concrete cylinder compressive load testings for Violation (VIO) 72-34/2002-001-01, Inadequate Procedure to Use the Correct Air Content Acceptance Criteria for Concrete to Ensure Adequacy. The inspectors also reviewed Problem Evaluation Reports (PERs) 02-015393-000 and 02-013982-000 associated with this issue. This item is considered closed based on the records reviewed.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On October 2, 2003, the resident inspectors presented the inspection results to Mr. Rick Purcell and members of his staff, who acknowledged the findings.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

J. Bajraszewski, Licensing Engineer
D. Clift, Maintenance and Modifications Manager
H. Cothran, Steam Generator Manager
J. Gates, Business & Work Performance Manager
M. Gillman, Operations Manager
C. Kent, Radcon/Chemistry Manager
D. Koehl, Engineering and Site Support Manager
D. Kulisek, 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
D. Thompson, Security Manager

NRC personnel:

S. Cahill, Chief, Reactor Projects Branch 6
R. Bernard, Region II, Senior Reactor Analyst
M. Marshall, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000327/2003005-001	URI	Use of Alternate Source Term in Operability Evaluations

Closed

07200034/2002001-001	VIO	Inadequate Procedure to Use the Correct Air Content Acceptance Criteria for Concrete to Ensure Adequacy

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Sections R02: Evaluation of Changes, Tests, or Experiments (see Section R17)

Section R04: Equipment Alignment

- 0-SI-OPS-067-682.M, ERCW Flow Balance Valve Position Verification, Revision 21
- 1-SI-OPS-000-186.M, Locked Valve Position Verification, Revision 10
- 2-SI-OPS-000-186.M, Locked Valve Position Verification, Revision 10
- 0-SO-67-1, Essential Raw Cooling Water, Revision 42
- 1,2-47W845-1, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 38
- 1,2-47W845-2, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 77
- 1-47W845-3, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 39
- 2-47W845-3, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 22
- 2-47W845-4, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 11
- 1,2-47W845-5, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 54
- 1-47W845-6, Mechanical Flow Diagram - Essential Raw Cooling Water System, Revision 20

Section R05: Fire Protection

- FOR2003A0381, FP impairment permit for isolation of HPFP supply to EDG building
- FOR2003A0365, FP impairment permit for isolation of HPFP supply to U2 reactor building and portions of auxiliary building
- FOR2003A0030, FP impairment permit for inoperable fire detection zone 353, U2 containment lower compartment coolers, El. 693
- PER 03-000188-000, Fire detection zone 353, U2 containment lower compartment coolers removed from service and will not be restored within 14 days as specified by the FOR due because restoration requires U2 outage
- PER 03-011649-000, NRC inspector-identified concerns with compensatory actions in FOR2003A0365 for HPFP supply outage to U2 reactor building and portions of auxiliary building
- Tagout 0-TO-2003-0006, Clearance 0-26-0500-W/W, Isolate portions of HPFP to facilitate WO 02-008849 and repair HPFP header leak over VCT
- WO 02-008849-000, Repair HPFP header leak
- Tagout 0-TO-2003-0006, Clearance 0-26-0500F-W/W, Sixth revision, expanded isolation of HPFP to facilitate WO 02-008849 and repair HPFP header leak over VCT
- 0-PI-FPU-317-537.Q, Fire and Medical Emergency Equipment Inventory, Revision 11
- PFP No. RXB-0-679-02, Fire Protection Pre-Fire Plan for Reactor Building El. 679, Revision 3

- PFP No. RXB 0-701-02, Fire Protection Pre-Fire Plan for Reactor Building Annulus Area El. 701 and 721, Revision 3
- PFP No. RXB-0-734-02, Fire Protection Pre-Fire Plan for Reactor Building El. 734 and Annulus El. 740, 759, and 778, Revision 3
- FP No. AUX-0-759-02, Fire Protection Pre-Fire Plan for Auxiliary Building El. 759 (Unit 2 side) and Unit 2 AEB, Revision 2
- PFP No. AUX 0-734-03, Fire Protection Pre-Fire Plan for Auxiliary Building (U-X), El. 734, Revision 4
- SQN-IPEEE-005, Individual Plant Examination for External Events - Fires (IPEEE - Fires)
- PFP No. AUX-0-734-01, Revision 6, Aux. Bldg. Unit 1 side, El. 734, Electrical Board Rms.
- PFP No. AUX-0-734-02, Revision 5, Aux. Bldg. Unit 2 side, El. 734, Electrical Board Rms.
- PFP No. CON-0-732-00, Revision 5, Control Building, El. 732.
- PER 02-011727-000, HPFP leak off the eight inch header west of the AERCW cooling towers

Section R07: Heat Sink Performance

Procedures

- 0-TI-SXX-000-146.0, Program for Implementing NRC Generic Letter 89-13, Rev. 0
- 0-TI-SXX-000-109.0, Nondestructive Testing of Stainless Steel Welds to Assess Damage Resulting from Microbiologically-Influenced Corrosion (MIC), Rev. 3
- 0-TI-XXX-000-704.0, Evaluation of Raw Water System Ultrasonic Pipe Wall Thickness Measurement Data, Rev. 4
- 0-PI-DXX-000-704.1, Degradation Monitoring Program for Raw Water Systems, Rev. 3

Inspection/Test Records

- Inspect D/G 2B-2 heat exchanger for clams, mic, and other degradation, clean as required;
  - WO No. 02-008686, dated 02/10/2003
  - WO No.00-004320, dated 02/12/2001
- Inspect D/G 2B-1 heat exchanger for clams, mic, and other degradation, clean as required;
  - WO No. 02-008685, dated 02/10/2003
  - WO No. 00-004321, dated 02/12/2001
- Inspect D/G 2A-2 water cooler for clams, mic, and other degradation;
  - WO No. 00-004932, dated 02/05/2001
  - WO No. 02-009318, dated 02/03/2003
- Inspect D/G 2A-1 water cooler for clams, mic, and other degradation;
  - WO No. 00-04931, dated 02/05/2001
  - WO No. 02-009381, dated 02/03/2003
- Inspect D/G 1B-2 water cooler for clams, mic, and other degradation, clean as required;
  - WO No. 00-003929, dated 01/29/2001
  - WO No. 02-008347, dated 01/27/2003

- Inspect D/G 1B-1 water cooler for clams, mic, and other degradation clean as required;
  - WO No. 00-003928, dated 01/29/2001
  - WO No. 02-008348, dated 01/22/2003
- Inspect D/G 1A-2 heat exchanger for clams, mic and other degradation, clean as required;
  - WO No. 02-009839, dated 01/21/2003
  - WO No. 00-005196, dated 01/23/2001
- Inspect D/G 1A-1 heat exchanger for clams, mic and other degradation, and clean as required;
  - WO No. 02-009840, dated 01/21/2003
  - WO No. 00-005195, dated 01/23/2001
- Component cooling heat exchanger 0B2 clam inspection;
  - WO No. 00-000849, dated 04/12/2000
  - WO No. 02-009156, dated 05/20/2003
- Component cooling heat exchanger 0B1 clam inspection;
  - WO No. 01-002984, dated 06/19/2001
  - WO No. 02-009155, dated 05/10/2003
- Component cooling heat exchanger 2A2 clam and mic inspection;
  - WO No. 00-011268, dated 04/18/2001
  - WO No. 02-011449, dated 05/18/2003
- Component cooling heat exchanger 2A1 clam and mic inspection;
  - WO No. 00-006894, dated 04/17/2001
  - WO No. 02-011450, dated 05/04/2003
- Component cooling heat exchanger 1A2 clam and mic inspection;
  - WO No. 02-002610, dated 03/22/2002
  - WO No. 02-010453, dated 04/28/2003
- Component cooling heat exchanger 1A1 clam and mic inspection;
  - WO No. 02-002608, dated 03/19/2002
  - WO No. 02-010452, dated 05/05/2003
- 2001 Survey of ERCW Intake, dated 08/09/2001
- 2nd Quarter 2002 Report for the Raw Water Chemical Treatment Program, dated 08/08/2002
- -PI-CEM-000-460.4, Raw Water Quaternary Amine Treatment Monitoring, dated 06/25/2003
- 0-PI-CEM-067-712.0, Clam Control, dated 07/12/2003
- Eddy Current Examination Report on Diesel Generators 1AA and 2AA Coolers 1 and 2, dated 01/1999
- Eddy Current Examination Report on Diesel Generator Cooler Heat Exchanger 2A-2, dated 05/1989
- 0-PI-SFT-070-002.0; Performance Testing of Component Cooling Heat Exchangers 0B1, 0B2; dated 03/18/2003, 03/07/2003, 04/12/2002, 10/22/2001
- 1-PI-SFT-070-001.0; Performance Testing of Component Cooling Heat Exchangers 1A1, 1A2; dated 03/18/2001, 03/17/2003, 03/06/2003, 02/08/2002, 12/09/2002, 09/16/2002, 06/18/2002, 10/22/2001
- 2-PI-SFT-070-001.0; Performance Testing of Component Cooling Heat Exchangers 2A1, 2A2; dated 03/07/2003, 12/23/2002, 10/01/2002, 06/25/2002, 04/12/2002, 02/08/2002
- 0-PI-SFT-067-005.A; ERCW A Train System Flow Balance Using Hydraulic Modeling; dated 02/28/2002

- 0-PI-SFT-067-005.B; ERCW B Train System Flow Balance Using Hydraulic Modeling; dated 02/25/2002

Problem Evaluation Reports

- 01-005036, 2-FCV-70-156 thermalled out while being throttled
- 01-007878, Fouling factors have typically exceeded 0.0003
- 02-001183, Leak was discovered in the ERCW piping downstream of valve 1-67-537A
- 02-001369, The fouling factor in the 1A1/1A2 CCS HX increased
- 02-004822, Asiatic clams have been discovered on several places
- 02-004843, ERCW supply lines to the TDAFWPs contain an unventable volume
- 02-006970, Received motor tripout on the P-B ERCW pump
- 02-009996, External corrosion found on 24" carbon steel piping downstream of ERCW strainer B2B-B
- 03-000269, Design basis document does not clearly exist that validates minimum river level of 670 EL
- 03-005724, Functional evaluation for the ERCW Type CPSJ cable failure does not adequately address the extent of condition
- 03-005732, Chemistry is unable to continuously chlorinate the ERCW
- 03-006466, Four Asiatic clam shells were found in valve 1-67-773
- 03-006553, 24" ERCW header contained a number of Asiatic clam shells
- 03-006860, Clam inspection on the OB1 CCW Heat exchanger found approximately 6 to 10 asiatic clam shell halves
- 03-007064, Drain valve 1-67-696C broke off from the header
- 03-007597, Leak was identified at the pipe to coupling weld at valve 0-67-625B
- 03-007360, Complete or partial blockage was observed for Unit 1 thermal relief valves

Miscellaneous

- System Status SQN U1 & U2 Essential Raw Cooling Water, 1st quarter FY03
- System Status SQN U1 & U2 Essential Raw Cooling Water, 4th quarter FY02
- SQN, BFN, and WBN Plants Response to GL 89-13, Service Water System Problems Affecting Safety-Related Equipment, dated 01/26/1990
- SQN Revised Program Regarding NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” dated 10/24/1990
- SQN Revised Program Regarding NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” dated 07/29/1992
- SQN Revised Program Regarding NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” dated 09/22/1995
- Component Cooling HX Fouling History; 01/1996 - 06/2003
- ERCW Pipe Leak List; 03/1990 - 06/2003
- SQN-ENG-01-02; Self-Assessment Report; Heat Sink Inspection; 07/09/2001 - 07/27/2001

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

- O&SSDM 4.8, Critical Evolution Meeting, Revision 0
- 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Revision 7
- SPP-7.1, On Line Work Management, Revision 4
- Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants
- NUMARC 93-01, Nuclear Energy Institute Industry Guideline for Monitoring The Effectiveness of Maintenance at Nuclear Power Plants, Revision 2

Section R15 Operability Evaluations

- 0-TI-PDM-000-057.6, Lubrication, Revision 18
- ASME Code Case N-513, dated August 14, 1997
- Generic Letter 90-05, dated June 15, 1990
- Calculation N2-67-A31A, Summary of Piping Analysis, Revision 7

Section R17: Permanent Plant Modifications and Section R02: Evaluation of Changes, Tests, or Experiments

- DCN 21358 Stage 3, Add Redundant Manual Start Contacts on 6.9-kV Shutdown Board Containment Spray Pump 2A-A W2 Handswitch located in Main Control Room, Revision A

Implementing Program Procedures

- SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 8
- SPP-9.4, 10 CFR50.59 Evaluations of Changes, Tests, and Experiments, Rev. 5

50.59 Evaluations

- DCN D20421A, CCS- Surge Tank Vent Valve and Level Instrumentation
- DCN D20922A, System No. 691 (Yard structures - Miscellaneous), ISFSI Structure
- TACF 1-01-003-062 - Reach Rod Remote Operator for 1-VLV-062-0538
- TACF 1-03-029-057, Disable the trip function of relay 159GN3
- TACF 2-03-015-047, Lift wire TT14 to electrical disable relay.
- TACF 2-02-018-063, Removing ground faulted 9-kW heater from service.
- TACF 0-03-007-012, Section of System 012 (Aux Boiler & assoc F.O.) Abandoned piping running from the aux building to the turbine building.

Screened Out Items

- DCN D20664A, Replace cold leg accumulator level Transmitters, Rev.0
- DCN D20843A, Change Unit 1 Generator protective relay types 160, 121GB, and 151G, Rev. 0
- DCN D21180A, Revise Air Start Solenoid Control Circuit, Rev
- PIC 21364, Repair Leaking CRDM Seal Welds on the Reactor Vessel Head at Location J11

Screening Reviews and 50.59 Evaluations Associated with Procedure Revisions

- AOP-C.01, Rod Control System Malfunctions, Rev. 6, Screening Review
- AOP-R.01, Steam Generator Tube Leak, Rev. 10, Screening Review
- AOP-R.03, RHR System Malfunction, Rev. 8, Screening Review
- AOP-S.04, Condensate or Heater Drains Malfunction, Rev. 7, Screening Review
- AOP-S.06, Turbine Trip, Rev. 7, Screening Review
- AOP-T.01, Security Events, Rev. 1, Screening Review
- E-1, Loss of Reactor or Secondary Coolant, Rev. 19, Screening Review
- EA-0-3, Minimizing Secondary Plant Contamination, Rev. 2, Screening Review
- EA-32-3, Isolating Non-Essential Air to Containment, Rev. 0, Screening Review and 50.59 Evaluation
- EA-62-5, Establishing Normal Charging and Letdown, Rev. 5, Screening Review
- ES-0.2, Natural Circulation Cooldown, Rev. 11, Screening Review
- ES-1.3, Transfer to RHR Containment Sump, Rev. 9, Screening Review
- ES-1.4, Transfer to Hot Leg Recirculation, Rev. 3, Screening Review
- FR-Z.1, High Containment Pressure, Rev. 12, Screening Review and 50.59 Evaluation
- STI-160, Zero Pressure Test of Auxiliary Building Ventilation System, Rev. 0, Screening Review and 50.59 Evaluation
- 1-AR-M4-B, NIS/ROD Control, Rev. 17, Screening Review and 50.59 Evaluation
- 0-SO-30-10, Auxiliary Building Ventilation Systems, Rev. 22, Screening Review and 50.59 Evaluation
- 0-SO-30-3, Containment Purge System Operation, Rev. 25, Screening Review and 50.59 Evaluation

Procedure Revisions Associated with Design Changes

- 1-MI-EFT-57-100.0, Generator 1 and Main transformer Relay Functional Test, Rev.9, (DCN D20843A)
- 1-PI- EFT-082-001.R, Set point Verification and calibration for time delay relays associated with Diesel Generator 1A-A Logic, Rev. 7, (DCN D 21180A)
- 1-SO-70-1, System Operating Instruction, Component Cooling Water System “A” Train, Rev.32, (DCN D20421A)
- 0-AR-M27-B-B,0-XA-55-27B-B, Annunciator Response, Component Cooling, Rev. 12, (DCN D20421A)
- 0-P-82-160, Set point and scaling document, Rev. 4, (D21180A)
- 1-L-63-82, Set point and scaling document, Rev. 4, (D20664A)
- 1-L-63-81, Set point and scaling document, Rev. 3, (D20664A),
- 1-L-63-60, Set point and scaling document, Rev. 4, (D20664A)

Training Lesson Plans

- EGT024-007, Rev. 9, Qualified 50.59 Preparer Training
- EGT024-008, Rev. 6, Retraining for Qualified 50.59 Preparers

Section R19: Post Maintenance Testing

- DCN 21358 Stage 3, Add Redundant Manual Start Contacts on 6.9-kV Shutdown Board Containment Spray Pump 2A-A W2 Handswitch located in Main Control Room, Revision A
- MI-10.4, 6900-V Breaker Inspection, Revision 47
- SPP-6.7, Instrumentation Setpoint Scaling and Calibration Program, Revision 1
- Set Point and Scaling Document (SSD) 1-P-47-73, Revision 3
- SSD 1-P-47-74, Revision 3
- SSD 1-P-47-75, Revision 3

Section R22 Surveillance Testing

- SPP-8.1, Conduct of Testing, Rev. 2
- 0-SI-SLT-000-160.0, Primary Containment Total Leak Rate, Revision 2
- 0-PI-EBM-000-001.2, 125-VDC Vital Battery Quarterly Operability, Revision 8, performed on vital battery II charger on August 28, 2003

Section 4OA5 Other Activities

Procedures and Specifications

- TVA General Engineering Specification G-2, Plain and Reinforced Concrete
  TVA Sequoyah Nuclear Plant Modifications & Additions Instruction M&AI-21, Revision 8, Concrete Placement and Repair, Quality Related
- ASME B30.2-2001, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)

Miscellaneous Documents

- Calculation No. SCG1S616, Rev. 0, 3D Seismic Acceleration Time Histories & Stability of HI-STORM 100S at the Railroad Bay Slab Elevation (Holtec Report Nos. HI-2012720 and HI-2012722)
- Calculation No. SCG-1-98, Revs. 19 and 20, Auxiliary Building 706.0 and 714.0 Floor Slabs (Including Holtec Report No. HI-2012687)
- Calculation No. SCG2S02003, Rev. 0, Calculation for Conduit Support Variance 47A056-DV1066-0028
- Calculation No. SCG1S621, Rev. 0, Design Analyses for Work Platform and Cask Loading Stands (Holtec Report Nos. HI-2012734)
- Calculation No. CEB-44N300C7, Rev. 4, 125-Ton Crane - Auxiliary Building
- Calculation No.SCG1S576, Rev. 1, Removable Rubber Door Seal at Access Door at EL. 706'-0"
- Calculation No. SCG-1-74, Rev. 6, Auxiliary Building X-Y Line Walls
- DCN 20958, Rev. A, Design and Install Cask Work Platform & Cask Support Stand in Cask Setdown Area of Spent Fuel Pit
- DCN 20959, Rev. A, Modify Railroad Bay Floor Slab to Allow Movement of a Loaded Spent Fuel Cask
- TVA-NQA-PLN89-A, Rev. 13, Nuclear Quality Assurance Plan
- QA Assessment No. SQN-PROJ-03-002, Self-Assessment Report

- Work Order 01-008186-000, Upgrade Auxiliary Building 125/10-Ton Crane and Load Test
- Holtec Drawing No. 3492 , Sheet 11, Rev. 11, Haul Road
- Holtec Drawing No. 3582, Sheet 6, Rev. 2, Railroad Bay Modification Cask Egress Pad
- Holtec Drawing No. 3689, Sheets 1 - 4 , HI-TRAC 125D Cask Support Stand Assembly
- Holtec Drawing No. 3688, Sheets 1 - 2 , Work Platform Assembly
- Ederer Drawing No. D-41827, Rev. B, Main Hoist Assembly for 125/10-Ton Capacity ,
- Problem Evaluation Report (PER) Nos. 03-012911-000, 03-012912-000, 03-012913-000, 03-013009-000,03-013010-000, 03-013011-000, 03-013012-000, 03-013013-000, and 03-013027-000
- PER 02-015393-000, Document the NRC Notice of Violation found In Inspection Report 72-34/2002-001
- PER 02-013982-000, Failed Air Content in Concrete Pour #1 of the ISFSI Pad
- Compressive Strength Test Results for Pour # 2 & 3
- Qualification, Certification, and Medical Records for a Crane Operator