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

April 23, 2004

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 05000327/2004002 AND 05000328/2004002

Dear Mr. Scalice:

On March 27, 2004, 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 April 1, 2004, with Mr. Rick Purcell and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). Both of these issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Sequoyah facility.

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000327/2004002 AND 05000328/2004002
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:	05000327/2004002 and 05000328/2004002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	December 27, 2003 - March 27, 2004
Inspectors:	S. Freeman, Senior Resident Inspector R. Telson, Resident Inspector S. Shaeffer, Senior Project Engineer (Sections 4OA5.2, 4OA5.3) R. Carrion, Project Engineer (Sections 1R06, 4OA1)
Approved by:	S. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2004002, IR 05000328/2004002; 12/27/2003 - 03/27/2004; Sequoyah Nuclear Power Plant, Units 1 & 2; Personnel Performance During Non-Routine Events, Other Activities.

The report covered a three-month period of inspection by resident inspectors and an announced inspection by one region-based inspector. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification (TS) 6.8.1 for a self-revealing failure to comply with plant general operating procedures. While shutting down Unit 2 in November 2003, operators failed to close the motor-operated reheater steam supply valves to all six moisture-separator reheaters. This resulted in an open steam flow path through the reheaters to the main condenser and caused a Reactor Coolant System temperature decrease to the point where operators had to close the Main Steam Isolation Valves to maintain control.

  This finding is more than minor because it affected the availability of the power conversion system to handle an anticipated transient. This finding is of very low safety significance because there was no actual loss of a safety function, no loss of a TS-required system or loss of a risk-significant maintenance rule system for greater than 24 hours, and no increase in risk from external events. The cause of the finding is related to the cross-cutting element of human performance (Section 1R14).

- Green. A non-cited Severity Level IV violation of 10 CFR 50.48(a) and the Unit 1 and 2 Operating License Conditions was identified for the licensee making an inappropriate change to the approved fire protection program. This change removed the requirement to implement fire watches for impaired fire protection systems and features.

  This finding is more than minor because the lack of a posted fire watch could adversely affect the ability to achieve and maintain safe shutdown in the event of a severe fire in the affected area. This was based on recognition that the ability of the fire watch was not limited to fire identification, but also included mitigating actions taken in the event of fires, such as the ability to close doors limiting fire exposure to adjacent areas and providing more timely fire detection capability in certain cases. This finding is of very low safety significance because, based on an assessment of the impacts of the identified fire protection features removed from service, the licensee's overall safe shutdown capabilities and related fire protection features remained adequate to achieve and maintain safe shutdown conditions. Therefore, this finding is characterized as Green. (Section 4OA5).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100% rated thermal power (RTP) during the inspection period except for an automatic reactor trip on March 15, 2004, due to a spurious main transformer relay actuation. Following repairs and outage activities related to a hydrogen leak on the main generator, the unit was taken critical on March 19, 2004. The unit returned to 100% RTP on March 21, 2004.

Unit 2 began the period shutdown to repair a hydrogen leak on the main generator. Repairs were completed and the unit was taken critical on January 8, 2004, and returned to 100% RTP on January 9, 2004. On January 16, 2004, Unit 2 was shutdown for a different hydrogen leak on the main generator. Repairs were completed and the unit was taken critical on January 25, 2004. The unit returned to 100% RTP on January 27, 2004.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns

The inspectors performed a partial walkdown of the following four 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.

- 1B Residual Heat Removal (RHR) Train during unavailability of 1A RHR Train
- 1B Emergency Core Coolant System (ECCS), Containment Spray (CS), Emergency Gas Treatment System (EGTS), Auxiliary Building Gas Treatment System (ABGTS), and Penetration Room Coolers and 2B ABGTS during A-Train Essential Raw Cooling Water (ERCW) Header Maintenance
- 1B CS Train during unavailability of the 1A CS Heat Exchanger
- Unit 2 Turbine-Driven and 2A Motor-Driven Auxiliary Feedwater (AFW) Trains during unavailability of 2B Motor Driven Train

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted a tour of the eleven 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.

- Auxiliary Building Elevation 734 (6.9-kV Shutdown Board Rooms)
- Essential Raw Cooling Water Building
- Auxiliary Building Elevation 734 (480-V Shutdown Board Rooms, Battery Board Rooms)
- Auxiliary Building Elevation 749 (480-V Reactor MOV Board Rooms)
- Control Building Elevation 669 (250-VDC Battery and Battery Board Rooms)
- Emergency Diesel Generator Building
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Control Building Elevation 706 (Spreading Room)
- Control Building Elevation 732 (Relay Room)
- Auxiliary Building Elevation 714 (Corridor)
- Auxiliary Building Elevation 690 (Corridor)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including: Updated Final Safety Analysis Report (UFSAR) Sections 2.3 and 2.4, including Appendix 2.4A, Flood Protection Plan; Design Criteria Document SQN-DC-V-12.1, Sequoyah Nuclear Plant - Flood Protection Provisions; the Unit 1 Probabilistic Safety Assessment, Sections 3.3.8, Internal Flooding Analysis, and 3.4, Results and Screening Process, for licensee commitments. In addition, the inspectors reviewed licensee drawings: 47W200-7, Revision 10, Equipment Plan - El 727.5, El 706.0, El 653.0, and El 653.0; 1,2-47W476-2, Revision 0, Containment Drains and Embedded Piping; 1,2-47W479-1, Revision 4, Drains and Embedded Piping; and 1,2-47W852-1, Revision 17, Flow Diagram Floor and Equipment Drains, to identify areas and equipment that may be affected by internal flooding. The inspectors performed a walkdown of risk-significant areas, susceptible systems, and equipment in the lower elevations of the Auxiliary Building to verify that the floor drain system, including room sump pumps, level switches, etc. were operable. Plant Procedure 0-PI-IFT-040-001.0, Functional Test of Auxiliary and Reactor Buildings Flood Alarms, Revision 3, and the latest associated Surveillance Task Sheet, which documented the surveillance results of various flood-related components, were also reviewed.

The inspectors also reviewed the licensee's corrective action documents with respect to flood-related items identified in Problem Evaluation Reports (PERs) written in 2003 to verify the adequacy of the corrective actions. The most significant PERs written with respect to internal flooding during the period included the following:

- PER 03-010314-000, The Main Control Room received an alarm indicating that the Control Building floor and equipment room sump level was high. The auxiliary unit operator (AUO) dispatched to the scene had to manually pump down the sump.
- PER 03-016269-000, The Main Control Room received an alarm indicating that there was flooding at elevation 690 of the Unit 2 Auxiliary Building. The AUO dispatched to the scene found no sign of flooding and initiated a work order to check the level switch which triggered the alarm.
- PER 03-017024-000, A large section of the Unit 2 Penetration Room at elevation 669 was flooded during the draining of the RHR system. This event occurred as a result of overflowing the stand pipe drain in the southeast corner of the room.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training on February 2, 2004. The training involved a series of equipment failures, including a main feed pump oil line rupture, leading operators to emergency stop the pump; an automatic turbine run-back; and rapid manual boration. This was followed by a pressurizer safety valve failure leading to a reactor trip, safety injection, containment spray actuation, and alert declaration. 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 TS actions; and group dynamics involved in crew performance. The inspectors also reviewed simulator fidelity to verify that differences between Unit 2 and the simulator were appropriately addressed. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation

a. Inspection Scope

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

- Search for grounds on 125-Volt Direct Current (VDC) Vital Battery Board 1
- Stripped threads on Air Start Fittings on 1A-A and 1B-B Emergency Diesel Generators (EDGs)
- Failure of 2A CS Pump Breaker to close during Post-Modification Test (PMT) following routine maintenance

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following seven 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 re-assessed 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. Documents reviewed are listed in the attachment.

- Removal of 1A RHR Train for maintenance
- Troubleshooting of ground on 125 VDC Vital Battery Board 1
- Removal of Unit 1A ERCW Engineered Safeguard Feature (ESF) header for maintenance
- Failure of 2A CS pump to start following maintenance

- Testing of 2A 6.9-kV Shutdown Board
- Failure of A Train Auxiliary Air Compressor
- Removal of 2B Motor Driven AFW Train for maintenance

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

During the three non-routine evolutions identified below, the inspectors reviewed the operator logs, plant computer information, associated Problem Evaluation Reports (PERs), and conducted interviews with personnel involved to determine what occurred, how personnel responded, and if the response was in accordance with the associated procedures and training. The inspectors also evaluated the initiating causes of any unplanned activities for any personnel error contribution.

- Unit 2 Cooldown following Unit Trip for Fall 2003 refueling outage
- Shutdown of Unit 2 to repair generator hydrogen leak on January 16, 2004
- Unit 1 Automatic Reactor Trip on March 18, 2004

b. Findings

Introduction: The inspectors identified a green non-cited violation (NCV) for a self-revealing failure to follow procedures while shutting down Unit 2 in November 2003.

Description: On November 9, 2003, Unit 2 was tripped from approximately 20% of full power in accordance with General Operating Instruction 0-GO-6, Power Reduction From 30% Reactor Power to Hot Standby, Revision 22, in preparation for the Cycle 12 refueling outage. Following the trip, the RCS temperature decreased to where operators were not able to control the temperature by reducing AFW flow alone. At this point, in accordance with the Emergency Operating Procedure for recovery from a reactor trip, the operators closed the MSIVs and the RCS temperature returned to normal. The operators then proceeded to cool the unit by using the atmospheric relief valves with the MSIVs remaining closed.

In reviewing this incident, the licensee determined that the air-operated steam supply valves to the reheater portion of two of the Moisture Separator Reheaters (MSRs) were blocked open and a steam flow path existed through the reheaters to the main condenser. The steam flow through these paths caused the decrease in RCS temperature. The licensee later concluded that these valves had been blocked open since March 31, 2003. In order for this to have been a viable flow path, steam would also have had to pass an upstream motor-operated valve on each line. The licensee's investigation revealed that the motor-operated reheater steam supply valves had not been closed on all six of the MSRs before the unit was tripped. This was not in compliance with procedure 0-GO-6, which contained a step to close these valves on shutdown prior to tripping the unit.

Analysis: This finding was more than minor because it affected the availability of the power conversion system to handle an anticipated transient. In addition, the licensee had suspected a primary-to-secondary leak of 0.2 gallons per day in one of the steam generators and was monitoring RCS dose-equivalent-iodine closely due to a degraded fuel assembly. Therefore, closure of the MSIVs and loss of the condenser heat sink was not desirable. Because there was no loss of a safety function, no loss of a TS-required system or risk-significant maintenance rule system for greater than 24 hours, no increase in risk from external events, no detectable releases, and the calculated primary-to-secondary leakage did not increase, this finding is considered to be of very low significance (Green).

Enforcement: TS 6.8.1.a requires that procedures be implemented covering the activities in Regulatory Guide (RG)-1.33, Revision 2, Appendix A. Paragraph 2.i of Appendix A recommends procedures for general operating procedures for plant shutdown to hot standby. Licensee procedure 0-GO-6 provided instructions for shutting down the unit from 30% power to hot standby. Contrary to the instructions in procedure 0-GO-6, on November 9, 2003, the licensee failed to close the motor-operated reheater steam supply valves, as required by Step 5.1[g] of that instruction.

Because this violation was determined to be of very low safety significance (Green), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000328/2004002-01, Failure to Comply with Procedure for Shutting Down Unit 2. This violation is in the licensee's corrective action program as PER 03-016160-000.

1R15 Operability Evaluations

a. Inspection Scope

For the seven 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 that 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 03-019208-000, 480-Volt ERCW Motor Control Centers (MCCs) 1B and 2B Cross-connected
- PER 04-000264-000, Low Operating Temperatures on EDG 1B Engine 1 Compared to Engine 2

- PER 04-000243-000, Main Control Room Chiller A High Oil Temperature
- PER 04-770224-000, Damage to Intake Traveling Screens
- PER 04-000256-000, U2 T-Hot 3 on Loop 2 Activity is Causing Loop 2 Delta T Swings
- PER 04-770294-000, Vent Chute Closed in Unit 2 East Valve Vault Room
- PER 04-000019-000, Change to EOP FR-Z.1 Not Correct for Ice Condenser Plants

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- WO 03-009990-000, Replace 1A RHR Pump ABB Breaker with Siemens Breaker
- WO 03-006418-000, Inspect Security Backup Diesel Generator
- WO 04-771255-000, Source Check Failed on Radiation Monitor 1-RM-90-106
- WO 04-771801-000, Troubleshoot, Repair, or Replace 2A CS Pump Breaker and Handswitch
- WO 03-015520-000, Repair Unit 1 Turbine-Driven AFW Pump Steam Leak
- WO 04-771701-000, Troubleshoot and Repair Letdown Pressure Control Valve 2-62-81
- WO 04-772378-000, Replace U2 Eagle Protection Set 2 Test Signal Processing Card Seven

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the five 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, 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:

- 0-SI-EBT-250-100.4, Modified Performance Testing of 125-VDC Vital Batteries and 125-VDC Vital Battery Charger Test, Revision 14
- 2-SI-SXP-062-201.B, Centrifugal Charging Pump 2B-B Performance Test, Revision 10*
- 0-SI-NUC-000-126.0, Hot Channel Factor Determination, Revision 17
- 2-SI-OPS-82-007.A, Electrical Power System Diesel Generator 2A-A, Revision 31
- 0-SI-OPS-068-137.0, Reactor Coolant System Water Inventory, Revision 12

*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 temporary modification described in Temporary Alteration Control Form (TACF) 1-04-001-062, Crimp and Furmanite Drain Line Between Valves 1-62-541 and 1-62-542, Revision 0, 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.

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 drill on March 23, 2004, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) 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 three PIs listed below for the period from January 1 through December 31, 2003, for both units. To verify the accuracy of the PI data reported during that period, definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Cornerstone: Initiating Events

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

The inspectors reviewed the licensee's methods for compiling and reporting PIs to the NRC and discussed them with cognizant engineering personnel for the period of January through December 2003. The inspectors reviewed portions of the operations logs and raw PI data developed from monthly operating reports, reviewed Licensee Event Reports (LERs), and 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. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

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

1. Annual Sample Review of Problems With Pressurizer Power-Operated Relief Valves (PORVs) Popping Open When Block Valve Cycled

a. Inspection Scope

The inspectors reviewed licensee actions to resolve a recurring problem with the pressurizer PORVs opening momentarily when the associated block valves were cycled because the problem has existed for some time and occurred as recently as January 2004. The inspectors reviewed PERs back to May of 2002 and interviewed engineering personnel on this issue. Documents reviewed are listed in the attachment.

b. Findings and Observations

No findings of significance were identified; however, the inspectors noted that the licensee has yet to identify a definitive cause of the problem even though several PERs contained a completed apparent cause.

The inspectors observed that five of the reviewed PERs contained an apparent cause description. Two of the five attributed the cause to slugging of water against the seat of the PORV when the block valve was opened after being closed for approximately 50 minutes or more. The water condensed from steam trapped in the pipe between the block valve and the PORV. When the block valve was opened, the sudden repressurization due to steam from the pressurizer space rapidly forced the water into the PORV and caused it to momentarily open. Two of the PERs described a series of possible causes instead of an apparent cause. The final PER of the five attributed the cause to the design of the valve coil and springs, which were not capable of keeping the valve shut.

When the inspectors questioned the apparent differences among the different PERs, the licensee provided a letter, not referenced in a PER, from the PORV vendor that addressed the issue of spurious opening of the type of valve used as the PORV. The letter gave a description of testing done by the vendor and provided a recommendation to mount the valve with the bonnet pointing downward so that it remained filled with liquid. The letter included a report from the American Society of Mechanical Engineers

(ASME) that analytically determined that the phenomenon would most likely occur when a valve with air or steam in the bonnet was suddenly exposed to high pressure water. This was because the valve used system pressure to assist it in closing and when exposed to high pressure water the small openings in the valve plug would not allow the water to flow fast enough to prevent the sudden differential pressure between the valve inlet and the bonnet from opening the valve. The vendor recommended bonnet-down installation to avoid this.

However, the inspectors noted that the valves were installed as the vendor recommended yet the problem has continued to occur. From this the inspectors determined that the overall cause evaluation for the momentary opening of the PORVs when the block valve was cycled was not finalized. Without a definitive cause, the licensee was unable to identify corrective actions that would appropriately address the problem.

4OA3 Event Follow-up

Unit 1 Automatic Reactor Trip

On March 15, 2004, following an automatic Unit 1 reactor trip due to the spurious actuation of a generator main bank transformer protective relay, 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. A ground loop on a 250-VDC Battery Board caused the spurious relay actuation. The event was reported to the NRC as event notification (EN) 40589 and documented in the licensee corrective action program as PER 33278.

4OA4 Cross Cutting Aspects of Findings

The finding in Section 1R14 describes human performance errors where the licensee improperly blocked open the air-operated steam supply valves to the reheater portion of two of the MSRs and licensee operators failed to close the motor-operated reheater steam supply valves on all six of the MSRs before tripping the unit as part of a planned shutdown. Consequently, a steam flow path existed through the reheaters to the main condenser and RCS temperature decreased to the point where operators had to close the MSIVs to maintain control.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000327/2003005-01, Use of Alternate Source Term in Operability Evaluations

This URI was opened when the licensee used the 10-minute release time assumption found in RG 1.183, Alternate Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Plants, to justify that the Unit 1 EGTS remained operable with degraded annulus vacuum. The inspectors questioned this approach partly because the use of alternate source term methods was outside the current license basis for Sequoyah and, in accordance with 10 CFR 50.67, the licensee needed a license amendment to implement the methods.

In a letter, dated October 22, 2003, NEI submitted a draft white paper to the NRC, entitled “Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability,” that dealt with similar issues as this URI. On January 30, 2004, the NRC responded to that paper in a letter from Eric Leeds of the Office of Nuclear Reactor Regulation (NRR) to James W. Davis of NEI. In that letter, the NRC concluded that alternate source term methods could be used on operability determinations to evaluate control room in-leakage test results with some restrictions. The main restriction being that, unless already approved to use alternate source term methods, the licensee must still use current license basis acceptance criteria from the original source term. The letter also indicated that the position on control rooms could have broader applications, but recommended discussions on the specifics.

Following discussion with the Office of NRR, the inspectors considered the use of alternate source term methods acceptable in this case because the operability evaluation in question only used the timing portion of the alternate source term and that resulted in no dose consequences. However, the inspectors also determined that future uses would be evaluated on a case-by-case basis and that any use of alternate source term methods to justify continuing or recurring problems would be considered a de facto change to the license basis and would need a license amendment. There were no violations of NRC requirements. This URI is closed. Documents reviewed are listed in the attachment.

.2 (Closed) URI 05000327, 328/2002004-01, Review Licensee Evaluation of Changes Made to the Approved Fire Protection Program.

Introduction: A Severity Level IV (Green) NCV of 10 CFR 50.48(a) and the Unit 1 and 2 Operating License Conditions was identified for the licensee making an inappropriate change to the approved fire protection program (FPP). This change removed the requirement to implement fire watches for impaired fire protection systems and features.

Description: The URI was based on inspector’s observations that the licensee had modified their FPP concerning the establishment of fire watches as compensatory measures for degraded conditions. On November 19, 1998, the licensee implemented changes to their fire protection implementing procedures utilizing their fire protection license change process. These changes revised the FPP to permit the removal of fire suppression systems and/or fire rated barrier assemblies from service without compensatory measures being implemented (i.e., fire watches being posted) in affected plant areas. The licensee’s FPP is implemented, in part, to satisfy the separation and suppression requirements of 10 CFR 50, Appendix R, Section III.G. The Sequoyah Unit 1 Facility Operating License Condition 2.C (16) and Sequoyah Unit 2 Facility Operating License Condition 2.C (13) require, in part, that the licensee may make changes to the approved FPP without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown (SSD) in the event of a fire.

The inspectors reviewed the subject changes and licensee analysis and determined that the lack of a posted fire watch could adversely affect the ability to achieve and maintain SSD in the event of a severe fire in the affected area. This was based on recognition that the ability of the fire watch was not limited to fire identification in a timely manner, but also on mitigation actions an established fire watch could take in the event of fires. These could include such actions as the ability to close doors limiting fire exposure to adjacent areas and providing more timely fire detection capability in certain cases. The inspectors concluded that the licensee inappropriately used the fire protection program change process to revise the FPP to permit removing fire suppression systems and/or fire rated barrier assemblies from service without enhancing other elements as a compensatory measure. This constituted a change from the approved program that required NRC approval prior to implementation. However, no NRC approval was obtained by the licensee.

Analysis: The inspectors reviewed the situations where the licensee incorporated the change into their fire protection implementation planning and evaluated the potential impact on the safety-related structures and components. No risk significant impacts were identified. This fire protection change process finding was considered to potentially impede or impact the regulatory process. It was therefore dispositioned using the traditional enforcement process instead of the SDP and was assessed in accordance with guidance in Sections IV.A.1 through IV.A.4 and Section IV.B of the NRC's Enforcement Policy. Based on this guidance, this violation is classified as a Severity Level IV violation because it resulted in conditions that were evaluated as having very low safety significance. The issue was important because the licensee's change process for the FPP allowed degraded fire barriers to exist without previously established compensatory actions and this change was made without prior NRC approval. The inspectors concluded that this issue had a realistic likelihood of affecting safety because the licensee's failure to properly evaluate the removal of fire watch posting requirements could adversely affect or degrade the ability for achieving and maintaining SSD from the main control room, local shutdown stations, or alternate shutdown stations. However, the inspectors determined that this finding was of very low significance because, based on an assessment of the impacts of the identified fire protection features removed from service, the licensee's overall SSD capabilities, and related FPP features (fire brigade) remained adequate to achieve and maintain SSD conditions. Therefore, following NRC management review, this finding is also characterized as Green. On March 25, 2004, Revision 16 of the Sequoyah Fire Protection Report became effective which returned the licensee to their original compensatory measures with regard to fire watches for inoperable equipment outside of containment. This corrective action was documented in PERs 03-011649-000 and 03-11569-000, which addressed this issue.

Enforcement: 10 CFR 50, Appendix R, Section III.G, the Sequoyah Unit 1 Facility Operating License Condition 2.C (16) and Sequoyah Unit 2 Facility Operating License Condition 2.C (13) require, in part, that the licensee may make changes to the approved FPP without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown (SSD) in the event of a fire. Contrary to the above, on November 19, 1998, the licensee revised the FPP to permit the removal of fire suppression systems and/or fire rated barrier assemblies from service without compensatory measures being implemented (i.e., fire watches being posted) in affected plant areas. Because this change to the FPP is of very low safety significance and has been entered into the licensee's corrective action program (PERs 03-011649-000 and 03-11569-000), this violation is being treated as an NCV in accordance with Section VI.A.1 of the NRC's Enforcement Policy: NCV 05000327, 328/2004002-02, Inappropriate Change to the Approved Fire Protection Program. Documents reviewed are listed in the attachment.

.3 <u>(Closed) Notice of Violation (VIO) 05000327, 328/2001007-02,</u> Discrimination Against a Contract Security Officer (EA 01-317)

The inspector reviewed the licensee's response dated February 1, 2002, which addressed the subject violation. The response indicated that the licensee's investigation of the condition substantiated the issue. Corrective actions for the Violation included a variety of awareness training measures such as standdown meetings reinforcing the need to follow procedures and build a safety conscious work environment. In addition, training was provided by the TVA Concerns Resolution Staff designed to preclude similar issues and an independent consultant conducted an attitude/climate survey of the licensee's workforce which resulted in a variety of improvement recommendations. The inspectors concluded that the licensee adequately addressed the issue. No additional concerns were identified. This violation is closed.

4OA6 <u>Meetings, including Exit</u>

.1 <u>Exit Meeting Summary</u>

On April 1, 2004, the resident inspectors presented the inspection results to Mr. Rick Purcell and other members of his staff, who acknowledged the findings.

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

.2 <u>Annual Assessment Meeting Summary</u>

Subsequent to the end of this inspection period, on April 21, 2004, the NRC's Chief of Reactor Project's Branch 6 and the Senior Resident Inspector assigned to the Sequoyah Nuclear Plant met with the Tennessee Valley Authority (TVA) to discuss the NRC's Reactor Oversight Process (ROP) and the Sequoyah annual assessment of safety performance for the period of January 1, 2003 - December 31, 2003. The major topics addressed were: the NRC's assessment program, the results of the Sequoyah assessment, and planned NRC inspection activities. Attendees included Sequoyah site management, members of site and corporate staff, and local county and state emergency management agency representatives.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML041100579. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION
PARTIAL LIST OF PERSONS CONTACTED

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
J. Traister, 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 and Closed

05000328/2004002-01	NCV	Failure to Comply with Procedure for Shutting Down Unit 2 (Section 1R14).
05000327, 328/2004002-02	NCV	Inappropriate Change to the Approved Fire Protection Program (Section 4OA5.2).

Closed

050000327/2003005-01	URI	Use of Alternate Source Term in Operability Evaluations (Section 4OA5.1).

05000327, 328/2002004-01	URI	Review Licensee Evaluation of Changes Made to the Approved Fire Protection Program (Section 4OA5.2).
05000327, 328/2001007-02	VIO	Discrimination Against a Contract Security Officer (EA 01-317) (Section 4OA5.3).

LIST OF DOCUMENTS REVIEWED

Section R11: Licensed Operator Requalification

Simulator Exam Guide 0PL273S0401, Pressurizer Vapor Space Accident, Revision 0

Section R12: Maintenance Rule Implementation

0-PI-EBT-250-001.0, Ground Detection and Isolation on Battery Systems 82 and 250
WO 04-770379-000, 125 VDC Vital Battery Board I Ground Indication
WO 03-019184-000, Test "T" Attached to EDG Engine 1A1 Starting Air Has Stripped
WO 03-019362-000, Repair Stripped Threads on 1B-1 EDG Starting Air Test Connections
WO 04-771801-000, Troubleshoot Control Circuit for 2A CS Pump
PER 03-019185-000, EDG 1A-1 Starting Air Test "T" Stripped
PER 03-019365-000, EDG 1B-2 Starting Air Test "T" Stripped, Repeat Occurrence
PER 04-000556-000, 2A CS Pump Failed to Start During Section XI Test
1,2-45N703-1, Wiring Diagram, 125VDC Vital Battery Board I Single Line Sheet-1

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

0-SO-20-10, Auxiliary Building Ventilation Systems
PER 03-000735-000, Auxiliary Air Compressor "A" tripped 3 Times on Low Oil Reservoir Level
1-TO-2004-0007, Clearance 1-30-0078B-W/W, ESF Pen Rm Clrs ERCW Hdr - Clam Inspection
O&SSDM 4.8 - Critical Evolution Meeting, Revision 0

Section R15: Operability Evaluations

1-SI-OPS-082-007.B, Electrical Power System Diesel Generator 1B-B, Revision 29
0-PI-OPS-000-006.0, Freeze Protection, Revision 34
WO 03-009265-000, Balance the Woodward EDG-13P actuators on 1B-B EDG
WO 04-771095-000, U2 T-Hot 3 on Loop 2 is Very Active Causing Loop 2 Delta T Swings
PER 03-002446-000, Common Cause Evaluation for Freeze Protection
47W920-43, Mechanical Heating, Ventilating, and Air Conditioning, Revision 2
Westinghouse EPG FR-Z.1, Response to High Containment Pressure, Revision 1C

Section R19: Post Maintenance Testing

0-PI-OPS-000-677.0, Operability Performance of Security Backup Diesel Generator, Revision 15
PER 04-000556-000, 2A CS Pump Failed to Start During Section XI Test
PER 04-000524-000, Received U2 MCR Alarm for Low Press LTDN Flow High Pressure
PER 04-000701-000, Placed 2-HIC-62-81A in Manual Due to Excessive Pressure Swings
PER 04-00750-000, Perform an Extent of Condition to Determine if End Device Testing Has Been Waived

Section R22: Surveillance Testing

0-SI-NUC-092-079.0, Power Range Monitor Channel Calibration By Incore/Excore Axial Imbalance Comparison, Revision 10
TI-53, Flux Mapping, Revision 31
1,2-45N703-2, Wiring Diagram, 125 VDC Vital Battery Board II Single Line Sheet-2
SQN-CPS-057, Vital Control Power System Loading Channel I, and Continuous Loading Evaluation of Protective Devices in the 120 VAC Vital Instrument Power Boards, Revision 56
SQN-CPS-058, Vital Control Power System Loading Channel II, and Continuous Loading Evaluation of Protective Devices in the 120 VAC Vital Instrument Power Boards, Revision 56
March 11, 2004 U1 RCS Leakage Rate Report
March 9, 2004 U2 RCS Leakage Rate Surveillance Package Key: P1191
DCN D-21441 & 21405, Units 1 and 2 RCP No. 3 Leakoff Path Modification

Section 4OA1: Performance Indicator Verification

LER 50-327/2003-001, Manual Reactor Trip as a Result of a Main Generator Trip and Loss of Load
LER 50-328/2003-001, Reactor Trip Signal as a Result of a Low-Low Steam Generator Level
LER 50-328/2003-002, Limiting Conditions for Operation 3.0.3 Was Entered When Two Refueling Water Storage Tank Level Transmitters Failed During Cold Weather Conditions
LER 50-328/2003-003, Excessive Leakage of a Containment Purge System Containment Isolation Valve
LER 50-328/2003-004, Reactor Trip Resulting From a Neutral Over-Current Condition on the 2B Hotwell Pump and a Failure to Perform a Technical Specification Required Action
LER 50-328/2003-005, Reactor Trip Resulting From a Spurious Turbine Vibration Trip Signal
LER 50-328/2003-006, Failure to Meet Technical Specification Limiting Condition for Operation Action Time for the Component Cooling System

Section 4OA2: Identification and Resolution of Problems

SPP-3.1, Corrective Action Program, Revision 6
PER 04-000556-000, 2A CS Pump Failed to Start During Section XI Test
PER 04-00750-000, Perform an Extent of Condition to Determine if End Device Testing Has Been Waived
PER 04-000475-000, ERCW Pump P-B Breaker Failure to Close for PMT
PER 01-009568-000, Consolidation of Siemens Breaker Issues
PER 03-008296-000, Consolidation of Additional Siemens Breaker Issues
PER 04-000109-000, PORVs Popped Open During Functional Test When Block Valve Reopened
PER 03-015456-000, During Functional Test PORV Opened and Reclosed When Block Valve Opened
PER 03-012288-000, NSRB Questioned the Basis for Accept-as-is Assessment of PORV Momentarily Opening Following Block Valve Opening
PER 03-010168-000, During Channel Calibration the PORV Popped Open and Reclosed
PER 03-001753-000, During Functional Test the PORV Popped Open and Remained Open for 2-3 Seconds
PER 02-014329-000, Pressurizer Enclosure Temperature Decreased When Block Valve Closed

PER 02-009792-000, Unit 1 PRT Inleakage Indicates 0.06 gpm
PER 02-005477-000, Pressurizer PORV Failed its Stroke Time Going Closed
1052020-3, Target Rock Solenoid Power Operated Relief Valve, Sheet 2, Revision E
1-47W465-7, Reactor Coolant Aux & Misc Piping, Revision 2
Letter From V. Liantonio to D.K. Vater, Spurious Opening of Pilot Operated Solenoid Valve, dated October 2, 1984

Section 4OA5: Other Activities

PER 03-011569-000, Verify Appropriateness of Fire Protection Program Change Referenced in NRC URI 50-327, 328/02-04-01
PER 03-011649-000, NRC Concerns with Fire Protection Compensatory Actions
Sequoyah Nuclear Plant Fire Protection Report, Revision 16