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

October 26, 2007

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

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2007004 AND 05000328/2007004

Dear Mr. Campbell:

On September 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on October 8, 2007, with Mr. Tim Cleary and other members of your staff.

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

This report documents one self-revealing finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Sequoyah Nuclear Plant.

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

Sincerely,
/RA/

Stephen C. O'Connor, Acting Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000327/2007004 and 05000328/2007004
w/Attachment: Supplemental Information

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

James R. Douet
Vice President
Nuclear Support
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

H. Rick Rogers
Vice President
Nuclear Engineering &
Technical Services
Tennessee Valley Authority
Electronic Mail Distribution

Timothy P. Cleary
Site Vice President
Sequoyah Nuclear Plant
Electronic Mail Distribution

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

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

Beth A. Wetzel, Manager
Corporate Nuclear Licensing and
Industry Affairs
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

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

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

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

County Mayor
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Larry E. Nicholson, General Manager
Licensing and Industry Affairs
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Ann Harris
341 Swing Loop
Rockwood, TN 37854

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

Letter to William R. Campbell, Jr. from Stephen C. O'Connor dated October 26, 2007

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2007004 AND 05000328/2007004

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Bob Pascarelli, NRR
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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:	05000327/2007004 and 05000328/2007004
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	July 1, 2007 - September 30, 2007
Inspectors:	S. Freeman, Senior Resident Inspector M. Speck, Resident Inspector R. Taylor, Reactor Inspector (Section 4OA5)
Approved by:	S. O'Connor, Acting Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2007-004, IR 05000328/2007-004; 07/01/2007 - 09/30/2007; Sequoyah Nuclear Plant, Units 1 and 2; Maintenance Effectiveness.

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation of Technical Specification 6.8.1a was identified for failure to follow procedures when performing maintenance on the intake fire damper for Emergency Diesel Generator 2A. Because of this failure, the damper inadvertently closed and rendered the emergency diesel generator incapable of starting for 2.5 hours. The licensee entered this issue into their Corrective Action Program as Problem Event Report (PER) 129463.

  This finding was more than minor because it affected the mitigating systems cornerstone objective of availability by rendering the emergency diesel generator incapable of starting and was associated with the equipment performance attribute. This finding was of very low safety significance because redundant equipment was available to provide the safety function. The finding had a cross-cutting aspect in the area of Human Performance because the cause of the finding was related to the Work Practices aspect of communicating expectations regarding procedural compliance (H.4(b)). (Section 1R12)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100% rated thermal power (RTP) until September 3, 2007 when it began a gradual power reduction in preparation for a refueling outage. Unit 1 was at 79% RTP at the end of the inspection period.

Unit 2 operated at or near 100% RTP for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

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

- Unit 1 Emergency Core Cooling System (ECCS) Train B during maintenance on Centrifugal Charging Pump (CCP) 1A
- Unit 1 ECCS and Containment Spray (CS) Train B during maintenance on ECCS and CS Train A Component Outage
- Unit 1 Residual Heat Removal (RHR) Train A during Train B maintenance

Complete System Walkdown. The inspectors performed a complete system walk-down of the Unit 1 Safety Injection (SI) system to verify proper equipment alignment, to identify any discrepancies that could impact the function of the system and increase risk, and to verify that the licensee properly identified and resolved equipment alignment problems that could cause events or impact the functional capability of the system.

The inspectors reviewed the Updated Final Safety Analysis Report (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 system equipment lineup and the correct lineup. In addition, the inspectors reviewed outstanding maintenance work requests and design issues on the system to determine whether any condition described in those work requests could adversely impact current system operability. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

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

- Control Building Elevation 669 (Mechanical Equipment Room, 250-VDC Battery and Battery Board Rooms)
- Control Building Elevation 706 (Cable Spreading Room)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Emergency Diesel Generator (EDG) Building
- Auxiliary Building Elevation 714 (Corridor)
- Control Building Elevation 732 (Mechanical Equipment Room and Relay Room)
- Auxiliary Building Elevation 690 (Corridor)
- Auxiliary Building Elevation 734 (Switchgear Rooms and Battery Board Rooms)
- Auxiliary Building Elevation 749 (Reactor Board Rooms, Battery Rooms, Transformer Rooms, and Mechanical Equipment Rooms)

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed as-found simulator training on August 6, 2007. The training involved a trip of the 1C #3 Heater Drain Tank Pump followed by a trip of the 1A Main Feed Pump resulting in a plant runback which included a stuck control rod. After the plant was stabilized, a reactor coolant pump (RCP) tripped without an expected reactor trip. Operators initiated a manual reactor trip and a steam generator tube leak ensued which degraded into a tube rupture requiring manual initiation of safety injection and plant cooldown and depressurization. Anomalies included a failure of the turbine-driven auxiliary feedwater (AFW) pump governor and the A-train motor-driven AFW pump failed to start. 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 observed the evaluators' critique and reviewed simulator fidelity to verify that it matched actual plant response. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following two maintenance activities to verify the effectiveness of the activities in terms of: 1) appropriate work practices; 2) identifying and addressing common cause failures; 3) scoping in accordance with 10 CFR 50.65 (b); 4) characterizing reliability issues for performance; 5) trending key parameters for condition monitoring; 6) charging unavailability for performance; 7) classification in accordance with 10 Code of Federal Regulations (CFR) 50.65(a)(1) or (a)(2); 8) appropriateness of performance criteria for 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.

- Elevated Temperatures in Vital Battery Rooms
- PER 129614, Damper Closure During Maintenance Resulted in Lockout of EDG 2A

b. Findings

Introduction: A green self-revealing NCV was identified for failure to follow procedures when performing maintenance on the EDG 2A intake fire damper. Because of this failure, the EDG was rendered incapable of starting for 2.5 hours.

Description: On August 24, 2007, while inspecting the EDG 2A intake fire damper, a fire operations foreman found the damper to be improperly aligned. Instead of initiating a minor maintenance work order, as required by Appendix A of Procedure SPP-6.1, Work Order Process Initiation, Revision 4, the foreman classified the misaligned damper as tool pouch maintenance, which did not require a Work Order (WO), and instructed two fire operators to adjust it. While making the adjustment the damper closed and, because a ventilation exhaust fan was running, caused a high crankcase pressure lockout of EDG 2A. Because of the improper use of tool pouch maintenance the control room was not notified before work began and the EDG was subsequently made inoperable for approximately 2.5 hours.

Analysis: The finding was more than minor because it caused EDG 2A to be unavailable, and inoperable, for 2.5 hours. This negatively affected the mitigating systems cornerstone objective of availability and was associated with the equipment performance attribute. The finding was of very low safety significance (Green) because redundant equipment was available to provide the safety function. The finding had a cross-cutting aspect in the area of Human Performance because the cause of the finding was related to the Work Practices aspect of communicating expectations regarding procedural compliance. The individual involved decided tool pouch maintenance was acceptable without fully reading SPP-6.1 due to perceived time pressure (H.4(b)).

Enforcement: TS 6.8.1a requires procedures to be implemented covering the activities in Regulatory Guide 1.33, Revision 2, Appendix A. Paragraph 9a of Appendix A, requires that maintenance affecting the performance of safety-related equipment be properly preplanned and performed in accordance with written procedures. Licensee Procedure SPP-6.1 required workers, before initiating maintenance, to determine if tool pouch maintenance could be performed and required that tool pouch maintenance NOT interrupt the flow of fluid, air or current to standby equipment, or defeat the automatic operation of protective systems. Contrary to the above, on August 24, 2007, licensee personnel classified work on the EDG 2A intake damper as tool pouch maintenance even though the work stopped ventilation flow to the EDG and caused it to be unavailable for 2.5 hours. Because this violation was determined to be of very low safety significance (Green), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000327,328/2007004-01, Failure to Follow Procedure when Adjusting EDG Intake Fire Damper. This violation is in the licensee’s corrective action program as PER 129463.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following five activities to verify that the appropriate risk assessments were performed prior to removing equipment from service for maintenance. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65 (a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee’s risk assessment tool and risk categories in accordance with Procedure SPP-7.1, On-Line Work Management, Revision 9, and Instruction 0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 8. Documents reviewed are listed in the Attachment.

- Unit 1 ECCS and CS Systems Outage
- Removal of Unit 1 Turbine-Driven AFW Pump from service for testing
- Removal of Unit 2 Turbine-Driven AFW Pump from service for testing

- Elevated Offsite Power Risk due to Maintenance on two 500kV Lines and Emergent Work on SI Pump 2B
- Component Cooling System (CCS) Alignment Changes to Perform Pump C-S Discharge Check Valve Testing

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

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

- PER 128514, Unplanned EDG 1B-B Inoperability Due to Fire Damper Failing Closed - Common Cause Evaluation
- PER 127894, Improper Valve Operation Results in Inoperable CS System Train 1A During B-Train Pump Performance Testing
- PER 128345, Refueling Water Storage Tank (RWST) Moat Overflow Arrangement
- PER 127610, Compensatory Actions for Meteorological Monitoring Tower Out of Service

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- WO 07-777527-000, Replace Unit 2 Leading Edge Flow Monitor Flow Path Transducers
- WO 06-781520-000, Lubricate, Bridge and Meggar Safety Injection Pump 1B
- WO 07-777950-000, Unit 1 Cold Leg Accumulator #2 Level Indicator Deviating High
- WO 07-771289-000, Perform Motor Operated Valve Tests (MOVATs) on Valve 1-FCV-74-21
- WO 04-780658-000, Replace 6.9kV Alternate Supply Breaker to 2B2-B 480V Shutdown Board Transformer

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

- 0-SI-NUC-000-007.0, Measurement of the At-Power Moderator Temperature Coefficient, Revision 13
- 2-SI-OPS-030-286.0, Cumulative Time That Containment Purge Supply and Exhaust Isolation Valves are Open, Revision 3
- 0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 5 (Unit 2)*
- 0-SI-EBT-250-100.2, 125-VDC Vital Battery Quarterly Operability (Battery II)
- 2-SI-TDC-202-235.B, 6.9kV Shutdown Board Loss of Voltage, Overvoltage, and Degraded Voltage Relay Calibration - Train B, Revision 13
- 1-SI-MIN-061-108.0, Ice Condenser Intermediate Deck Doors - Visual Inspection, Lift Test and Ice Removal, Revision 2
- 2-SI-SXP-074-201.A, Residual Heat Removal Pump 2A-A Performance Test, Revision 13**

*This procedure included testing of a large containment isolation valve.
**This procedure included in-service testing of a risk-significant pump.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of a routine licensee emergency drill on July 26, 2007, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room to verify that event classification and notifications were done in accordance with Procedure EPIP-1, Emergency Plan Classification Matrix, Revision 39. The inspectors also attended the licensee critique of the drill 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

4OA2 Identification and Resolution of Problems

.1 Daily Review

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

.2 Annual Sample Review of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the operator workaround (OWA) program to verify that OWAs were identified at an appropriate threshold, were entered into the CAP, and that corrective actions were appropriate and timely. Specifically, the inspectors reviewed the licensee's workaround list and repair schedules, performed CAP word searches, conducted tours and interviewed operators and operations department support staff. Additionally, the inspectors checked for undocumented workarounds by observing operators perform rounds, reviewed operator deficiency lists, reviewed appropriate system health documents, and attended a plant health committee meeting. The inspectors evaluated all workarounds for their aggregate impact.

b. Findings

No findings of significance were identified. However, the inspectors noted that one temporary plant modification was not listed as an OWA. The inspectors reviewed WO 07-779317-000 and Temporary Alteration Control Form (TACF) 1-07-005-001 which installed a temporary electrical jumper in the control circuitry for valve 1-FSV-1-98, the 1C1 Main Steam Reheater (MSR) high pressure main supply valve. The jumper defeated the interlock which closes the main supply valve should the associated bypass supply valve, 1-FCV-1-298, come off its open seat. It also defeated a feature that automatically closes the supply valve on a turbine trip. Unexpected bypass valve positioner movement previously resulted in the main supply valve shutting and subjecting the plant to a several megawatt drop in thermal power and an unplanned reactivity transient. Prior to installing the jumper, operators were briefed on the effects of the alteration and that following a unit trip, operators may have to take action to shut the MSR valve if plant cooldown is excessive. Licensee procedure OPDP-1, Conduct of Operations, Revision 8, defined a Priority 1 OWA as one where the "Operator must take compensatory actions during response to accidents or transients." The inspectors questioned whether this alteration resulted in a Priority 1 OWA and when the deficiency was expected to be corrected. The inspectors also interviewed operators on what actions may be required following a plant trip as a result of this alteration. All responses were consistent and indicated they fully understood the implications of the alteration and the actions required. The work order was administratively coded as an OWA following the inspectors' questions. The deficiency was scheduled for correction during the next scheduled plant outage in October 2007. Inspectors did not identify any other deficiencies not classified as OWAs and that all identified OWAs were scheduled to be corrected. The inspectors determined that failing to classify a WO as an OWA was of minor significance because operators were aware of required actions as a result of the alteration, contingency actions of procedure ES-0.1, Reactor Trip Response included appropriate operator actions, the deficiency was scheduled for timely correction, and no additional corrective actions were required.

4OA3 Event Followup

(Closed) Licensee Event Report (LER) 05000327/2006-001-00, Potential Loss of Component Cooling Water to the Seal Water Heat Exchanger During an Appendix R Fire

On February 9, 2006, during an operating experience review, licensee personnel identified a scenario in which a 10 CFR 50 Appendix R fire event could result in a loss of CCS to the chemical and volume control system seal water heat exchanger. This loss of CCS would result in high suction temperature on the running CCP causing a lack of adequate suction head. The temperature could be high enough to potentially damage both the CCP and the RCP seals. The enforcement aspects of this finding are discussed in Section 4OA7. This LER is closed. This LER was inadvertently closed in the previous NRC Integrated Inspection Report (05000327/2007003 and 05000328/2007003) under section 4OA3 Event Followup.

4OA5 Other Activities

(Closed) Unresolved Item (URI) 05000327/2007002-01, Heat Sink Issues

As described in URI 05000327/2007002-01, NRC inspectors identified an issue related to the adequacy of licensee corrective actions related to the Essential Raw Cooling Water (ERCW) flow degradation events of the summer of 2006. This URI was left open pending completion of the next molluskicide treatment scheduled for May 2007.

Upon further review, the NRC has determined that the licensee has taken steps to ensure the ERCW system performance was not degraded due to clam intrusions. This determination was made after a review of the results of the May 2007 molluskicide treatment and flow test which did not identify instances of flow reduction due to clam shells in the ERCW system. Because the previous ERCW system flow degradation did not affect the availability of the system to perform its design basis function and the licensee's corrective actions were appropriate, the finding is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This URI is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 8, 2007, the resident inspectors presented the inspection results to Mr. Tim Cleary and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV. Documents reviewed are listed in the Attachment.

Sections 2.C(16), for Unit 1 and 2.C(13), for Unit 2, of the Operating License require the licensee to implement and maintain all the provisions of the approved fire protection plan discussed in the UFSAR. Part III of that plan describes the safe shutdown capability and requires one CCP to be operational to provide for RCS makeup and RCP seal integrity for hot shutdown. Contrary to this, from the time of the Appendix R upgrade effort in 2000 until February 2, 2006, the safe shutdown analysis failed to provide for CCS pump availability for hot shutdown for all required fires. The failure of CCS would result in a loss of RCS makeup and RCP seal cooling. This issue is in the licensee's corrective action program as PER 95602. This finding was determined to be of very low safety significance (Green) because of the very low frequency of fires needed to fully disable the CCS and because one train of the emergency core cooling system equipment would be free from fire damage and available to mitigate a loss of coolant accident through the RCP seals.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Bodine, Chemistry/Environmental Manager
D. Boone, Radiation Protection Manager
C. Church, Plant Manager
T. Cleary, Site Vice President
K. Jones, Engineering Manager
Z. Kitts, Licensing Engineer
G. Morris, Licensing and Industry Affairs Manager
M. Palmer, Operations Manager
K. Parker, Maintenance and Modifications Manager
J. Proffitt, (Acting) Site Licensing Supervisor
R. Reynolds, Site Security Manager
N. Thomas, Licensing Engineer
K. Wilkes, Emergency Preparedness Manager

NRC personnel:

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

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000327,328/2007004-01	NCV	Failure to Follow Procedure when Adjusting EDG Intake Fire Damper (Section 1R12)

Closed

05000327/2006001-00	LER	Potential Loss of Component Cooling Water to the Seal Water Heat Exchanger During an Appendix R Fire (Section 4OA3)
05000327/2007002-01	URI	Heat Sink Issues (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section R04: Equipment Alignment

1-47W845-6, Mechanical Flow Diagram, Essential Raw Cooling Water, Revision 19
System Health Report Cards - Safety Injection, Unit 1- System 063, FY2005-2007
System 063 Safety Injection Completed Work Order Listing 7/1/2005-7/18/2007
System 063 Safety Injection Open Work Orders as of 7/19/2007
UFSAR Section 6.3, Emergency Core Cooling System
1-47W811-1, Flow Diagram Safety Injection System, Revision 71
1-47K435-502-01, Isometric Safety Injection Pump 1A-A Drains and Vents, Revision 1
1-47K435-503-01, Isometric Safety Injection Pump 1B-B Drains and Vents, Revision 1
NRC Information Notice 2006-26: Failure of Magnesium Rotors in Motor-Operated Valve Actuators
1-SO-63-5, Attachment 1, Emergency Core Cooling System Power Checklist 1-63-5.01, Change 19
1-SO-63-5, Attachment 2, Emergency Core Cooling System Valve Checklist 1-63-5.01, Change 31

Section R11: Licensed Operator Requalification

AOP-C.01, Rod Control System Malfunctions, Revision 17
AOP-S.04, Loss of Normal Feedwater, Revision 12
E-0, Reactor Trip or Safety Injection, Revision 29
E-3, Steam Generator Tube Rupture, Revision 16
EPIP-1, Emergency Plan Classification Matrix, Revision 39
ES-0.5, Equipment Verifications, Revision 0

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