March 10, 2000

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: NRC INTEGRATED INSPECTION REPORT NO. 50-327/00-01 AND 50-328/00-01

Dear Mr. Scalice:

On February 12, 2000, the NRC completed an inspection at your Sequoyah 1 & 2 reactor facilities. The enclosed report presents the results of this inspection. The inspection was an examination of 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. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, the inspection covered routine resident inspections.

Based on the results of this inspection, the NRC identified one issue of low safety significance that has been entered into your corrective action program and is discussed in the summary of findings and in the body of the attached inspection report. The issue involved a violation of NRC requirements, but because of its low safety significance the violation is not cited. If you contest this non-cited violation, 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 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.

The NRC received your response dated February 22, 2000, to a Notice of Violation which was issued in a letter dated January 26, 2000. We have evaluated your response and found that it meets the requirements of 10 CFR 2.201. We will examine the implementation of your corrective actions during future inspections.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Paul E. Fredrickson, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: NRC Inspection Report
w/attached TIA 99-02
and NRC's Revised Reactor
Oversight Process

cc w/encl:
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Nuclear Operations
Tennessee Valley Authority
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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:	50-327/00-01, 50-328/00-01
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 & 2
Location:	Sequoyah Access Road Hamilton County, TN 37379
Dates:	January 2 through February 12, 2000
Inspectors:	Russell Gibbs, Senior Resident Inspector D. Starkey, Resident Inspector R. Telson, Resident Inspector W. Bearden, Reactor Inspector (Sections 4OA4.3, 4OA4.4 and 4OA4.5)
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

Sequoyah Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-327/00-01, 50-328/00-01

The report covers a six-week period of resident inspection. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the NRC's Significance Determination Process, as discussed in the attached summary of the NRC's Revised Reactor Oversight Process.

Mitigating Systems

- Green. A non-cited violation was issued for the failure of maintenance personnel to follow procedures which resulted in the unplanned simultaneous unavailability of both trains of the safety injection system for a period of approximately 30 minutes and entry into Technical Specification 3.0.3 for about two and a half hours. Specifically, maintenance technicians changed the oil in the 2B-B safety injection pump motor, which rendered it inoperable, while the 2A-A safety injection pump was tagged out of service for routine maintenance. The technicians should have changed the oil in the 2A-A safety injection pump. The risk significance of this finding was low due to the brief time that both safety injection pumps were out-of-service and that safety injection from other emergency core cooling system equipment was maintained (Section 1R14.2).

Report Details

Unit 1 operated throughout the inspection period at or near 100 percent power until, on January 27, it began a power coast down for the scheduled February 2000 Cycle 10 refueling outage. Unit 1 ended the period at 86 percent power.

Unit 2 began the inspection period at 100 percent power. On January 18, a reactor trip and safety injection occurred. The unit was placed in Mode 3 and remained there until January 20. The unit was returned to 100 percent power on January 22 and operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

1R03 Emergent Work

a. Inspection Scope

The inspectors evaluated the licensee's work prioritization and risk determination associated with selected activities, listed below, to determine, as appropriate, whether necessary steps were planned, controlled, and executed.

- Troubleshooting and repair of the 120 VAC vital instrument power inverter 2-IV static switch following a reactor trip which was triggered, in part, by its failure
- Replacement of the auxiliary feedwater pump 2A-A circuit breaker following a failure of the breaker to close and remain closed following an automatic close signal
- Replacement of steam line pressure transmitter 2-PT-1-27B, following a safety injection (SI) and main steam line isolation triggered, in part, by a spurious low pressure signal generated by the transmitter

b. Observations and Findings

No findings were identified and documented through this inspection.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted equipment alignment partial walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with the other train or system inoperable or out of service. The walkdowns included, as appropriate, consideration of plant procedures and reviews of documents to determine correct system lineups, and verification of critical components to identify any discrepancies which could affect operability of the redundant train or backup system.

! The three operable emergency diesel generators (EDGs) were walked down while the redundant 2A-A EDG was out-of-service

! The A-A train of the emergency gas treatment system was walked down while the redundant 2B-B train was out of service

! The 2B-B SI train was walked down while the 2A-A train was out of service

b. Observations and Findings

No findings were identified and documented through this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the auxiliary instrument room and the computer room to evaluate, as appropriate, conditions related to (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R09 Inservice Testing of Pumps and Valves

a. Inspection Scope

The inspectors reviewed inservice testing of selected risk significant mitigating system pumps and valves, listed below, to evaluate the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program to determine equipment availability and reliability. The inspectors evaluated, as appropriate, (1) testing procedures, (2) acceptance criteria, (3) testing methods, (4) compliance with the licensee's inservice testing program, technical specifications (TS), and code requirements, (5) range and accuracy of test instruments, and (6) required corrective actions.

! Surveillance Instruction 1-SI-SXP-003-201.S, Turbine Driven Auxiliary Feed Water Pump 1A-S Performance Test

! 2-SI-SXV-003-219.0, Auxiliary Feedwater Check Valve Test During Operation

b. Observations and Findings

No findings were identified and documented through this inspection.

1R11 Licensed Operator Re-qualification

a. Inspection Scope

The inspectors observed operator performance in the plant's simulator during licensed operator retraining. In addition, the inspectors verified that the training program included high risk operator actions, emergency plan implementation, and lessons learned from previous plant experiences.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R13 Maintenance Work Prioritization and Control

a. Inspection Scope

The inspectors reviewed the licensee's control of plant risk and configuration through the review of selected structures, systems, and components (SSCs), listed below, within the scope of the maintenance rule or which were otherwise risk-significant. Emphasizing potential high risk configurations and high priority work items, the inspectors evaluated, as appropriate, (1) effectiveness of the work prioritization and control; (2) level of maintenance support; (3) assessment of integrated risk of the work backlog; and (4) safety assessments and/or management activities performed when SSC's are taken out of service.

! 120V AC vital instrument power inverter 2-IV

! Containment spray pump 2B-B

! EDG 2A-A

b. Observations and Findings

No findings were identified and documented through this inspection.

1R14 Non-routine Plant Evolutions

.1 Unit 2 Reactor Trip And Safety Injection

a. Inspection Scope

Personnel performance was evaluated during a January 18 event in which a momentary interruption of 120 VAC power to vital instrument bus 2-IV caused a Unit 2 reactor trip which was complicated by a partial SI (A-train only), a main steam line isolation, and a failure of motor-driven AFW pump 2A-A to start. In addition, inspectors evaluated personnel performance during the subsequent January 20 Unit 2 restart and power ascension.

As appropriate, the inspectors: (1) reviewed operator logs, plant computer data, or strip charts to determine what had occurred and how the operators responded; (2) determined if operator responses were in accordance with the response required by procedures and training; (3) evaluated the occurrence and subsequent personnel response using the significance determination process (SDP); and (4) confirmed that personnel performance deficiencies were captured in the licensee's corrective action program.

b. Observations and Findings

During the trip there were findings related to operation of the emergency raw cooling water and main steam systems. These findings are discussed in Special Inspection Report 50-328/00-03. During the restart and power ascension, no findings were identified and documented through this inspection.

.2 Technical Specification 3.0.3 Entry Due to Both Safety Injection Pumps Inoperable

a. Inspection Scope

The inspectors followed-up on a January 26 event which resulted in the simultaneous inoperability of both Unit 2 SI pumps. As appropriate, the inspectors: (1) reviewed operator logs, plant computer data, or strip charts to determine what had occurred and how the operators responded; (2) determined if operator responses were in accordance with the response required by procedures and training; (3) evaluated the occurrence and subsequent personnel response using the SDP; and (4) confirmed that personnel performance deficiencies were captured in the licensee's corrective action program.

b. Observations and Findings

A non-cited violation (NCV) was identified for the failure of maintenance personnel to follow procedures which resulted in the unplanned simultaneous unavailability of both trains of the SI system for a period of approximately 30 minutes and entry into TS 3.0.3 for about two and a half hours. Specifically, maintenance technicians changed the oil in the SI pump 2B-B motor, which rendered it inoperable, while SI pump 2A-A was tagged out-of-service for routine maintenance. The technicians should have changed the oil in SI pump 2A-A.

On January 26 at 12:39 a.m., the licensee rendered both the A and B trains of SI inoperable. The A train pump had been previously removed from service on January 25 to perform, in part, a routine oil change of the SI pump A motor bearings. The work was performed using instructions in Work Order (WO) 99-011252-000 and Preventive Maintenance (PM) Procedure 800832001, Revision 6. Step 1.4 of the PM directed the technicians to perform Maintenance Instruction 0-MI-EPM-317-103.0, "Lubrication of Westinghouse Motors," Revision 3 on SI pump 2A-A. The technicians, however, drained oil from the SI pump 2B-B motor. This action rendered pump 2B-B inoperable. The licensee discovered the problem, entered TS 3.0.3 beginning 12:39 a.m., and subsequently returned pump 2B-B to operable status at 3:16 a.m. by completing its required operability test. The licensee determined that SI pump 2B-B was not available (i.e., oil drained) for about 30 minutes and entered the issue into the licensee's corrective action program as PER 00-0637. The inspectors confirmed the facts associated with the event by a review of the operating logs, the associated work package, and through discussions with plant personnel.

The inspectors determined that the inoperability of both SI pumps for the time described represented low risk significance in that the capability of SI injection from other emergency core cooling system equipment was maintained and the out-of-service time frame was short. The finding was determined to be Green using Phase 2 of the SDP.

TS 6.8.1.a. requires, in part, that procedures shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, "Quality Assurance Program Requirements (Operations)." Contrary to the above, the licensee failed to follow WO 99-011252-000 and PM procedure 800832001, Revision 6, instructions when technicians drained oil from SI pump 2B-B versus SI pump 2A-A. This failure rendered both SI pumps inoperable simultaneously for about 2.5 hours. The NRC is treating this violation as an NCV, consistent with the Interim Enforcement Policy for pilot plants. The violation is identified as NCV 50-328/00-01-01, Failure to Follow Procedures Resulting in Simultaneous Inoperability of Both Unit 2 Safety Injection Pumps.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant mitigating systems, listed below, to assess, as appropriate, (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) where compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; (5) where continued operability was considered unjustified, the impact on TS LCOs and the risk significance in accordance with the SDP.

! Technical operability evaluation (TOE) 0-00-030-0593 was issued on 1/27/00 to evaluate a past operability concern when, on 1/24/00, a Unit 2 purge supply duct access door was found fallen inside a duct which forms part of the auxiliary building secondary containment enclosure boundary. Inspectors also reviewed a related PER 00-030-0593 and log entries.

! TOE 0-99-311-10290 was issued on 11/29/99 to evaluate a present A-train control building emergency ventilation system operability concern when, on 10/16/1999, testing of the B-train determined that it was unable to meet TS Surveillance

Requirement 4.7.7.e.3. and the train was inoperable due to excessive control building envelope leakage.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated, as appropriate, selected risk-significant operator workarounds, listed below, for potential affects on the functionality of mitigating systems. The workarounds were reviewed to determine (1) if the functional capability of the system or human reliability in responding to an initiating event was affected, (2) the effect on the operator's ability to implement abnormal or emergency procedures, and (3) if operator workaround problems were captured in the licensee's corrective action program.

! Operator Workaround (OWA) SQ99007WA concerned a potential isolation of the intermediate pressure feedwater heater strings following a turbine/reactor trip

! SQ96025WA concerned frequent venting of the charging pumps due to gas buildup

b. Observations and Findings

No findings were identified and documented through this inspection.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post maintenance test (PMT) procedures for the equipment below to ensure the equipment was returned to service satisfactorily. The inspectors evaluated the PMT to ensure it properly addressed the work performed.

! 120V AC vital inverter 2-IV static transfer switch

! AFW pump 2A-A breaker replacement

! Loop 4 channel 2 steam line pressure transmitter 2-PT-1-27B replacement

b. Observations and Findings

No findings were identified and documented through this inspection.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the outage risk control plan for the upcoming February 2000 Unit 1 Cycle 10 refueling outage to assess whether the licensee had appropriately considered risk, industry experience and previous site specific problems, and to confirm that the licensee had mitigation/response strategies for losses of key safety functions.

b. Observations and Findings

No findings were identified and documented through this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of selected risk-significant SSCs, listed below, to assess, as appropriate, whether the SSCs met TS, updated final safety analysis report (UFSAR), and licensee procedure requirements, and to determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

- 0-SI-SFT-30-149.B, Auxiliary Building Gas Treatment System

- 2-SI-SXP-003-201.B, Motor Driven Auxiliary Feed Water Pump 2B-B Performance Test

b. Observations and Findings

No findings were identified and documented through this inspection.

4 OTHER ACTIVITIES

4OA3 Event Followup

a. Inspection Scope

The inspectors responded to a reactor trip and SI on Unit 2 on January 18 initiated by a momentary interruption of power to vital 120V AC instrument bus 2-IV. The inspectors observed operator performance in the control room, performance of mitigating systems, and the licensee's event notification. The inspectors observed the licensee stabilize the unit to Mode 3. The inspectors reported risk factors surrounding the event to the Region II senior reactor analyst for determination of the event's conditional core damage probability.

b. Observations and Findings

Based on preliminary risk assessment and in accordance with Management Directive 8.3, "NRC Incident Investigation Procedures," a special inspection was initiated on January 19 to gather additional facts and to determine the risk significance of the event findings using the SDP. Observations and findings regarding the event follow-up are discussed in Special Inspection Report 50-328/00-03.

4OA4 Other

.1 (Closed) Licensee Event Report (LER) 50-327, 328/99003-00: Control Room Emergency Ventilation System Start as a Result of the Smell of Smoke in the Control Room. The inspectors reviewed the circumstances associated with the event to ensure that TS 3.7.7 requirements were satisfied for demonstration of operability.

.2 (Closed) Apparent Violation (AV) 50-327, 328/99-04-01, EA 99-199: Failure to Include the Storm Drain System within the Scope of the Maintenance Rule. The NRC issued a Notice of Violation (NOV) for this issue by letter dated January 26, 2000. The NOV was based on a failure to include the storm drain system within the Scope of the Maintenance Rule as of July 10, 1996. The system should have been scoped in the Maintenance Rule because its failure could have resulted in a reactor trip or actuation of a safety-related system. Although this issue was screened under the SDP as a Green finding, the violation was cited because the licensee was not in compliance at the time of the inspection. Based on the issuance of the NOV, this apparent violation is closed and a violation opened and identified as VIO 50-327, 328/00-01-02, Failure to Include the Storm Drain System within the Scope of the Maintenance Rule.

.3 (Closed) URI 50-327, 328/98-13-04: Evaluation of Ice Density Increase and Effects of Ice Voiding. During a review of ice basket servicing in the ice condenser (IC), the inspectors identified a concern associated with density changes in the ice mass from thermal drilling and its potential impact on design calculations and the IC ability to perform its design safety function. This item remained open pending NRC evaluation of the licensee's planned review of this issue.

The licensee completed their review of this issue. The results of that review were documented in a TVA report dated December 18, 1998. The staff reviewed the licensee's report along with Westinghouse Topical Report WCAP-8282 which described the results associated with testing conducted at the Waltz Mill Test Facility. That testing included the addition of water to simulate expected ice densification levels. The NRC review of this issue was completed on December 13, 1999, and is included as an attachment to this report. Based on this review the staff concluded that the effect of ice densification due to thermal drilling to correct the effects of voiding and coning will not have a significant effect on the thermal performance of the IC.

.4 (Closed) URI 50-327, 328/98-04-02: Potential Inadequate Sampling of Ice Condenser Ice Baskets and Ice Basket Weights Due to Frozen Baskets. This item involved whether the TS required "representative" sample could be obtained due to many frozen baskets which were unable to be weighed and was left open pending further NRC evaluation of this issue. The concern was that a substantial number of frozen ice baskets may accumulate thus eliminating a significant portion of the total ice inventory from the weighed sampling process.

A detailed review of ice weighing data was performed and documented in NRC inspection report 50-327, 328-98-13. During that review the inspectors noted that the sublimation rates experienced were well within the allowed margin. The TS required weight allows for a sublimation rate of 15 percent plus a 1 percent error margin to the safety analysis minimum allowable requirement of 922 pounds per basket. However, of the 144 samples picked to be weighed for the TS sample, 70 alternative baskets (allowed by the TS) had to be picked due to frozen baskets. Only by review of additional weight data were the inspectors able to gain a reasonable assurance of sufficient ice.

The licensee provided a response dated October 1, 1999, to an NRC request for additional information related to this issue. In that response the licensee had stated that any number of baskets could be stuck without affecting the validity of the statical method used in the sampling process. The NRC review of this issue was completed on December 13, 1999, and is included as an attachment to this report. The staff reviewed the licensee's response and determined that the response associated with validity of the statistical method was not acceptable. However, the staff also concluded that the licensee's current approach taken to supplement their statistical sampling was adequate to demonstrate sufficient total ice inventory. Although, correction of this weakness in the current form of the TS requirements is being pursued by the nuclear industry, similar supplementary measures will be required.

.5 (Closed) URI 50-327, 328/98-06-01: Potential Deficiencies in Maintenance and Inspection Procedures which Resulted in Ice Condenser Ice Basket Damage and did not Promptly Identify the Damage. This item involved a question as to the adequacy of maintenance procedures to have identified damaged ice baskets due to excessive force placed on the bottom of the ice basket.

The licensee subsequently adopted acceptance criteria for punctured and/or dented ice baskets. This criteria was developed by Westinghouse and documented under Task No. TVA-98-083, dated September 18, 1998. The inspectors performed subsequent reviews of the adequacy of licensee identification and repair/replacement of degraded IC ice baskets during the next two scheduled refueling outages. During the Unit 1 refueling outage, the licensee visually inspected the ice baskets and identified approximately 60 ice baskets (including 10 damaged ice baskets from the April 1998 forced outage) exhibiting various degrees of damage. The inspectors noted that out of the 60 damaged ice baskets, the licensee determined that 51 did not meet the acceptance criteria from Task No. TVA-98-083 and required repair, modification or replacement. In addition, out of the 69 damaged ice baskets identified during the Unit 2 refueling outage, 31 did not meet the acceptance criteria and required repair, modification or replacement. The licensee had evaluated the condition of the ice baskets and determined that the damaged ice baskets would not have had any impact on IC operability. These reviews were documented in NRC Inspection Reports 50-327, 328/98-13 and 50-327, 328/99-03. As the result of these reviews NCVs 50-327/98-13-03 and 50-328/99-03-05 were issued for failure to promptly identify and correct damaged ice baskets.

The NRC review of this issue was completed on December 13, 1999, and is included as an attachment to this report. The staff concluded that the physical damage to ice baskets is expected to be limited to a small percentage of the total ice basket population and that the licensee's evaluation regarding IC operability was reasonable.

4OA5 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on February 17, 2000. The licensee acknowledged the findings presented.

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

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Bajestani, Site Vice President
H. Butterworth, Operations Manager
E. Freeman, Maintenance and Modifications Manager
J. Gates, Site Support Manager
C. Kent, Radcon/Chemistry Manager
D. Koehl, Plant Manager
M. Lorek, Site Engineering Manager
B. O'Brien, Maintenance Manager
P. Salas, Manager of Licensing and Industry Affairs
J. Valente, Engineering & Support Services Manager

NRC

R. Bernhard, Region II Senior Reactor Analyst
W. Rogers, Region II Senior Reactor Analyst

ITEMS OPENED AND CLOSED

Opened

50-327, 328/00-01-02	VIO	Failure to Include the Storm Drain System within the Scope of the Maintenance Rule (Section 4OA4.2).

Opened and Closed

50-328/00-01-01	NCV	Failure to Follow Procedure Resulting in Simultaneous Inoperability of Both Unit 2 Safety Injection Pumps (Section 1R14.2).

Closed

50-327, 328/99003-00	LER	Control Room Emergency Ventilation System Start as a Result of the Smell of Smoke in the Control Room (Section 4OA4.1).
50-327, 328/99-04-01	AV	Failure to Include the Storm Drain System Within the Scope of the Maintenance Rule (Section 4OA4.2).
50-327, 328/98-13-04	URI	Evaluation of Ice Density Increase and Effects of Ice Voiding (Section 4OA4.3).
50-327, 328/98-04-02	URI	Potential Inadequate Sampling of Ice Condenser Ice Baskets and Ice Basket Weights Due to Frozen Baskets (Section 4OA4.4).
50-327, 328/98-06-01	URI	Potential Deficiencies in Maintenance and Inspection Procedures Which Resulted in Ice Condenser Ice Basket Damage and Did Not Promptly Identify the Damage (Section 4OA4.5).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 2O555-O1

December 13, 1999

MEMORANDUM TO: Loren R. Plisco, Director
Division of Reactor Projects
Region II

FROM: Suzanne C. Black, Deputy Director ***/RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: NRR RESPONSE TO TASK INTERFACE AGREEMENT (TIA) 99-02, SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2- ADEQUACY OF SEQUOYAH ICE CONDENSER ICE BED AND BASKETS (TAC NO. MA4999)

By memorandum dated February 25, 1999, Region II requested technical assistance regarding the adequacy of the Sequoyah ice condenser (IC) ice beds and baskets via TIA 99-02. The subject TIA describes the results of a Fall 1998 special inspection conducted by Region II regarding the Sequoyah Unit 1 IC system. Attached to the TIA were NRC Inspection Report No. 50-327,328/98-13, which documented Region II's inspection findings, and various documents supplied by the Tennessee Valley Authority (TVA), the licensee for Sequoyah. The inspection findings led to identification of three specific concerns with respect to the Sequoyah IC condition:

1. Does the densification of the ice bed by thermal drilling and the formation of cones and voids meet the IC design basis, so that the IC is still able to perform its safety function of lowering steam pressure from a loss-of-coolant accident?

2. Can the licensee assure sufficient ice weight at a 95 percent confidence level with approximately 45 percent of the baskets unable to be weighed? Have Technical Specification requirements been met? If so, what is the upper bound of unweighable baskets where this confidence level would not be met?

3. What is the upper bound of damaged ice baskets that can be sustained so that the ice bed can perform its intended safety function?

We have reviewed the documents supplied with TIA 99-02, as well as various other applicable documents relating to Sequoyah's licensing basis. We also made this issue a topic of discussion at a public meeting with the Ice Condenser Minigroup at NRC Headquarters on August 11, 1999. As a result of this meeting, we sent a request for information to TVA on September 10, 1999. TVA provided a response to our questions by letter dated October 1, 1999. We have now completed the review requested by the TIA. The attached assessment is the result of the subject review by the Office of Nuclear Reactor Regulation (NRR).

We understand that ICs are the subject of ongoing regulatory discussions and inspections, and may require additional input from NRR. However, we consider that this memorandum and its attachment complete our review and evaluation efforts associated with TIA 99-02 (TAC Number MA4999). We believe the attached information to be responsive to your request.

Attachment 1

Please contact Ronald Hernan, the NRR Project Manager for Sequoyah, at 301-415-2010, if you have any questions on the attached evaluation.

Docket Nos. 50-327 and 50-328

Attachment: NRR Assessment

cc w/attachment: M. Oprendek, Region I
G. E. Grant, Region III
K. E. Brockman, Region IV

RESPONSE TO TIA 99-02 ON ADEQUACY OF SEQUOYAH

ICE CONDENSER ICE BED AND BASKETS

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

Introduction

The subject Task Interface Agreement (TIA) describes the results of a Fall 1998 special inspection conducted by Region II of the U.S. Nuclear Regulatory Commission (NRC) regarding the Sequoyah Unit 1 Ice Condenser (IC) system. The inspection findings led to the identification of three specific questions with respect to the Sequoyah IC condition:

1. Does the densification of the ice bed by thermal drilling and the formation of cones and voids meet the IC design basis, so that the IC is still able to perform its safety function of lowering steam pressure from a loss-of-coolant accident?

2. Can the licensee assure sufficient ice weight at a 95 percent confidence level with approximately 45 percent of the baskets unable to be weighed? Have Technical Specification (TS) requirements been met? If so, what is the upper bound of unweighable baskets where this confidence level will not be met?

3. What is the upper bound of damaged ice baskets that can be sustained so that the ice bed can perform its intended safety function?

The following discussion describes the review findings by the Office of Nuclear Reactor Regulation (NRR) with respect to the above questions.

Formation of Ice Voids and Thermal Drilling Effects on IC Safetv Function (Question # 1)

It is the nature of the ice in ICs to change from solid state to gaseous state (sublimate) during the course of normal operation of the reactor plant. The sublimation rate is highly dependent upon the location of the ice (axially and radially) within the IC. As part of the special inspection in Fall 1998, the NRC inspectors noted that sublimation had caused many ice baskets to lose ice mass in the course of a fuel cycle and a significant number of baskets were not full, as evidenced by voids and "coning." In one case, ice appeared to be completely missing in a lower 6- to 8-foot section. In addition to localized missing ice, it has been observed that sublimation rates across the ice bed may be uneven. As noted in documentation provided by the NRC inspectors and by the Tennessee Valley Authority (TVA, the licensee), thermal drilling is part of a process utilized at Sequoyah to replace the ice that has sublimated and the thermal drilling does lead to some localized ice melting, followed by re-freezing. This practice can result in (a) ice baskets freezing in place and in (b) non-uniformity in ice mass distribution.

ATTACHMENT

Non-uniformity in ice mass distribution across the ice bed, in turn, may create a potential for uneven melt progression in the event of a steam release from a design basis accident. This in turn may lead to a premature melt-through of a portion of the IC, thus creating a steam bypass path and compromising the thermal performance of the IC. Problems associated with frozen ice baskets are discussed in more detail below.

In response to the NRC inspector observations, TVA conducted a review of the "voiding and coning" issue and documented their conclusions in an internal TVA report dated December 18, 1998, which was included with the Region II TIA request. In the report, the licensee notes that, upon replenishment with flake ice, normal fusion and compaction takes place such that there is no apparent difference between the old and new ice. The effect of this ice densification on the thermal performance of the ice condenser has been addressed by previous analyses that have been reviewed and approved by the NRC staff. Specifically, tests were conducted at the Westinghouse Electric Company Waltz Mill Test Facility that included the addition of water to simulate all expected ice densification levels. These tests and test results are described in Westinghouse Topical Report WCAP-8282. The tests also included flake ice experiments where a portion of the baskets were loaded with block ice. The results showed only minor differences in the thermal performance of the IC.

The December 18,1998, TVA report noted that ice baskets are routinely inspected visually for ice voids and other characteristics of low-weight baskets. Inspections of the ice bed are performed to identify baskets with significant loss of ice, including direct visual inspections of lower portions of ice baskets from inside the lower ice condenser. Video cameras are also used to view the entire length of selected baskets. Baskets with significant loss of ice are designated for ice addition in the ice bed servicing plan. NRR has reviewed the TVA analysis and concludes that, although no bounding analysis of ice mass nonuniformity acceptance criteria was performed, the visual inspections and servicing plan actions should be adequate to assure IC operability.

On the basis of the above discussions, the NRR staff believes that the observed effect of ice densification due to thermal drilling to correct the effects of voiding and coning will not have a significant effect on the thermal performance of the IC.

<u>Statistical Adequacy of Ice Mass Inventory Verification (Question #2)</u>

TS 4.6.5.1.d.2 prescribes that a representative sample of at least 144 ice baskets (of a total of 1944 baskets) is to be selected for determining the total weight of ice in the IC. Each basket selected is weighed to verify that it contains at least 1071 pounds of ice. In selecting the 144 sample baskets, the TS requires selecting six rows per bay, one each from radial rows 1, 2, 4, 6, 8, and 9 in each of the 24 bays.

Statistical sampling of ice baskets for weighing was established as a means of determining the status of the total ice inventory. The sample size of 144 baskets was determined to be sufficient to give 95% confidence level that the estimated inventory is representative of the actual amount of ice present. Since as-found baskets may be frozen in place, and thus may not be weighable, alternate ice basket sampling is permitted by the TS. However, the same specifications do not place any limit on the number of alternate baskets that may be sampled. Hence, a substantial amount of frozen ice baskets may accumulate (approximately 45% in the recent inspection findings for Sequoyah), thus eliminating a significant portion of the total ice inventory from the sampling process.

In the October 1, 1999, response to an NRC request for additional information dated September 10, 1999, the licensee stated that any number of baskets can be stuck without affecting the validity of the statistical method. The licensee's bases for the above statement are that stuck baskets are rejected solely because they cannot be lifted and that the TS requires that the alternate be obtained from the same row and adjacent bay as the original sample basket. The staff has reviewed the licensee's response, and finds it not acceptable. When an unlimited number of original sample baskets are not weighable and are excluded from the statistical sampling process, the staff believes that the new sample is no longer a representative one and that it does not provide 95% confidence level for the total of 1944 baskets. The lack of a limit in using alternatives reflects a weakness in the current form of the TS requirements, rather than a TS compliance Issue.

To address this weakness, the nuclear industry is working on a proposed set of revised TSs. The staff expects that the industry group will pursue development of improved TS to address this weakness on an expedited basis.

Regarding the safety aspects of this issue, the NRR staff believes that the current approach that the licensee has taken to supplement their statistical sampling is adequate to demonstrate that they have sufficient total ice inventory. Specifically, the licensee's supplementary measures included weighing a large number of unfrozen baskets (1200 and 1306 for Unit 1 Cycle 9 and Unit 2 Cycle 9 respectively), replenishing ice, visually inspecting those that are stuck, and weighing those frozen baskets that were freed. Based on these supplementary measures and engineering judgment, as reported in Inspection Report 98-13, the staff believes that these measures should provide sufficient assurance that the ice condenser contains adequate ice inventory for the current cycle of operation. Further cycles would require similar supplementary measures, in the absence of revised TS.

In summary, the NRR staff believes that allowing an unlimited number of alternate ice baskets is a weakness in the current TS and that the current practice of supplementing the TS-required sampling, as discussed above, constitutes an adequate interim approach to ice inventory determination until acceptable, revised, TS are developed.

Ice Basket Damage Implications (Question #3)

This review did not examine any analyses that may exist with respect to establishing an upper bound on the number of baskets that may be physically damaged without loss of IC operability. However, the following discussion may provide some insight relative to the importance of observed basket damage.

One effect of damaged baskets on IC operability is the potential for steam flow blockage. Another effect is basket ejection into upper containment. In this case, the consequences are ice mass loss through basket ejection, as well as the potential for missile damage to safety-related systems in the upper containment (e.g., hydrogen igniters). The latter, however, is not an IC operability concern.

Ice-basket damage of the type typically found in recent ice-condenser inspections includes dents, torn ligaments and failed or missing basket hold-down assemblies. If the damage is sufficient to cause major basket deformation or relocation, the most direct consequence could be flow blockage. However, the potential for affecting IC operability is believed to be low. In the early phase of a blowdown (i.e., the first few seconds) there would be a substantial inventory of ice within the baskets, which would provide significant resistance to basket reconfiguration. Later into the blowdown, when an appreciable mass of ice is melted, if there were any basket deformation, it would be self-compensating in the sense that localized basket displacement would produce flow restrictions as well as increased flow areas.

Damaged baskets may also be considered to lead to ejection due to seismic and hydrodynamic forces associated with a DBA. The resulting loss of ice mass could lead to degradation of IC operability. Westinghouse has indicated (DAP-90-633, November 11, 1999, letter to Duke Power in reference to McGuire and Catawba Nuclear Stations) that the potential for ice-basket ejection is limited to about 0.3% of the total ice basket population. This conclusion was made on the basis of a statistical determination of failed basket hold-down assemblies and the number of baskets (about 30%) having the appropriate location such that their missile trajectory would permit entry into upper containment without hitting existing structural barriers. Hence, the number of baskets that could be postulated to be ejected is limited to something less than six. To determine the effect of this on ice-condenser operability, it is necessary to compare the ice mass in the six baskets with the margin built into the total ice mass inventory. The staff has reviewed the above reference and finds the qualitative argument regarding hold-down assembly failure statistics and potential basket ejection trajectories to be reasonable.

In summary, basket physical damage is expected to be limited to a small percentage of the total basket population. Furthermore, basket damage would affect IC operability mainly through the mechanism of loss of ice mass. Hence, an available ice mass margin is an important figure of merit when considering basket damage. The program established by the licensee, as described in Attachment 3 to the subject TIA, should assure sufficient ice mass margin to offset any degradation of IC performance resulting from damaged ice baskets.

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.