U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Facility:	Sequoyah Nuclear Plant, Units 1 & 2
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Dates:	August 30 through October 10, 1998
Inspectors:	M. Shannon, Senior Resident Inspector R. Starkey, Resident Inspector R. Telson, Resident Inspector R. Carrion, Project Engineer (Section M4.1) E. Girard, Reactor Inspector (Section E8.1) J. Blake, Reactor Inspector (Sections M1.2, M1.3) D. Jones, Senior Radiation Specialist (Sections R1.2, R8.1, 8.2)
Approved by:	Harold O. Christensen, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure 2

EXECUTIVE SUMMARY

Sequoyah Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-327/98-09, 50-328/98-09

This integrated inspection included aspects of licensee operations, maintenance, engineering, plant support, and effectiveness of licensee controls in identifying, resolving, and preventing problems.

Operations

- Operational performance during the Unit 1 plant shutdown, outage, reduced inventory operation and startup was good and contributed to a successful 29 day refueling outage. (Sections O1.2 and O1.3)

- A non-conservative operator action was observed when a heater drain pump was operated in excess of the maximum allowed running current for 40 minutes. (Section O1.4)

- Operators overloaded the 1A-A EDG for less than 30 seconds when they failed to use all available instrumentation. The engineering evaluation of the EDG overload event concluded that the short duration overload was insignificant. (Section O1.5)

Maintenance

- Maintenance personnel contributed to the overall success of the Unit 1Cycle 9 refueling outage by implementing several significant plant modifications and completing a well planned outage schedule. (Section M1.1)

- Corrective actions following the failure of a Westinghouse Type DS-532 480 Vac circuit breaker have brought significant improvements to the Type DS Breaker maintenance program. (Section M2.1)

- A violation was identified with two examples of failure to meet 10 CFR 50, Appendix B, Criterion XI, Test Control, when Type DS-532 480 Vac circuit breakers failed to perform satisfactorily in service. (Section M2.1)

- The licensee's implementation of Steam Generator and Inservice Inspection programs were thorough and done in a very professional manner. The steam generator eddy current testing program expanded the state-of-the-art by implementing a depth-sizing procedure for Primary Water Stress Corrosion Cracking. (Section M1.3)

Engineering

- The licensee determined that close monitoring of unidentified leakage was an acceptable compensatory measure to startup and operate Unit 1 for one operating cycle with dried boron on the exterior of the pressurizer. (Section E2.1)

- A concern was identified for the potential failure to properly categorize a maintenance preventable functional failure. (Section E2.2)

Plant Support

- Radiological controls during the U1C9 outage were effective as evidenced by a record low total dose for Unit 1 of 200 rem. (Section R1.1)

- Personnel entering the Radiologically Controlled Area were adequately briefed on radiological hazards and protective measures. (Section R1.2)

- Maximum individual radiation exposures were controlled to levels which were well within the licensee's administrative limit and the regulatory limits for occupational dose specified in 10 CFR 20.1201(a). (Section R1.2)

- The licensee was generally successful in meeting established ALARA goals (Section R1.2)

- The licensee had implemented an effective shutdown chemistry control plan and closely monitored primary coolant chemistry during the shutdown for the Unit 1 refueling outage. (Section R1.2)

- The licensee was properly monitoring and controlling personnel radiation exposure during the Unit 1 Refueling Outage and posting area radiological conditions in accordance with 10 CFR Part 20. (Section R1.2)

Report Details

Summary of Plant Status

Unit 1 began the inspection period at approximately 88% power in coast down for a refueling outage. On September 9, 1998, the unit was taken off line for refueling outage cycle nine. The outage ended 29 days later on October 8, 1998 when the main generator was synchronized to the grid. When the inspection period ended, the unit was at approximately 75% power and proceeding to full power.

Unit 2 began the inspection period in Mode 3 and was starting up following an automatic reactor trip from 100% power on August 27, 1998. The unit reached 100% power on September 1, 1998, and continued to operate at 100% power throughout the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was considered to be good.

O1.2 Plant Outage and Startup Observation, Unit 1

a. Inspection Scope (71707)

The inspectors observed various outage and startup activities during the Unit 1 refueling outage.

b. Observations and Findings

During the inspection period, the inspectors observed portions of Unit 1 shutdown, initiation of RHR cooling, core off-load, core reload, reduced inventory operations, ice condenser maintenance activities, containment spay heat exchanger replacement, plant startup and synchronization to the grid. The inspectors noted that evolutions were well controlled with effective communications. In addition, the inspectors observed a significant level of senior management oversight present in the control room during sensitive plant activities.

c. Conclusions

Operational performance during the Unit 1 plant shutdown, outage, and startup was good and contributed to a successful 29 day refueling outage.

O1.3 Reduced Inventory Operations

a. Inspection Scope (71707)

Inspectors reviewed licensee activities associated with reactor coolant system (RCS) midloop/reduced inventory activities during the Unit 1 Cycle 9 refueling outage.

b. Observations and Findings

The inspectors reviewed the licensee's preparations for operation in reduced inventory and mid-loop conditions. Midloop operations were required in order to remove steam generator nozzle dams installed earlier in the outage with fuel removed from the vessel. The inspection included a review of the licensee's response to GL 88-17, Loss of Decay Heat Removal, along with implemented actions based on their response.

The licensee's assessment of risk provided effective controls and considered the impact of evolutions which affected shutdown risk in the areas of reactivity, core cooling, RCS inventory, power availability, and containment integrity.

c. Conclusions

The licensee implemented effective controls to minimize risk during Unit 1 reduced inventory operations.

O1.4 Overload of Unit 1 No. 3 Heater Drain Pump Motor

a. Inspection Scope (71707)

The inspectors observed operators' response when maximum allowed running current for a No. 3 Heater Drain Pump Motor was exceeded.

b. Observations and Findings

As power was being reduced during the shutdown of Unit 1, various secondary pumps were being stopped when the unit operator noted that a No. 3 Heater Drain Pump (HOT) motor current was exceeding the maximum allowed running current. The unit operator promptly brought this condition to the attention of the unit supervisor. However, the ongoing power reduction was continued and the No. 3 heater drain pump motor was left in an overloaded condition.

Approximately 40 minutes later the stator winding temperature reached the alarm set point and the operators took appropriate corrective action. Following the high temperature alarm, the operators discovered an error in the recently revised operating procedure which resulted in an inappropriate valve lineup. The configuration was corrected and motor amps returned to normal.

c. Conclusions

A non-conservative operator action occurred when a heater drain pump was operated in excess of the maximum allowed running current for 40 minutes.

01.5 Overload of EDG 1A-A

a. Inspection Scope (71707)

The inspectors reviewed the 1A-A EDG overload event and the licensee's follow-up evaluation of the event.

b. Observations and Findings

On September 10, 1998, during performance of 1-SI-OPS-082-026.A, Loss of Offsite Power With Safety Injection-D/G 1A-A Test, operators overloaded the 1A-A EDG to approximately 5.2 MW, 5.6 MVA. The EDG engine is rated for continuous operation at 4.4 MW with a short time (two hours per 24 hours) rating of 4.84 MW, and a maximum rating (30 minutes per year) of 5.073 MW. The EDG generator has a continuous rating of 5.0 MVA and a short time (two hours per 24 hours) rating of 5.5 MVA). The event occurred while EDG 1A-A was being loaded to 4.0 MW in preparation for a two hour loaded run. When the operator increased the load on the EDG, the control room MW indicator became stuck at approximately 3.5 MW. The operator continued to increase the EDG load setting as he attempted to obtain 4.0 MW. When the operators (RO and SRO) failed to get the expected MW increase as indicated on the MW meter, they observed that the temporarily installed instrumentation, which was positioned on the floor in front of the control room EDG panel, indicated approximately 5.1 MW and 2.050 MVAR. The operator immediately reduced load to 4.0 MW. This event was documented in PER No. SQ981231OPER.

Personnel involved in the EDG 1A-A loading evolution estimated that the overload condition lasted less than 30 seconds. During the event, personnel in the EDG engine room observed that the engine fuel racks went to the full fuel position. A data acquisition recorder was in use at the EDG room to monitor EDG control parameters (KW, KVA, AMPS, RPM, Volts), but it was in the display-only mode and had not been setup to store EDG control parameters. Also, the plant integrated computer (ICS) was not setup to monitor EDG control parameters.

After the EDG load was established at 4.0 MW, operators conferred with technical support engineers to determine what actions, if any, should be taken regarding the overload condition. Engineering informed operations that both the engine and generator had remained within their ratings and that there were no impacts on operability. Based on the assessment by engineering, operators continued with the two-hour loaded run at 4.0 MW with no irregularities noted in generator or engine performance during the remainder of the run. The following day, September 11, 1998, the EDG vendor representative responded in writing to the licensee that he concurred with the licensee's written evaluation, dated September 10, and that the incident did not have any impact on the operability of the EDG.

The inspector, after observing the overload event from the control room, was concerned that operators may have proceeded with the two hour loaded run of the 1A-A EDG without the benefit of a thorough engineering evaluation of the overload condition. The September 11 written engineering evaluation stated that the calculated loading on the generator was approximately 5.496 MVA which was below the two hour rating of 5.5 MVA, and was within the 2 hour rating at all times.

When the inspectors question the evaluation, engineering initiated revision 1 of the September 11 evaluation which was completed on September 29, 1998. Revision 1 stated that the calculated loading on the generator was approximately 5.608 MVA which slightly exceeded the generator two hour rating of 5.5 MVA. Revision 1 concluded that the short duration in which the generator load was above the two hour rating was insignificant in that no heating damage would occur in the generator windings. The licensee also determined that no engine inspections or maintenance activities were required from the incident. The EDG vendor representative concurred with revision 1 of the licensee's evaluation. The inspectors determined that the licensee's revised evaluation was reasonable and that operability of EDG 1A-A did not appear to be affected by the short duration overload condition.

c. Conclusions

Operators overloaded the 1A-A EDG for less than 30 seconds when they failed to use all available instrumentation. The engineering evaluation of the 1A-A EDG overload event concluded that the short duration overload was insignificant.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726 & 62707)

Using inspection procedures 61726 and 62707, the inspectors conducted frequent reviews of ongoing maintenance and surveillance activities. The inspectors observed and/or reviewed all or portions of the following work activities and/or surveillances:

- 0-SI-NUC-000-001.0 — Estimated Critical Conditions
- 1-SI-OPS-082-026.A — Loss of Offsite Power with Safety Injection-DG 1A-A Test
- 0-PI-ICC-046-057.0 — Calibration of Auxiliary Feedwater Pump A-S Loop F-46-57/S-46-57
- 2-SI-SXP-003-201.A — Motor Driven Auxiliary Feedwater Pump 2A-A Performance Test

- 1-SI-IFT-092-N42.2 — Functional Test of Power Range Nuclear Instrumentation System Channel N42
- WO NO: 98-007516 — Visual inspection of breaker frame alignment and pole unit integrity; visual inspection of contacts; contact compression; obtaining as-found trip force for Breaker SQN-1-BCTB-201-DE/4B
- WO NO: 98-007541 — Refurbishment of Breaker SQN-1-BCTB-201-DM/5B
- WO NO: 98-009960 — Balance Unit 2 CRDMG sets

In addition to the above activities, the inspectors reviewed the scope and the implementation of the licensee's Unit 1 cycle 9 refueling outage plan.

b. Observations and Findings

The inspectors reviewed the scope of the U1C9 refueling outage and observed several outage related maintenance activities. The inspectors noted that the licensee successfully completed the planned 28 day outage in 29 days, which set a record for the licensee and the industry for ice condenser plants. A total of 49 major maintenance activities were completed which included: replacement of the 1B containment spray heat exchanger, modifications, servicing and repairs to the ice condenser, replacement of the 1A-A motor driven auxiliary feedwater pump, installation of a bypass line from the number 2 feedwater heater to the main condenser and replacement of the 1B-B containment spray pump.

The above maintenance activities were completed in accordance with procedures and performed by knowledgeable craft.

c. Conclusions

Maintenance personnel contributed to the overall success of the U1C9 refueling outage by implementing several significant plant modifications and completing a well planned outage schedule.

M1.2 Steam Generator (SG) Eddy Current Testing (ET)

a. Inspection Scope (IP 50002)

The inspector reviewed the eddy current inspections of the Unit 1 SGs.

b. Observations and Findings

The licensee's SG degradation management program was previously inspected and discussed in Section M3.1 of the NRC Integrated Inspection Report Nos. 50-327,328/98-06. Subsequent to that inspection, the licensee completed qualification of a Plus-Point Eddy Current technique for measurement of the depth of primary water

stress corrosion cracking (PWSCC) in the vicinity of tube support plates. A qualified, reliable method for measurement of the depth of PWSCC cracking allowed the licensee to use the Technical Specification plugging limit of 40% through-wall, in lieu of plugging, or repair, on detection of the cracking. (Qualification of this procedure had been discussed with, and reviewed by, NRC's Office of Nuclear Reactor Regulation.) The use of this procedure allowed the licensee to leave 63 tubes in service that would have otherwise been plugged.

The inspectors reviewed the application of the newly qualified Plus-Point ET procedure as it was used to measure the depth of PWSCC. The inspectors' review included selected eddy current data from tubes which were reported to have PWSCC with measured depths between 30 and 40 percent through wall. The results of the inspectors' data review were in agreement with the licensee's results.

M1.3 Inservice Inspection (ISI)

a. Inspection Scope (IP 73753)

The inspector reviewed the results of the licensee's ISI of containment, components, piping and supports.

b. Observations and Findings

The Fall 1998 Unit 1 refueling outage was the final outage of the first period of the second 10-year ISI interval for piping and components. The inspectors reviewed the status of completion of the licensee's ISI programs to determine if the required percentage of inspections had been completed for the first period.

During the review of the inspection data, the inspectors noted that the licensee had found a number of loose bolted connections on the supports for the SGs and main coolant pumps. The inspectors noted that the resulting documentation did not include an attempt at a root cause evaluation for the loose connections. The licensee's approach was to expand the sample to 100% of the bolted major equipment supports, repair all loose connections, and schedule a 100% re-inspection of the connections during the next refueling outage. The inspector agreed with the licensee's disposition of the problem, noting that attempts at root cause evaluations would not be warranted unless some connections were found to be loose during the next inspection.

c. Conclusions SG ET and ISI

The licensee's implementation of Steam Generator and Inservice Inspection programs were thorough and done in a very professional manner. The steam generator eddy current testing program expanded the state-of-the-art by implementing a depth-sizing procedure for Primary Water Stress Corrosion Cracking.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Inadequate Testing of 480 Vac Westinghouse Type DS 532 Circuit Breakers

a. Inspection Scope (62707/40500/37551)

The inspectors reviewed preventive and corrective maintenance work documents and problem evaluation reports (PERs) completed by the licensee both prior to and subsequent to the May 19, 1998, failure of the shutdown board 1A1-A alternate supply breaker to properly close on demand. In addition, the inspectors reviewed the September 11, 1998, failure of the shutdown board 1A2-A alternate supply breaker to perform satisfactorily following maintenance.

b. Observations and Findings

In order to provide background information for the DS-532 breaker issues, findings from inspection reports 98-06, 98-07, and 98-08 are summarized below:

- On January 14, 1997, Shutdown Board 1A1-A alternate supply breaker 1-BCTB-201-DJ/5B-A failed to close on demand, causing the shutdown board to be temporarily de-energized. The licensee inspected and refurbished the breaker but was unable to determine the cause of the failure.

- On May 19, 1998, the same breaker failed to fully close on demand. The shutdown board was again de-energized and a vital static inverter powered by the shutdown board failed causing a reactor trip. This was the first closure attempt since a February 27, 1998, breaker refurbishment.

- The licensee's inspection of the breaker following the May 19th failure revealed badly burned C phase main contacts with a small burned area on the B-phase main contact and some evidence of burning on the A-phase arcing contact. A Westinghouse representative inspected the breaker and identified a gap on the C phase contacts that could not be properly adjusted even with allowances for missing contact material on the C phase contacts. A and B phase arcing contacts were touching when C phase had a gap and less-than-adequate compression was observed on all contacts. It was also noted that the main crank arm was not rotating into the fully closed position. Cracking of one of the pole insulators and a bow in the breaker structural angle were also observed.

- The licensee's investigation and corrective action plan, captured in detail in PER No. SQ980595PER included: 1) a root cause determination: "Inadequate detail provided in the vendor manual, which subsequently lead to a breaker maintenance procedure with inadequate detail for setting contact compression" (completed), 2) an extent of condition evaluation to include inspection and/or performance of 10 year maintenance on all of the site's 27 Type DS-532 breakers), 16 of which provided normal or alternate power feed to the site's 480 Vac Shutdown boards, 3) thermal imaging of the in-service shutdown board supply breakers to ensure that the main contacts were not overheating, 4) an

evaluation of previous similar events, 5) consultation with the vendor, 6) procedure upgrades, 7) establishment of a priority schedule for inspection and refurbishment of all Type DS-532 breakers, 8) evaluation of the site training program, and 9) submitting a posting on the Nuclear Network to inquire as to what other plants in the industry do for post maintenance testing. The corrective action due dates were documented in the PER.

- Inspectors identified additional contributors to the May 19, 1998, breaker failure and the observed degradation of Type DS-532 breakers to include: 1) lack of specification in the maintenance and testing guidance, 2) the decision to waive post maintenance testing (PMT) in the case of the failed breaker, 3) an ineffective trending program to identify degrading breaker conditions, 4) maintenance and testing procedures which permitted omission of and out-of-sequence performance of maintenance and testing steps and 5) weak maintenance and testing documentation which made accurate assessment of as-performed work sequencing difficult to ascertain.

- On July 22, 1998, Westinghouse released technical bulletin ESBU-TB-98-02 in response to the Type DS-532 breaker adjustment problems and recommended revisions to maintenance procedures.

- The licensee's DS-532 breaker refurbishment program revealed extensive material deficiencies which included degraded closing springs and operating mechanisms, incomplete crank arm travel, incorrect main/arcing contact synchronization and compression, excessive bowing of breaker frames, cracked pole insulators, excessive trip forces, and dry hardened grease in operating mechanisms.

The inspectors concluded that the failure to perform all testing required to demonstrate that the shutdown board 1A1-A alternate supply breaker would perform satisfactorily in service resulted in the failure of the breaker on May 19, 1998. This is identified as a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control (VIO 50-327/98-09-01). This issue was previously identified as an Unresolved Item (URI 50-327/98-06-02) for the apparent improper compression setting and potential inadequate post maintenance testing.

During the current inspection period, inspectors observed a second instance in which a breaker failed to perform satisfactorily immediately following maintenance:

- On September 7, 1998, the licensee completed refurbishment of breaker SQN-1-BCTB-201-DM/5B-B, which had been removed from 480V shutdown board 1B2-B. The opening and closing springs, trip shaft, and the levering-in mechanism were replaced. The operating mechanism was replaced twice, the first replacement operating mechanism having been found defective. According to work documents, the technician had difficulty setting contact compression and achieving the required crank arm travel both before and after replacement of the operating mechanism. Main and arcing contacts were adjusted, having initially been found out of synchronization and underparallel. Work had been accomplished using Revision 33 of MI-10.5, Westinghouse Type DS Breaker

and Switchgear Maintenance. All required verifications and related testing were completed at this time.

- On September 11, while attempting to install the rebuilt breaker in the shutdown board 1A2-A alternate breaker position, the new levering-in mechanism was discovered to be defective. The breaker was returned to the maintenance shop and the levering-in mechanism was replaced. Replacing the levering-in mechanism required the operating mechanism to be removed and reinstalled.

Following replacement of the levering-in mechanism, the breaker was returned to shutdown board 1A2-A. The breaker was closed but the closing spring charging motor failed to actuate. The breaker remained in service for two hours, then was taken out of service and tested. The charging motor malfunction was duplicated in one of five test closing cycles.

The breaker was returned to the electrical maintenance shop where the technician determined that the main contacts were misaligned in that contact compression was set too high. As a result, the breaker had failed to fully close and the crank arm had not rotated far enough to actuate the spring charging motor. PER Nos. SQ981220PER and SQ981283PER, identified as trending PERs, address the event as well as the physical condition of the breaker.

- Inspectors discussed the main contact misalignment with electrical shop personnel. The licensee confirmed that the electrical technician had omitted applicable steps of the maintenance procedure, MI-10.5, which would have verified proper main contact alignment and operation. The omission was contrary to MI-10.5 Step 3.5, Precautions and Limitations, which required general foreman, electrical engineer, or the maintenance shift supervisor approval to omit any applicable steps. The inspectors determined that the omitted verification would have identified the misaligned main contacts.

- Subsequent to the September 11, 1998, event, the licensee instituted additional maintenance verification measures requiring performance of a three page checklist prior to installing refurbished Type DS-532 circuit breakers.

The inspectors concluded that the failure to perform all testing required to demonstrate that the shutdown board 1A2-A alternate supply breaker would perform satisfactorily in service resulted in the failure of the breaker on September 11,1998. This is identified as the second example of violation VIO 50-327/98-09-01.

c. Conclusions

Corrective actions, following failures of Westinghouse Type DS-532 480 Vac circuit breakers, have brought significant improvements to the licensee's Type DS Breaker maintenance program.

A violation was identified with two examples of failure to meet 10 CFR 50, Appendix B, Criterion XI, Test Control, when Type DS-532 480 Vac circuit breakers failed to perform satisfactorily in service.

M4 Maintenance Staff Knowledge and Performance

M4.1 Balancing Unit 2 Control Rod Drive Motor-Generator Sets

a. Inspection Scope (61726)

The inspectors observed a corrective maintenance activity involving the balancing of the Unit 2 Control Rod Drive Motor Generator (CRDMG) Sets.

b. Observations and Findings

The licensee identified that the A phase directional over current relay for the 2B CRDMG set was approaching the trip position due to its unbalanced state and also due to relatively high equipment vibration. The work to correct the unbalanced condition was performed per Work Order 98-009960. The system engineer thoroughly briefed the work activities with the technicians prior to starting the evolution, including the equipment to be used and contingencies. The control room was notified and a comprehensive control room pre-job work briefing was held to discuss cautions, the work, and its potential effect on the operating unit in a "worst case" scenario. The inspectors observed that the actual work was done in accordance with the procedures listed in the work package and was completed without incident.

c. Conclusions

The inspectors concluded that the licensee personnel were knowledgeable of the requirements for performance of the maintenance evolution and the work activities were well performed.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) URI 50-327/98-06-02, Potential Improper Corrective Maintenance Activities Related to Improper Breaker Contact Compression Setting and Inadequate Post Maintenance Testing: This issue and its resolution were discussed in Section M2.1 of this report.

M8.2 (Closed) LER 50-327/98-01-00, Failure of the 480-Volt Shutdown Board 1A1-A Alternate Feeder Breaker Results in a Unit 1 Reactor Trip on a Low Steam Generator Water Level. This event was discussed in Section M2.1 of this report. No new issues were revealed by the LER.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Boric Acid Contamination on Exterior of Unit 1 Pressurizer

a. Inspection Scope (37551)

The inspectors reviewed the licensee's evaluation and corrective actions related to the exterior contamination of the Unit 1 pressurizer with borated reactor coolant.

b. Observations and Findings

On September 10, 1998, during the Unit 1 Cycle 9 (U1C9) refueling outage, an estimated 240 to 480 gallons of reactor coolant, at 2,000 ppm boron concentration, leaked through the pressurizer upper manway wetting down the outside of the pressurizer, its insulation, and components beneath the pressurizer in the containment. The manway leak was the result of a leaking flexatalic gasket. The leakage allowed boric acid residue to deposit on the outside of the carbon steel pressurizer vessel.

The licensee prepared a risk evaluation, with Westinghouse concurrence, for operation of Unit 1 for one cycle with boric acid residue left on the pressurizer exterior. The evaluation concluded that, as long as conditions remain dry, severe corrosion and material wastage would not occur on the pressurizer and continued operation of the pressurizer could be justified. As noted in the evaluation, boric acid corrosion (or wastage) of carbon steel components is a well-documented phenomenon addressed in IE Notice 86-108, Generic Letter 88-05, a Westinghouse letter to TVA, TVA-87-819, and TVA Safety Evaluation Report for Generic Letter 88-05 for SQN Units 1 and 2.

The inspectors reviewed the corrective actions as stated in PER No. SQ980116PER, revision 2, the licensee's risk evaluation, and the Westinghouse assessment and noted that none of the documents specified how the dryness of the boron would be assured until the next refueling outage. The inspectors then discussed with the plant manager and the site engineering manager whether compensatory measures were in place to identify conditions which could result in wetting of the boron on the exterior of the pressurizer. The inspectors were informed that site management would maintain a heightened awareness for any increase in unidentified leakage which could potentially result in wetting the boron on the pressurizer. Chemistry analysis of the containment atmosphere would be used to help in identifying the source of unidentified leakage and plant management would then develop an action plan based on location of the leak. The inspectors will conduct additional reviews of the licensee's boron corrosion control program to determine the adequacy. This is identified as an Inspection Followup Item (IFI 50-327/98-09-02).

c. Conclusions

The licensee determined that close monitoring of unidentified leakage was an acceptable compensatory measure to startup and operate Unit 1 for one operating cycle with dried boron on the exterior of the pressurizer. The inspectors continued to evaluate the licensee's boron corrosion control program.

E2.2 Breaker Failure Not Categorized As a Maintenance Preventable Functional Failure

a. Inspection Scope (37551)

The inspectors reviewed the licensee's maintenance preventable functional failure determinations for the breaker failure on May 19, 1998, which resulted in a Unit 1 reactor trip.

b. Observations and Findings

On May 19, 1998, the 480 Vac Shutdown Board 1A1-A was transferred from its normal supply to its alternate supply (see Section M2.1). The alternate feeder breaker had recently been overhauled. The "C" phase of the alternate supply breaker main line contacts failed to properly close which resulted in arcing and a sporadic single phase condition on the 480 Vac shutdown bus. This resulted in a failure of the 1-I Vital Inverter and a subsequent trip of Unit 1. The shutdown board was transferred back to the normal supply breaker and the alternate supply breaker was removed from service. Upon inspection, the licensee identified that the "C" phase of the alternate feeder breaker had damaged/burned main line contacts. The licensee noted that "An inspection of the failed breaker, by a Westinghouse representative, identified a gap on the "C" phase contacts that could not be properly adjusted even with allowances for missing contact material on the main "C" contacts.

In August 1998, the inspectors noted that the licensee had not identified the failed breaker event as a maintenance preventable functional failure. The licensee concluded that the breaker failure was not a preventable functional failure per 10 CFR 50.65. This conclusion was based on an evaluation that stated "there was no sustained loss of voltage (10 seconds), no condition that would have prevented the board from performing its intended function, and this is not considered a failure against system 201A." The inspectors noted that the sporadic single phase condition on the bus had caused the failure of the vital inverter and noted that if the shutdown board loads were lost due to the failure, the board could no longer perform its intended function and the event should have been considered to be a functional failure. In addition, the inspectors obtained data from the control room alarm printer that confirmed that the 480 Vac shutdown board had been de-energized for 18 seconds during the event which met the licensee's definition of sustained loss of voltage to the bus and should have been considered to be a functional failure.

In September 1998, the inspectors noted that the licensee had revised their evaluation; however, the failed breaker event again was not classified as a preventable functional failure. The licensee stated that the functional failure was determined to be "not preventable" under the maintenance rule based on the facts that, prior to the event, there were no previous failures of this type of breaker and there were no industry events of this type.

The inspectors noted that the licensee had refurbished the alternate shutdown board breaker using maintenance procedure MI-10.5, Revision 27, Westinghouse Type DS Breaker and Switchgear Maintenance. Sections 6.4.2.1 through 6.4.2.17 performed the breaker adjustments to obtain appropriate contact compression. These sections had

been signed off as having been completed. However, following the breaker failure on May 19, 1998, the licensee reported that "A Westinghouse representative identified a gap on the "C" phase contacts that could not be properly adjusted even with allowances for missing contact material on the main "C" phase contacts." The licensee could not explain why the "C" phase contacts were found out of adjustment following one inservice operation of the breaker. It appeared that the "C" phase contacts had been misadjusted during the refurbishment activity and that subsequent contact adjustment verifications had failed to identify the degraded condition. Although the licensee has identified additional problems with cam rotation, closing spring tension and frame bowing, these problems did not explain the observed gap on the "C" phase contacts.

Based on the observed as found misadjusted condition of the "C" phase contacts, it appeared to the inspectors that the May 19, 1998, breaker failure was a maintenance preventable functional failure and that the licensee had not properly categorized this event. The licensee's categorization appeared to be contrary to the requirements of the maintenance rule as required by 10 CFR 50.65. Pending further review, the potential failure to meet the maintenance rule requirements, is identified as an Unresolved Item (URI 50-327/98-09-03).

c. Conclusions

An unresolved item was identified for the potential failure to properly categorize a maintenance preventable functional failure.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Inspector Followup Item (IFI) 50-327, 328/97-18-08: Remaining GL 89-10 concerns.

This followup item was opened pending the licensee's completion of commitments which addressed issues raised during NRC inspections of the licensee's implementation of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The NRC inspectors reviewed records during the current inspection and confirmed that the commitments had been satisfactorily completed, as described below:

Commitments - Revise Torque Calculations for 18-inch and Smaller Pratt Butterfly Valves and Obtain Additional Support for the Torque Requirements Determined for 20 and 24-inch Pratt Butterfly Valves

The torque requirements for the Sequoyah's Pratt butterfly valves were determined by the vendor. However, as documented in Inspection Report 50-327, 328/97-18, preliminary test data from Duke indicated that the torque requirements determined for Sequoyah's 18-inch and smaller butterfly valves might be non-conservative. In addition, the licensee did not have adequate test data to support the torque requirements determined for Sequoyah's 20 and 24-inch Pratt butterfly valves. The licensee committed to revise the torque calculations for the 18-inch and smaller valves based on final Duke test data. For the larger valves, the licensee committed to obtain test data or

to use the Electric Power Research Institute (EPRI) Performance Prediction Methodology (PPM) to validate the vendor torque requirements.

In the current inspection, the inspectors found that the licensee had not revised the calculations for the 18-inch and smaller valves but had instead obtained further data to support the vendor predicted torque requirements previously established. This data addressed the full range of Sequoyah's Pratt butterfly valve sizes (up to and including 24-inch). The data was described and evaluated in Commitment Completion Tracking Nos. NCO970056014 and NCO970056015 and was subsequently incorporated into TVA Mechanical Design Standard DS-M18.2.21, "Motor Operated Valve Thrust and Torque Calculation," Revision 10. The data included the final results of torque testing performed by Duke, motor current measurements obtained during static and dynamic tests on Sequoyah's valves, and summary torque test information from other licensee's with Pratt butterfly valves. The inspectors considered the licensee's data and evaluation sufficient to support the capabilities of Sequoyah's Pratt butterfly valves in their current condition and to meet the intent of the licensee's commitments.

Duke identified significant Nylatron bearing degradation in several Pratt butterfly valves. For these valves, the vendor calculated torque requirements were exceeded. Although Sequoyah's valves contained this bearing material, there was no evidence that serious degradation was present. Sequoyah's valves generally had capabilities well-above the vendor-specified torque requirements (greater than 50% for most valves) and motor current measurements obtained during dynamic and static valve tests did not reveal any degradation. The inspectors questioned whether the licensee's commitment to Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," could be met. The licensee indicated that evaluations and trending of motor power measurements would be used to meet the Generic Letter 96-05 commitments. The issue of the acceptability of the licensee's method of meeting Generic Letter 96-05 commitments for the Pratt butterfly valves was discussed with the cognizant reviewer in the NRC Office of Nuclear Reactor Regulation on October 7, 1998.

<u>Commitment - Perform EPRI PPM Calculations for Valve Groups 1, 2, and 8 and Incorporate the Results into Their Design and Setup Calculations</u>

As described in Inspection Report 50-327, 328/97-18, inspectors were concerned that the licensee did not have adequate test data to support the valve factors assumed in calculating thrust requirements for valve Groups 1, 2, and 8. In response, the licensee committed to use the Electric Power Research Institute (EPRI) Performance Prediction Methodology (PPM) to determine thrust requirements for these valves (taking into account any blowdown performance requirements) and to incorporate the results into the valve design and setup calculations.

The inspectors confirmed that the licensee had completed the above commitment actions by reviewing the following examples of the design and setup calculations and verifying that they appropriately incorporated EPRI PPM results:

- Group 1 - Calculation 2-FCV-01-018 (GL 89-10); "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 2-FCV-01-018;" Revision 5.

- Group 2 - Calculation SQN-01-D053 EPM-RJP-040291; "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 1-FCV-01-017;" Revision 6.

- Group 8 - Calculation 2-FCV-63-001 (GL 89-10); "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 2-FCV-63-1;" Revision 3.

Commitment - Perform EPRI PPM Calculations and Maintenance Improvements on the Pressurizer Power Operated Relief Valve (PORV) Block Valves

Inspection Report 50-327, 328/97-18 identified a concern that the licensee had used pumped flow test results to establish thrust requirements for PORV block valves, whereas these valves experience more severe blowdown flow in worst-case design accident conditions. In response to this concern, the licensee committed to use the EPRI PPM to determine thrust requirements for these valves. In addition, the licensee committed to perform maintenance improvements on the valves' internals to assure they could accommodate the blowdown flow expected in a worst-case accident.

The inspectors confirmed that the licensee had satisfactorily completed these actions by reviewing the following examples of the block valve calculations and completed work orders (WOs) for the internal maintenance and inspections performed on the block valves:

- Calculation SQN-68-D053 EPM-RJP-041091; "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 1-FCV-68-332;" Revision 5.

- Calculation 2-FCV-68-332 (GL 89-10); "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 2-FCV-68-332;" Revision 3.

- WO 97-009286-000, Valve 1-FCV-068-332, Refurbish valve internals and take internal dimensions.

- WO 97-009285-000, Valve 1-FCV-068-333, Refurbish valve internals and take internal dimensions for NRC GL 89-10 commitment.

Commitment - Implement Design Changes to Increase Actuator Capabilities of Containment Spray Valves 1/2FCV-72-002/039

Inspection Report 50-327, 328/97-18 identified a concern that Containment Spray Valves 1/2FCV-72-002/039 had marginal capabilities to perform their design-basis functions. In response, the licensee committed to implement actuator design changes

to increase the capabilities of these valves. These changes were to be completed during the next outage for each unit.

The inspectors verified that the changes (gearing changes) had been implemented on the Unit 1 valves through a review of the following examples of the design change, calculation, work completion, and test records:

- Plant Modifications and Design change Control Closure Memo, Record No. S02 980930 851, Completion of work orders for Design Change Notice (DCN) M13771A.

- Calculation 1-FCV-72-002 (GL 89-10); "Documentation of Design Basis Review, Required Thrust Calc and Valve & Actuator Capability Assessment for 1-FCV-72-002;" Revision 3.

- DCN M13771A (Modify 1-FCV-72-002 and -039 to meet thrust requirements of GL 89-10. Also modify as required to alleviate pressure locking concerns of GO 95-07.), Revision 1.

- WO 98-001522-006, Regearing of valves 1-FCV-72-002 and -039, completed September 12, 1998.

- Test data sheets (diagnostic tests) for valves 1-FCV-72-002 and -039, completed September 12, 1998.

For the Unit 2 valves, the inspector verified that the licensee maintained tracking of the commitment with the changes to be completed at the next outage. The Unit 2 commitment was tracked as Item NCO970056013, scheduled for completion during the Unit 2, Cycle 9 refueling outage.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Review of U1C9 Outage

a. Inspection Scope (71750)

The inspectors reviewed the U1C9 radiological performance.

b. Observations and Findings

The licensee reported that the total dose for U1C9 was 200 rem, compared to the established goal of 212 rem. This dose represented the lowest dose in the history of Unit 1 and was 36 rem less than the previous record. The low dose was achieved even though there existed the radiological impact due to failed fuel. The licensee attributed the low dose to effective ALARA planning, scheduling, engineering controls, and personnel performance.

c. Conclusions

Radiological controls during the U1C9 outage were effective as evidenced by a record low total dose for Unit 1 of 200 rem.

R1.2 Occupational Radiation Exposure Control Program

a. Inspection Scope (83750)

The inspectors reviewed implementation of selected elements of the licensee's radiation protection program during the current Unit 1 Refueling Outage (RFO). The review included observation of radiological protection activities including personnel exposure monitoring, radiological postings, verification of posted radiation dose rates and contamination levels within the Radiologically Controlled Area (RCA), and primary coolant shutdown chemistry controls for dose rate reduction. Those activities were evaluated for consistency with the programmatic requirements, personnel monitoring requirements, occupational dose limits, radiological posting requirements, and survey requirements specified in Subparts B, C, F, G, and J of 10 CFR 20.

b. Observations and Findings

The inspectors conducted frequent tours of the RCA to observe radiation protection activities and practices. Personnel preparing for routine entries into the RCA were observed being briefed on the radiological conditions in the areas to be entered. The briefings were given by radiation control personnel before access was granted and covered the dosimetry and the protective clothing and equipment required by the Radiation Work Permit (RWP) for the entry. The administrative limits for the allowed dose and dose rate for the entry were emphasized during the briefings. The briefings provided thorough descriptions of the existing dose rates which could be encountered during the entry. The inspectors determined that personnel entering the RCA were adequately briefed on the radiological hazards which could be encountered while in the RCA and the radiological protective measures required to be taken during the entry. Individuals at selected job sites were interviewed and it was determined that the workers were aware of their administrative dose and dose rate limits, the work area dose rates, the proximate low-dose waiting areas, areas of high contamination, and protective clothing required by the RWP.

The inspectors observed the use of personal radiation exposure monitoring devices by personnel entering and exiting the RCA. Thermoluminescent dosimeters (TLDs) were used as the primary device for monitoring personnel radiation exposure. In addition, digital alarming electronic dosimeters (EDs) were used for monitoring the accumulated dose and the encountered dose rates during each RCA entry. The EDs were set to alarm at administrative limits established for the specific RWP under which the RCA entry was being made. As the individuals exited the RCA the accumulated dose and encountered dose rate information was transferred from the EDs to the Radiation Exposure System (REXS) data base in order to track individual exposures. During tours of the RCA the inspectors noted that the required dosimetry was being properly worn by personnel when entering and while in the RCA. The inspectors also noted that

personnel exiting the RCA routinely surveyed themselves for contamination using personnel contamination monitors (PCMs).

During tours of the RCA the inspectors noted that general areas and individual rooms were properly posted for radiological conditions. Survey maps indicating dose rates and contamination levels at specific locations within the RCA were posted at the entrance to the RCA. Survey maps were also posted at individual contaminated and high radiation areas. At the inspector's request, a licensee Health Physics Technician performed dose rate and contamination surveys in several rooms and locations. The inspectors verified that the survey instrument readings were consistent with the posted area dose rates. Independent contamination surveys performed around several posted contaminated areas indicated that contamination was not being tracked out of the contaminated areas.

The inspectors compiled the annual and outage collective dose data presented in the table below from the licensee's REXS and As Low As Reasonably Achievable (ALARA) reports. The annual collective doses were verified to be consistent with the REXS data base which is used by the licensee to record and monitor personnel radiation exposure. As indicated in the table, the licensee was usually successful (7 of 10) at meeting established ALARA goals.

Collective Dose (Man-Rem)							
Annual Dose				Outage Dose			
Fiscal Year	Actual	Goal	3 Year Mean	Unit/ Cycle	Actual	Goal	Days
1994	320[1]	400	251	U1C6[4]	361[2]	300	377
				U2C6[4]	236[2]	300	134
1995	239[1]	185	294	U1C7[5]	292[2]	250	64
1996	402[1]	408	320	U2C7[5]	212[2]	225	51
1997	280[1]	300	307	U1C8[5]	236[2]	244	51
				U2C8[5]	140[2]	173	30
1998	255[2,3]	450		U1C9[5]	82[2,3]	212	28

[1] TLD data
[2] ED data
[3] Year-to-date as of 9/17/98
[4] Extended outage due to corrosion/erosion in secondary piping
[5] RFO

The licensee also provided the inspectors with data from the Radiation Exposure System (REXS) data base pertaining to maximum individual radiation exposures for the years 1994 through 1997 and year-to-date 1998. The inspectors verified that the data were consistent with the REXS data base and tabulated the data in the table below.

Maximum Individual Radiation Doses (Rem)				
Year	TEDE	Skin	Extremity	Eye Lens
1994	1.735	1.795	3.608	1.743
1995	2.295	2.537	5.051	2.337
1996	1.482	1.980	2.930	1.513
1997	2.259	2.318	2.738	2.288
1998[1]	0.604	2.203	2.203	0.609
Regulatory and Administrative Limits				
10 CFR 20	5.000	50.000	50.000	15.000
Admin.	1.000	None	None	None

[1] Year-to-date as of 9/16/98

The administrative annual dose limits established by the license were delineated in section 8.2 of procedure RCI-3, Personnel Monitoring. That section of the procedure specified that the 1.0 rem administrative limit could be exceeded only if authorized by the Radiological Control and Chemistry Manager, and that exposures exceeding 5.0 rem required authorization by the Radiological Control and Chemistry Manager, the Plant Manager and the Site Vice President. As indicated in the table, the maximum individual radiation exposures during the years 1994 through 1997 and year-to-date 1998 were well within the regulatory limits for occupational dose specified in 10 CFR 20.1201(a).

The inspectors reviewed the licensee's procedures for follow-up actions to Personnel Contamination Events (PCEs) and reviewed selected records for those events which occurred during 1998. Procedure RMD FO-10 Personnel Contamination Reports, indicated that the threshold for initiating follow-up actions was skin or clothing contamination in excess of 100 net counts per minute (ncpm) as measured by a hand held frisker. The licensee's records indicated that 693 PCEs occurred prior to the start of the Unit 1 outage, 654 of which were due to Noble gas activity. Four days into the outage, which started on September 9, 156 PCEs had occurred, 105 of which were due to Noble gas activity. The licensee indicated that the Noble gas activity was present in the RCA as a result of leaking fuel in Unit 1 and that the leaking fuel would be replaced for the next fuel cycle. The inspectors noted that there were no uptakes of radioactive material in excess of one percent of the Annual Limit on Intake (ALI), and therefore, pursuant to section 6.5 of procedure RCI-11 Bioassay Program, no internal dose assignments were made. Skin dose assessments were initiated for four PCEs which occurred prior to and for one after the start of the outage. Procedure RCI-1 Personnel Monitoring, specified that skin dose assessments were to be initiated whenever a worker may have received a significant dose (>100 mrem) from skin or personal clothing contamination. At the time of this inspection three of those assessments were complete and the inspectors verified that the assigned doses had been entered into the individuals dose records in the REXS data base. No regulatory dose limits were exceeded.

The inspectors also reviewed the licensee's records for contaminated floor space within the RCA. Radiological Control personnel maintained records of the areas within the RCA, excluding the Containment Buildings, which had contamination levels in excess of 1000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2). The licensee tabulated the contaminated square footage, as of the last day of each month. The inspectors noted from that tabulation that the month ending values for contaminated floor space were less than one percent of the RCA floor space, year-to-date 1998.

The inspectors also reviewed the licensee's plans for primary chemistry controls during the reactor shutdown for the U1C9 RFO. The general plan for the shutdown chemistry controls included early injection of boric acid into the coolant during cooldown followed by injection of hydrogen peroxide after cooldown. The objective of the plan was to cause a controlled release of radioactive materials from the internal surfaces of the R.C.S. and to remove those materials from the coolant by use of the reactor water clean-up system. Specific plans consisted of injecting boron at a rapid rate to achieve an acid reducing environment, controlling the temperature and pH such that the coolant remained acidic until the coolant changed to an oxidizing environment by the injection of hydrogen peroxide, and maintaining the hydrogen concentration in the coolant above a specified level (10cc/kg) during cooldown in order to keep the released material in soluble chemical compounds. One specific goal of the chemistry control plan was to reduce the cobalt-58 (^{58}Co) concentration to less than 0.05 micro-Curies per milliliter (μCi/ml) to minimize radiation shine from refueling water during the outage. The ^{58}Co concentration peaked at 2.3 μCi/ml during the oxidizing phase of the process and was then reduced to 0.02 μCi/ml after 100 hours of clean-up operations. The licensee monitored and controlled many other chemical parameters throughout the process including the amount of ^{58}Co and nickel (Ni) removed from the R.C.S. Approximately one kilogram (kg) of ^{58}Co and 2.5 kgs of Ni were removed during this process. The inspectors reviewed analytical results for selected chemistry parameters and determined that the licensee had closely monitored and controlled primary coolant chemistry during the shutdown for the U1C9 RFO.

c. Conclusions

Based on the above reviews and observations, the inspectors concluded that the licensee was properly monitoring and controlling personnel radiation exposure during the Unit 1 refueling outage and posting area radiological conditions in accordance with 10 CFR Part 20. Personnel entering the RCA were adequately briefed on radiological hazards and protective measures. Maximum individual radiation exposures were controlled to levels which were well within the regulatory limits for occupational dose specified in 10 CFR 20.1201(a). The licensee was generally successful in meeting established ALARA goals. The licensee had implemented an effective shutdown chemistry control plan and closely monitored primary coolant chemistry during the shutdown for the Unit 1 refueling outage.

R.8 Miscellaneous RC&P Issues (92904)

R8.1 (Closed) IFI 50-327, 328/98003-08: Loss of data on computer generated survey maps. The licensee had discovered a software problem with a recently implemented computer system for generating survey maps. When files were updated from current survey

results and new survey maps were printed, the new maps did not show all of the survey results. The software vendor assisted the licensee in resolving this problem. The inspectors verified that hard copies of required survey maps were maintained for the period during which the problem existed. This item is closed.

R8.2 (Closed) URI 50-327, 328/98003-09: Use of alternate method for analyzing post accident hydrogen samples. During a telephone conversation on September 11, 1998, the licensee provided regional management with a chronology of the identified problems with the initial analytical method and the submittal of revisions to the FSAR which described the alternate analytical method. Based on the information provided, this item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on October 14, 1998 and, for region-based inspections, on October 2 and October 9, 1998. The licensee acknowledged the findings presented.

During the inspection period, the inspectors asked the licensee whether any materials would be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

*Bajestani, M., Site Vice President
*Burton, C., Engineering and Support Services Manager
*Butterworth, H., Operations Manager
*Gates, J., Site Support Manager
*Freeman, E. Maintenance and Modifications Manager
*Herron, J., Plant Manager
*Kent, C., Racon/Chemistry Manager
*Kehl, D, Assistant Plant Manager
*O'Brien, B., Maintenance Manager
*Sales, P., Manager of Licensing and Industry Affairs
*Valence, J., Engineering & Materials Manager

* Attended exit interview

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations

IP 71750: Plant Support Activities
IP 71707: Plant Operations
IP 92901: Follow-up - Operations
IP 92902: Follow-up - Maintenance
IP 92903: Follow-up - Engineering
IP 92904: Follow-up - Plant Support
IP 83750: Occupational Radiation Exposure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
VIO	50-327/98-09-01	Open	Inadequate PUT of Type DS 532 Breakers (Section M2.1).
IFI	50-327/98-09-02	Open	Review Boron Corrosion Control Program (Section E2.1).
URI	50-327/98-09-03	Open	Breaker Failure Not Categorized As a Maintenance Preventable Functional Failure (Section E2.2).

Closed

Type	Item Number	Status	Description and Reference
URI	50-327/98-06-02	Closed	Potential Improper Corrective Maintenance Activities Related to Improper Breaker Contact Compression Setting and Inadequate Post Maintenance Testing (Section M8.1).
LER	50-327/98-01-00	Closed	Failure of the 480-Volt Shutdown Board 1A1-A Alternate Feeder Breaker Results in a Unit 1 Reactor Trip on a Low Steam Generator Water Level (Section M8.2).
IFI	50-327,328/98-03-08	Closed	Loss of Data on Computer Generated Survey Maps (Section R8.1).
URI	50-327,328/98-03-09	Closed	Use of Alternate Method for Analyzing Post Accident Hydrogen Samples (Section R8.2).
IFI	50-327,328/97-18-08	Closed	Remaining GL 98-10 Concerns (Section E8.1).

LIST OF ACRONYMS USED

ALARA	-	As Low As Reasonably Achievable
amp	-	ampere
ANSI	-	American National Standards Institute
CFR	-	Code of Federal Regulations
CRDMG	-	Control Rod Drive Motor Generator
dc	-	Direct Current
DCN	-	Design Change Notice
EDG	-	Emergency Diesel Generator
EPRI	-	Electric Power Research Institute
GL	-	Generic Letter
GOI	-	General Operating Instruction
ICS	-	Integrated Computer System
IFI	-	Inspector Follow-up Item
IR	-	Inspection Report
ISI	-	In-Service Inspection
kV	-	Kilo-Volt
kVA	-	Kilo-volt ampere
LER	-	Licensee Event Report
MI	-	Maintenance Instruction
MVA	-	Mega Volt-Amperes
MVAR	-	Mega Volt-Amperes-Reactive
MW	-	Mega Watts
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
PER	-	Problem Evaluation Report
PMT	-	Post Maintenance Test
PORV	-	Power Operated Relief Valve
ppm	-	Parts Per Million
RCS	-	Reactor Coolant System
rem	-	radiological equivalent man
RHR	-	Residual Heat Removal
RO	-	Reactor Operator
rpm	-	revolutions per minute
RP&C	-	Radiological Protection and Chemistry
SI	-	Surveillance Instruction
SRO	-	Senior Reactor Operator
TS	-	Technical Specifications
TVA	-	Tennessee Valley Authority
U1C9	-	Unit 1 Cycle 9
URI	-	Unresolved Item
Vac	-	Voltage-Alternating Current
VIO	-	Violation
WO	-	Work Order