March 9, 2001

Mr. Charles H. Cruse
Vice President
Constellation Nuclear
Calvert Cliffs Nuclear Power Plant, Inc.
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: NRC's CALVERT CLIFFS NUCLEAR POWER PLANT INSPECTION REPORT 05000317/2000-012, 05000318/2000-012

Dear Mr. Cruse:

On February 10, 2001, the NRC completed an inspection at your Calvert Cliffs Nuclear Power Plant Units 1 & 2. The enclosed report documents the inspection findings which were discussed on February 23, 2001, with Mr. Katz and other members of your staff.

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

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

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Sincerely,

/RA by
William A. Cook
Acting For/

Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

Docket Nos.: 05000317 and 05000318
License Nos.: DPR-53 and DPR-69

Enclosure: Inspection Report 05000317/2000-012, 05000318/2000-012

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REGION I

Docket Nos:	05000317, 095000318
License Nos.:	DPR-53, DPR-69
Report No:	05000317/2000-012, 05000318/2000-012
Licensee:	Calvert Cliffs Nuclear Power Plant, Inc.
Facility:	Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Location:	1650 Calvert Cliffs Parkway Lusby, MD 20657-4702
Dates:	December 31, 2000 - February 10, 2001
Inspectors:	David Beaulieu, Senior Resident Inspector Tim Hoeg, Resident Inspector William Cook, Senior Project Engineer Neil Della Greca, Senior Reactor Engineer John Caruso, Operations Engineer Gregory Smith, Senior Physical Security Inspector
Approved by:	Michele G. Evans, Chief, Projects Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR05000317-00-12, IR05000318-00-12, on 12/31/00 - 2/10/01, Calvert Cliffs Nuclear Plant, Inc.; Calvert Cliffs Nuclear Power Plant, Units 1 & 2. Maintenance Rule Implementation, Surveillance Testing

The inspection was conducted by resident inspectors, regional senior projects and reactor engineers, a regional operations engineer, and a regional senior physical security specialist. The inspectors identified two Green findings, both of which involved Non-cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- **Green.** The inspectors identified two examples of a Non-cited Violation for inadequate design control. The first example involved the failure to ensure operability of the containment spray pump supply breaker following a test failure that occurred two days after a modification to the breaker. The second example involved the failure to ensure proper operability of five similar, but normally closed, replacement load center feeder breakers.

  The first example was found to be of very low significance because the loss of containment spray function does not affect core damage frequency; Manual Chapter 0609, Appendix H, did not list containment spray as a system having major impact on large early release frequency; and at the time of the inspection the licensee had already initiated corrective actions to address the failure and to prevent recurrence.

  The second example was also of very low significance because the licensee’s physical inspection of the switchgear indicated acceptable alignment between the breakers and their housing and between the racking block plate and the breaker front covers. If tripped as a result of a seismic event, the breakers could be closed manually by the operators. (Section 1R12).

- **Green.** The inspectors identified a Non-cited Violation for failure to satisfy the technical specification surveillance requirement to verify boron concentration in each safety injection tank (SIT) every 31 days. This violation occurred due to an inadequate technical justification of an alternate method of verifying boron concentration.

  The safety significance of this finding was very low because subsequent SIT sampling determined that SIT boron concentrations remained within the technical specification required band. (Section 1R22)

Report Details

Units 1 and 2 operated at or near 100 percent power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors conducted an equipment alignment partial walkdown to evaluate the operability of selected redundant trains or backup systems, while the affected train or system was inoperable or out-of-service. The walkdown included a review of system operating instructions to determine correct system lineup and verification of critical components to identify any discrepancies which could affect operability of the redundant train or backup system. The inspectors performed a partial system walkdown on the following system:

- Unit 1 High Pressure Safety Injection (HPSI) was inspected on February 6, 2001, while the 13 HPSI pump was out-of-service. The operating procedure used for the walkdown was OI-3A, “Operation of Emergency Core Cooling Systems (ECCS).”

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspector conducted a complete 125 Volt Vital DC system equipment alignment walkdown of both units to verify equipment alignment and to identify any discrepancies the may impact the function of the system. The inspector also reviewed the licensee’s actions to identify and resolve system equipment discrepancies which may cause an initiating event or impact the mitigation capability of an associated system. The walkdown included reviews of system operating instructions and electrical diagrams to determine correct system/breaker lineups. The inspectors reviewed the following station documentation:

- OI-26A, “125 Volt Vital DC,” Revision 12
- 125 VDC Electrical Power Distribution System Description No. 002, Revision 0
- Procedure EOP-8, “Functional Recovery Procedure,” Revision 19
- AOP-7I, “Loss of 4KV, 480 Volt or 208/120 volt Instrument Bus Power,” Revision 18
- AOP-7J, “Loss of 120 Volt Vital AC or 125 Volt Vital DC Power,” Revision 13

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed flood protection measures for external sources as described in the Updated Final Safety Analysis Report (UFSAR), Individual Plant Examination (IPE), and emergency procedures. This inspection included tours of plant areas identified as risk significant including the Unit 1 and 2 auxiliary feedwater pump rooms, emergency core cooling system pump rooms, and the service water heat exchanger rooms. Water tight doors, floor drains, penetrations, level alarm circuits, and sump pumping systems were verified to be functional.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Licensed Operator Simulator Training

a. Inspection Scope

On January 29, 2001, the inspector observed licensed operator simulator training to assess operator performance for a scenario involving a main steam line break inside containment. In particular, the inspector observed operators perform procedure AOP-7J, Loss of 120 Volt Vital AC or 125 Volt Vital DC Power, Section V, which the licensee's probabilistic risk assessment has determined involves important operator actions. Following the simulator exercise, the inspector observed the training instructor debrief and critique, which included a discussion of previous lessons learned.

b. Findings

No findings of significance were identified.

.2 Training and Qualification Effectiveness

a. Inspection Scope

The inspector reviewed the root cause analysis and planned corrective actions (Action Item Tracking No. IR200001014) for the exam security issues (Non-cited Violations) identified during the preparation phase of the operator licensing examinations administered in September 2000 (reference Initial Examination Report 05000317/2000-301 and 05000318/2000-301).

Additionally, the following areas were reviewed using the guidance provided in NRC Inspection Procedure 41500, Training and Qualification Effectiveness: security procedures for exam development for both the Licensed Operator Initial Training (LOIT) and Licensed Operator Requalification Training (LORT) programs, recent changes and revisions to the LOIT student qualification manual, employee concerns records, the licensee's guidance for updating training material for both the LOIT and LORT programs, as well as, a sample of lesson plans for both programs.

Interviews were also conducted with the instructors and supervisors for both the LOIT and LORT programs concerning exam security measures and recent updates to the LOIT student qualification manual.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Maintenance Rule Unavailability During Testing

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) 10 CFR 50.65 (a)(1) and (a)(2) classifications; (2) the appropriateness of performance criteria for SSCs classified as (a)(2); and (3) goals and corrective actions for SSCs classified as (a)(1). The following SSCs were reviewed:

- 1A Emergency Diesel Generator (EDG) - The review focused on the licensee's determination that the 1A EDG is considered available, per the maintenance rule, while performing surveillance procedure STP O-008A-1, "Test of 1A EDG and 11 4KV Bus LOCI Sequencer," which parallels the 1A EDG to the grid.

- Steam Driven Auxiliary Feedwater Pumps 11, 12, 21, and 22 - The review focused on the licensee's determination that the AFW pumps are considered available per the maintenance rule while performing surveillance procedure STP O-005A-1, "Auxiliary Feedwater System Quarterly Surveillance Test." In particular, the inspector evaluated the crediting of operator action to reset the trip/throttle valve and whether this satisfied the maintenance rule guidance of being an uncomplicated, simple action that is virtually certain to be successful. The inspector verified that minor discrepancies that were identified were appropriately documented in Issue Report No. IR3-041-489.

b. Findings

No findings of significance were identified.

.2 Number 12 Containment Spray Pump Circuit Breaker Failed to Close

a. Inspection Scope

As described in Licensee Event Report 50-317/2000-006, on November 10, 2000, the Asea Brown Boveri (ABB) 4-kV vacuum circuit breaker for the No. 12 containment spray pump failed to close upon receipt of an auto-start signal. The failure occurred while the breaker was undergoing functional testing, that included the automatic starting of the pump. Subsequent troubleshooting determined that the breaker was most likely in a trip-free condition when the breaker close signal was applied. This same circuit breaker had failed to close during similar surveillance testing on October 12, 2000.

The purpose of the inspection was to evaluate the circumstances surrounding the failures of the circuit breaker, the actions taken by the licensee after the first failure, and the basis for operability of the affected systems. The evaluation included interviews of engineering and maintenance personnel, physical inspection of the failed breaker and associated switchgear cubicle, and a review of applicable documents, including incident reports, operability evaluations, root cause analysis, and electrical schematics.

b. Findings

Switchgear/Breaker Design

To address aging concerns regarding the General Electric (GE) 4-kV Magne-Blast circuit breakers, the licensee decided to replace them with vacuum breakers designed by Asea Brown Boveri. The new breakers were constructed to fit in the GE-designed switchgear housing.

Similar to the GE breakers, the trip mechanism of the ABB breakers can be actuated by three separate and independent devices: the electrical trip coil; the front-mounted mechanical trip button; and the racking interlock assembly. All three devices operate in the same manner. When actuated, they depress the trip bar and cause the breaker to trip, if closed. The device responsible for the breaker trip-free operation was the racking interlock assembly. Its purpose is to ensure personnel safety by preventing breaker movement while the breaker is closed. Specifically, during the racking operation, insertion of the racking tool into the interlock assembly causes the interlock to lower the trip bar and trip the breaker. A close signal received by the breaker while the trip bar is in its lowered position, prevents the breaker from closing and causes it to go through a trip-free operation, i.e., causes the closing springs to be discharged, while the main contacts remain open (tripped).

During breaker insertion, a notched steel bar on the left side of the circuit breaker housing, maintains the spring-loaded interlock assembly engaged and the trip bar depressed until the breaker is fully inserted and the racking tool removed. The notch in the steel bar is positioned to allow the return of the racking interlock to its normal position and to provide positive indication when the breaker is fully inserted.

Containment Spray Pump Breaker Failure

On October 10, 2000, the licensee performed a modification to replace the existing GE Magne-Blast breaker for the containment spray pump with a new ABB breaker. Two days later, on October 12, 2000, while performing testing in accordance with surveillance test procedure (STP) O-7B-1, the ABB breaker failed to close. The failure occurred while verifying that the pump would start upon receipt of a safety injection actuation signal (SIAS). Troubleshooting by an electrical maintenance team, including a vendor technical representative, failed to duplicate and identify the source of the problem. Based on the breaker passing a subsequent STP, on October 13, 2000, the licensee declared the system operable, but decided, as a compensatory action, to retest the breaker in 29 days rather than quarterly.

On November 10, 2000, (29 days later) the breaker failed to close. Again, the licensee's troubleshooting team was not able to duplicate the problem, but later that day, during re-performance of STP O-7B-1, an electrician observed that the breaker cycled, but did not close, indicating trip-free operation. As a result of this third failure, the licensee replaced the failed ABB breaker with a refurbished GE breaker and initiated an investigation that included a priority 2 causal analysis (IR200001064). The licensee concluded that the three failures were most likely the result of the breaker not being fully racked-in. In the analysis the licensee also stated that the "Trip-free Interlock Mechanism is susceptible to a spurious actuation when the breaker is even a small distance away from being fully racked-in."

Follow-up and observations by the inspector determined that rubbing of the new ABB breaker against the housing in two areas prevented the smooth insertion of the breaker into its cubicle and potentially gave the technician a false indication of the insertion status of the breaker. In addition, dimension tolerances between the GE and the ABB breakers, not specifically verified by the licensee, potentially prevented the racking interlock assembly from returning completely to its normal position, after breaker insertion. In this case, the partially lowered trip bar caused the breaker to be intermittently in the trip-free position. The licensee believed that lack of full insertion and breaker movement during the STP close-open cycles caused the breaker to go intermittently into the trip-free position.

Based on the results of their analysis, the licensee initiated action to immobilize the racking interlock assembly of all the installed ABB breakers with an automatic start function. This was done by bolting the racking block plate to an existing bracket. The inspector confirmed that the bolting of the plate prevented any motion of assembly.

In addressing the safety significance of this breaker failure, the licensee stated that the 12 containment spray pump was considered inoperable from the installation of the ABB breaker on October 10, until its replacement with the GE breaker on November 12, 2000. Regarding the other installed ABB breakers, the licensee believed that the associated systems were always operable, based on the positive results of the functional tests performed.

The inspector determined that the licensee's failure to ensure operability of the containment spray pump following installation of the ABB breaker was more than a minor violation, in that it had a credible impact on safety. The inspector also determined that this issue was a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. This issue was assessed using the significance determination process and was found to be of very low significance (Green). The loss of containment spray function constitutes a reduction of the atmospheric pressure control capability, potentially affecting reactor containment integrity. Since this potential loss does not affect core damage frequency, it is considered a type B finding in Appendix H of Manual Chapter 0609. Table 2.2 of Appendix H does not list containment spray as a system having major impact on large early release frequency of pressurized water reactor containment, as in the case of the Calvert Cliffs station. Due to the overall low risk significance, this violation of 10 CFR 50, Appendix B, is being treated as a Non-cited Violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000317/2000-012-01)**

Extent of Condition

As stated previously, to ensure that the ABB breakers with an automatic start function would be able to close on demand, the licensee immobilized the racking interlock assembly by bolting the racking block plates. This was done to ensure that any breaker movement during the opening operation did not place the assembly and the breaker into a trip-free condition. Regarding an additional five normally-closed ABB breakers that supply power to five safety-related load centers, the licensee delayed the bolting of the assembly to an appropriate later date. Their decision was based, in part, on the fact that the breakers were already closed and, therefore, the trip-free operation did not apply.

The inspector concluded that the licensee's decision regarding the normally open automatic breakers was reasonable and acceptable. However, regarding the five normally closed breakers, the inspector concluded that the licensee's conclusions were not appropriate. As indicated previously, the failure of the containment spray pump breaker was the result of dimension tolerances and design differences between the GE and ABB breakers potentially complicated by breaker motion during breaker operation. The inspector concluded that, during a seismic event, a normally closed breaker was potentially affected by the same conditions, i.e., dimension tolerances, design differences, and motion. These conditions could combine to cause a normally-closed breaker to open (trip) inadvertently and result in a loss of power to shutdown equipment.

The inspector determined that the licensee's failure to ensure operability of the load center supply breakers following their replacement with ABB-designed breakers was more than a minor violation. The inspector determined that this is a second example of a violation of 10 CFR 50, Appendix B, Criterion III. The result of this Criterion III violation was assessed using the significance determination process and was found to be of very low significance (Green) for the following reasons: (1) the licensee's physical inspection of the switchgear indicated acceptable alignment between the breakers and their housing and between the racking block plate and the breaker front covers; (2) the licensee had experienced no operation anomalies during post-modification breaker functional testing; (3) the breakers had been seismically qualified, albeit under controlled conditions; (4) the inspector had no evidence that an interaction existed between the breakers and the housing such that it would cause the breakers to trip during a seismic event; and (5) if tripped, the breakers could be closed manually by the operators following the seismic event. Due to the overall low risk significance, this violation of

10 CFR 50, Appendix B, Criterion III, is being treated as a second example of **NCV 05000317; 318/2000-012-01.**

As a result of the inspector's observation, the licensee initiated an incident report and, subsequently, immobilized the affected breaker racking interlock assemblies.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For the selected Maintenance Order (MO) activities listed below, the inspectors verified: (1) risk assessments were performed in accordance with procedure NO-1-117, Integrated Risk Management; (2) risk of scheduled work was managed through the use of compensatory actions; and (3) applicable contingency plans were properly identified in the integrated work schedule. Specifically, the inspector observed and discussed the troubleshooting activities with the responsible technicians and supervisors involved with the emergent work associated with an electrical ground in the 2B emergency diesel generator (EDG) supervisory circuit and a blown fuse in the output breaker control circuit.

- MO 2200001476 2B EDG Bus 24 feeder breaker inspection per FTE-51
- MO 2200002031 Perform alarm/setpoint procedure on 2B EDG jacket cooling water temperature switches
- MO 2200002032 Perform alarm/setpoint procedure on 2B EDG temperature instruments

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant mitigating systems, to assess: (1) the technical adequacy of the evaluations, (2) whether continued system operability was warranted, and (3) whether other existing degraded conditions adversely impacted the evaluation or associated compensatory measures.

- Operability Determination No. 99-02, System 011, Unit 1, SRW Pump #12 and #13 Discharge Check Valves (1CKVSRW-315 and 1CKVSRW-316), dated February 17, 1999.

- Operability Determination No. 99-04, Safety Injection and Containment Spray Pumps, dated March 12, 1999.

- Operability Determination No. 00-07, Fire Suppression Water System Number 12 Pretreated Water Storage Tank, dated November 16, 2000.

- Operability of the 12 containment spray pump breaker, discussed in Section 1R12 of this report.

b. Findings

For the first three operability determinations list above, no findings of significance were identified. Findings associated with the 12 containment spray pump breaker are discussed in Section 1R12 of this report.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated selected risk significant operator workarounds for potential effects 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; (3) if operator workaround problems were captured in the licensee's corrective action program; and (4) cumulative effects of the workarounds on reliability, availability, and potential for system mis-operation. The operator workarounds reviewed included the following:

- MO 2200003713 Relief Valve 2RV5205
- MO 2200002405 Motor-Operated Valve 1-CVC-514
- MO 1200002473 Letdown Control Valve 1-CVC-110
- MO 1200003781 Level Switch 1-LS-1527
- MO 2199905255 2XL2C67L/DS7
- MO 2199902442 Valve 2-CV-631
- MO 2200004146 Control Element Drive Mechanism Panel Cooling Fan
- MO 2200003777 Unit 2 Loose Part
- MO 2199902156 Main Steam Seal Regulator 2-TGS-4681

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test procedures and associated testing activities for selected risk significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; and, (6) that equipment was returned to the status required to perform its safety function. The following maintenance activities were reviewed:

- MO1200001824 1B EDG Air Flask 1RV-4835 Replacement.
- MO1200003673 1B EDG local meter calibrations.
- MO 2200003211 Saltwater basket strainer to emergency core cooling system room air cooler.
- MO 2200001202 Perform procedure EDG-11 to inspect the air start check valves on the 2B EDG.
- MO 2200000061 Increase crankcase vacuum on 2B diesel generator by enlarging the orifice size in the ejector per ES No. 199901142-000.
- MO 2200001602 Perform inspection of 2B EDG rotor and stator and perform insulation resistance testing.
- MO 2200002040 Calibrate 2B EDG pressure instruments.
- MO 2200000469 Perform ETP 92-95R leak test on 2B-DSA-110.
- MO 2200002306 Perform calibration of 2B EDG local meters per FTE-56.
- MO 2200002308 2B diesel generator protective relay testing per FTE-59.
- MO 2200000503 Replace the lube oil heat exchanger vent tubing (copper and brass fittings) with stainless steel tubing and fittings per TUBE-01.
- MO 2200002031 Perform alarm/setpoint procedure on 2B EDG jacket cooling water temperature switches.
- MO 2200002032 Perform alarm/setpoint procedure on 2B EDG temperature instruments.
- MO 2200001476 EDG 2B Bus 24 feeder breaker inspection per FTE-51.

The inspectors reviewed the following associated maintenance documentation:

- EDG-11, Cleaning and Inspection of Air Start Distributer and Check Valves.
- FTE-56, Periodic calibration of switchboard, panel, integrating, graphic meters and associated meter equipment.
- FTE-59, Periodic maintenance, calibration, and functional testing of protective relays.
- TUBE-01, Fabrication and installation of Parker CPI and Swagelok compression fittings.

- FTE-51, 4KV General Electric Magne-Blast Circuit Breaker and Cubicle Inspection.

The inspectors reviewed the following post-maintenance tests conducted to verify appropriate system restoration and operability, following the completion of the above stated maintenance activities:

- STP O-8B-2 Test of 2B Emergency Diesel Generator and 4KV Bus 24 LOCI Sequencer, Sections 6.1, 6.3, 6.5, and 6.6.
- STP O-8B-1 1B EDG Fast Speed Start Test
- OI-29 Saltwater System

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Emergency Diesel Generator Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk-significant systems, structures, and components (SSCs) to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report, Technical Requirements Manual, and licensee procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- STP O-08B-1 1B Diesel Generator Fast Speed Start Test
- STP O-08B-2 Test of 2B Diesel Generator and 4KV Bus 24 LOCI Sequencer, Sections 6.1, 6.3, 6.5, and 6.6.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 05000317 and 05000318/200008-01: Verification of Safety Injection Tank Boron Concentration

a. Inspection Scope

The inspectors reviewed the licensee's 10 CFR 50.59 evaluation involving an alternate method to satisfy the technical specification monthly surveillance requirement for verifying safety injection tank (SIT) boron concentration. The inspectors evaluated whether the alternate method, monitoring for a 10 inch SIT level increase, was sufficient to satisfy technical specifications.

b. Findings

Technical Specification (TS) Surveillance Requirement 3.5.4.1, requires the licensee to "Verify boron concentration in each SIT is ≥2300 and ≤2700 ppm," every 31 days. Prior to September 2000, the licensee satisfied this requirement by sampling. The containment entries to draw these samples result in an annual site-wide dose of about 1 rem/year. For ALARA reasons, the licensee prepared an engineering evaluation to change the UFSAR and the TS basis to interpret the TS wording to allow them to "verify" SIT boron concentration by either (1) sampling or (2) monitoring SIT in-leakage. Surveillance Test Procedure (STP) O-107-1, Safety Injection Tank Boron Verification, specifies that SIT samples are required on startup from cold shutdown and every 180 days. Per the change, the 31-day TS surveillance is satisfied by either sampling the SITs or by in-leakage monitoring. In-leakage monitoring can only be used if the cumulative SIT level increases since the last sample are less than ten inches. Licensee calculations show that 12 inches of in-leakage of pure water would be needed to reduce boron concentration below the TS limit.

The inspector reviewed the licensee's engineering evaluation and identified that monitoring for a 10-inch SIT level increase does not necessarily satisfy TS, due to the possibility of unmeasurable, simultaneous in-leakage and out-leakage diluting boron concentration below the TS limit. The inspector observed that out-leakage from the SITs to the reactor coolant drain tank is common. For example, the net out-leakage from each of the four Unit 1 SITs is currently 20 to 40 gallons per day. The dilution of SIT boron concentration could occur if reactor coolant system water were to leak past two check valves into the SIT. Inspector review of actual SIT level data over the last 10 years shows significant in-leakage from the RCS into the SITs, particularly prior to 1997. This in-leakage was typically attributed to check valve seat leakage. Additionally, the inspector calculated that a very small in-leakage rate of 0.01 gallons per minute would be sufficient to result in a 10-inch (400 gallons) increase in SIT level in one month.

Consequently, the inspector concluded that the alternative in-leakage monitoring method does not provide sufficient information to determine if a potential boron concentration dilution has occurred. In addition, it would not be able to determine (or even bound) the extent of the dilution. Accordingly, in-leakage monitoring does not provide sufficient information to satisfy TS 3.5.4.1. The inspector discussed this issue with the licensee. The licensee agreed that the data and engineering evaluation did not support the alternative method and restored their previous monthly sampling method. This issue was entered it into the licensee's corrective action system.

The failure of the licensee to adequately verify SIT boron concentration is a violation of TS 3.5.4.1. This violation was considered more than minor because it has a credible impact on reactor safety due to the possibility of a significant dilution in boron concentration going undetected for an extended period. The issue was assessed using the significance determination process and was found to be of very low safety significance (Green) because subsequent sampling of each SIT showed boron concentrations were within the TS required band. Due to the overall low risk significance, this violation of TS 3.5.4.1 is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000317; 318/2000-012-02)**

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed selected risk significant temporary modifications to assess: (1) the adequacy of the 10 CFR 50.59 evaluations; (2) that the installations were consistent with the modification documentation; (3) that drawings and procedures were updated as applicable; and, (4) the adequacy of the post-installation testing. The following temporary alterations were inspected:

- Temporary Alteration No. 1-00-0081: Disable Channel “A” Reactor Vessel Level Monitoring System Alarms due to Failed Probe.
- Temporary Alteration No. 2-00-0045: 21 Emergency Core Cooling System Pump Room Air Cooler Service Water Inlet Relief Valve, 2RV5205, Leakage.

b. Findings

No findings of significance were identified.

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training January 29, 2001. The inspectors verified that emergency classification declarations and notification activities were properly completed.

b. Findings

No findings of significance were identified.

3PP4 Security Plan Changes

a. Inspection Scope

An in-office review was conducted of changes to the Contingency Plan, identified as Revision 14 and submitted to the NRC on June 29, 2000. The inspector confirmed that the changes were made in accordance with 10 CFR 50.54(p) and did not decrease the effectiveness of the Plan.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

40A1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data for the below listed cornerstones to verify individual PI accuracy and completeness. This inspection examined data and plant records including a review of PI Data Summary Reports.

- Reactor Coolant System Leakage
- Auxiliary Feedwater (AFW) System Unavailability

b. Findings

The inspectors evaluated the licensee's determination that the AFW system was considered available while performing surveillance procedure STP O-005A-1, Auxiliary Feedwater System Quarterly Surveillance Test. The inspectors found that the licensee had been inappropriately crediting operator action to reset the trip/throttle valve to recover the pump, in that this action does not meet the performance indicator guidance of being an uncomplicated, simple action, that is virtually certain to be successful. The licensee's review of completed surveillances, back to January 2000, showed that the amount of time the AFW system trip/throttle valve remained in a tripped condition ranged from 31 to 85 minutes, with an average time of 56 minutes. The licensee intends to correct the values in the next PI submittal and has entered this item into their corrective action program (Issue Report IR3-041-489). The additional AFW unavailability hours associated with this finding would not have caused previously submitted data to cross a Green/White threshold.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report 50-317/2000-006-00 & 01: Number 12 Containment Spray Pump Circuit Breaker Failed to Close

The NRC inspection of this Licensee Event Report (LER) is discussed in Section 1R12 of this report. This LER is closed.

4OA5 Other

.1 (Closed) Unresolved Item 05000317, 05000318/199908-001: Operation of Emergency Diesel Generator 1A at Low Loads

During a 1999 engineering team inspection, the NRC determined that emergency diesel generator (EDG) 1A, newly installed at the plant, might operate with a load that was below the 30% minimum recommended by the manufacturer. Institute of Electrical and Electronics Engineers (IEEE) Standard 387-1984, requires that the, “light or no load capability... [of the engine] shall be demonstrated by test.” At the time of the team inspection, the licensee could not produce documentation supporting qualification of the 1A EDG to IEEE 387-1984 requirements.

To address the inspectors questions regarding the light or no load capability of the EDG, the licensee initiated appropriate performance testing. Specifically, after an initial operation of the engine at varying load, the licensee lowered the engine load to approximately 3% of its rating and ran it at that load for 168 consecutive hours. As recommended by the IEEE standard, after a period of operation at 50% and 100% loads, the EDG was stopped and its condition evaluated. The test results were documented in two MPR Associates documents, MPR-2146, “SACM Model UD45 Diesel Engine Low-Load Test Report,” dated September 2000, and MPR-2181, “BGE Low-Load test SACM Model UD45 Engine Signature Analysis,” dated December 2000. The licensee informed the NRC of the test results and their conclusions in a letter dated December 20, 2000.

Based on the inspector’s review of the above documents and discussions with responsible licensee personnel regarding the test results and the bases for the selected loading and test duration, the inspector concluded that the licensee had appropriately demonstrated the qualification and capability of the 1A EDG to operate at the anticipated plant design loads. Based upon the test results, the inspector also concluded that no violation of NRC requirements occurred. This unresolved item is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on February 23, 2001. 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

C. Cruse, Vice President
D. Holm, Superintendent, Nuclear Operations
P. Katz, Plant General Manager
B. Montgomery, Director, Nuclear Regulatory Matters
M. Navin, Superintendent, Technical Support
K. Nietmann, Manager, Nuclear Performance Assessment Department
T. Pritchett, Manager, Nuclear Engineering Department
J. Spina, Superintendent, Work Management
L. Wechbaugh, Superintendent, Nuclear Maintenance

ITEMS OPENED AND CLOSED

Opened and Closed

05000317; 318/2000-012-01	NCV	Inadequate Design Control Associated with 4.1 kV Breaker Replacement (Section 1R.12.2)
05000317; 318/2000-012-02	NCV	Failure to Verify Safety Injection Tank Boron Concentration (Section 1R.22)

Closed

05000317/2000-006-00 & 01	LER	12 Containment Spray Pump Circuit Breaker Failed to Close (Section 4OA3)
05000317, 05000318/1999-08-001	URI	Operation of Emergency Diesel Generator 1A at Low Loads (Section 4OA5)

LIST OF ACRONYMS USED

ABB	Asea Brown Boveri
AC	Alternating Current
AFW	Auxiliary Feedwater
CCNPPI	Calvert Cliffs Nuclear Power Plant, Inc.
CFR	Code of Federal Regulations
DC	Direct Current
ECCS	Emergency Core Cooling Systems
EDG	Emergency Diesel Generator
GE	General Electric
HPSI	High Pressure Safety Injection
IEEE	Institute of Electrical and Electronics Engineers
IPE	Individual Plant Examination
IR	Inspection Report
LER	Licensee Event Report
LOIT	Licensed Operator Initial Training
LORT	Licensed Operator Requalification Training
MO	Maintenance Order
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
SDP	Significance Determination Process
SIAS	safety injection actuation signal
SIT	safety injection tank
SSC	Structure, System and Component
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) 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 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	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

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 little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to affect safety and would require the NRC to take additional actions. RED findings represent an unacceptable loss of safety margin and would result in the NRC taking significant actions that could include ordering the plant shut down.

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 incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. The color for an indicator corresponds to levels of performance that may result in increased NRC oversight (WHITE), performance that results in definitive, required action by the NRC (YELLOW), and performance that is unacceptable but still provides adequate protection to public health and safety (RED). GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections.

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. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, as described in the matrix. 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.