

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report: 50-317/86-03
50-318/86-03

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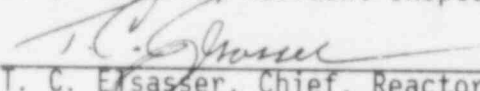
Licensee: Baltimore Gas and Electric Company

Facility: Calvert Cliffs Nuclear Power Plant, Units 1 and 2

Inspection At: Lusby, Maryland

Dates: January 20 - March 3, 1986

Inspectors: T. Foley, Senior Resident Inspector
D. C. Trimble, Resident Inspector

Approved: 
T. C. Eysasser, Chief, Reactor Projects Section 3C

3/19/86
Date

Summary: January 20-March 3, 1986: Inspection Report 50-317/86-03, 50-318/86-03

Areas Inspected: Routine resident inspection of the Control Room, accessible parts of plant structures, plant operations, radiation protection, physical security, fire protection, plant operating records, maintenance, surveillance, radioactive effluent sampling program, and reports to the NRC. Inspection hours totalled 136 hours.

Results: A significant weakness was shown in the licensee's administrative controls over the opening of a Containment isolation valve (Section 4). This is the subject of a violation.

The inspectors reviewed a licensee identified violation regarding the discovery of a check valve in the instrument air system which had been installed backwards (Section 12).

The cause of a Unit 2 Reactor trip on February 4 was not identified, raising the potential for trip recurrence (Section 5).

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DETAILS

1. Persons Contacted

Within this report period, interviews and discussions were conducted with various licensee personnel, including reactor operators, maintenance and surveillance technicians and the licensee's management staff.

2. Summary of Facility Activities

Unit 1 was operating at power at the beginning of the inspection period. On January 23, 1986, the Reactor tripped from full power due to the malfunctioning of a Reactor trip breaker during surveillance testing (see Section 5 for details). The unit returned to power on January 24 and remained at power through the end of the period. On January 24, a Containment isolation valve (for sampling system return to the Unit 1 Reactor Coolant Drain Tank) was inadvertently left open without proper administrative control (see Section 4). A small unplanned release of gaseous activity occurred on February 4 (see Section 10 for details).

Unit 2 also operated at full power for the major portion of the inspection period. On February 1, power was reduced to 10% to allow personnel access into Containment to add oil to #21A Reactor Coolant Pump. On February 4 the Reactor tripped. The cause of the trip could not be identified, and the unit returned to power on February 5, 1986 (see Section 5).

3. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (318/83-13-01) Pressurizer Level Falls Below Low Level Allowed by Technical Specifications Due to Reactor Coolant Temperature Swings at Low Power Levels. This problem was resolved by raising the automatic pressurizer level setpoint for low power operation from 144 inches to 160 inches. This new value is within the limits allowed by Technical Specifications. This item is closed.

(Closed) Inspector Follow Item (318/82-27-02) Licensee to Complete An Evaluation of Fusing Requirements for Inverter Supplied Vital 120 VAC Branch Circuits and Perform Any Necessary Fuse Modifications. The above evaluation was performed and modifications accomplished under completed Facility Change Request FCR 83-1001. This item is closed.

(Closed) Unresolved Item (318/82-27-03) Licensee to Evaluate Need for Inverter Current Limiting Devices and, If Appropriate, Remove Them. The licensee completed this evaluation and concluded these devices were not necessary. The current limiters have been removed. This item is closed.

(Closed) Inspector Follow Item (318/84-01-03) Slide Gates in Salt Water System Need to be Controlled When Out of Normal Position Under Locked Valve Deviation Control Program. The inspector reviewed Calvert Cliffs Instruction CCI 309A, dated September 27, 1985 ("Locked Valves") and the salt water system operating

instruction and confirmed that the slide gates are locked valves. As such, they are tracked, when out of their normal position, by the locked valve deviation program. This item is closed.

(Closed) Violation (317/82-07-10) Operation Without Operable Hydrogen Analyzer Due to Manual Sample Line Isolation Valves (inside Containment) Not Being Properly Repositioned Following Penetration Local Leak Rate Testing (LLRT). The surveillance test procedure for LLRT (STP M571-1, Revision 6, for Unit 1 and STP M571-2, Revision 13, for Unit 2) now requires a double verification by operations personnel for proper alignment of valves following LLRT's. The inspector reviewed STP M571-1 and confirmed that the positions of the specific valves that were improperly left shut at the time of the event (1-PS-1003 through 1-PS-1006) are required to be checked. Their positions are to be in accordance with the position specified in the governing Operation Instruction (OI-31B, Revision 8) valve lineup sheets. The inspector confirmed that those positions are listed as "locked open". Two added measures of assurance also exist. Valves required to be locked in a specified position for an existing plant operating mode are also tracked by means of a locked valve deviation log (Calvert Cliffs Instruction 309) when they are moved from their required position. Second, prior to plant startup following refueling outages, the time when LLRT's are performed, system lineup checks are performed on systems which have been altered/or realigned during the outage. The general policy existing is that the lineups for all plant systems are checked following these outages.

The above measures appear adequate to minimize the chance of event recurrence. This item is closed.

4. Review of Plant Operations

a. Daily Inspection

During routine facility tours, the following were checked: manning, access control, adherence to procedures and LCO's instrumentation, recorder traces, protective systems, control rod positions, Containment temperature and pressure, control room annunciators, radiation monitors, radiation monitoring, emergency power source operability, control room logs, shift supervisor logs, tagout logs, and operating orders.

No violations were identified.

b. System Alignment Inspection

Operating confirmation was made of selected piping system trains. Accessible valve positions and status were examined. Power supply and breaker alignment was checked. Visual inspection of major components as performed. Operability of instruments essential to system performance was assessed. The following system was checked:

- Plant Fire Protection Systems and Equipment during period of January 28-31, 1986.
- Unit 1 Boric Acid Injection System during period of January 27-30, 1986.
- Unit 2 Compressed Air (Salt Water Air Compressor loads) checked on February 19, 1986.

Boron Injection Subsystem

Inspection Scope

The Boron Injection Subsystem (BIS) of the Chemical and Volume Control System (CVCS) is important to safety in that it delivers concentrated boric acid to the suction side of charging pumps upon a safety injection actuation signal (SIAS). Flow from the BIS is credited in the small break loss-of-coolant accident (LOCA) calculations for Calvert Cliffs Units 1 and 2. The BIS also provides a backup shutdown capability.

During the period from January 27 to January 31, 1986 the Calvert Cliffs Unit 1 BIS was inspected in the following areas:

- System Documentation
- Physical Inspection
- Control Room Indications
- Maintenance
- Compliance with the Technical Specifications
- Control of Consumables (Boric Acid)
- Implications of Recent Probabilistic Risk Assessment.

Following the inspection, a meeting was held with plant management to discuss the findings.

System Documentation

The following documents were reviewed for clarity and detail: Calvert Cliffs FSAR, "Chemical and Volume Control System", System Description No. 6, September 1982, and Drawing OM-73, Rev. 26, "Chemical and Volume Control System".

The description of the BIS in the FSAR and system description were adequate and provided good detail on system design and operation. These documents should be updated to reflect that the BIS is credited for flow during a small break LOCA. The system drawings, OM-73, were well prepared using accepted industry symbology. System drawing interfaces were clearly marked and the use of a macro diagram on Sheet 1 of 3 made the overall system function very clear. Two minor problems, lack of several alarm panel locations and lack of a "SIAS STARTS" indication for the charging pumps, were referred to the licensee for corrective action. Prior to completion of the inspection, corrective action was verified.

Physical Inspection

Under SIAS conditions, the charging pumps discharge to either the CVCS charging line (primary path) or the high pressure safety injection header (secondary path).

The BIS was inspected to ascertain general equipment condition and housekeeping. The BIS pumps, valves, piping and supports appeared to be in good condition. Some pipe insulation in the boric acid pump room was found to be damaged or missing. Component insulation is important to assure that the redundant heat tracing, provided to prevent boric acid precipitation in the system, maintains its effectiveness. The licensee has corrected this situation. In addition, the housekeeping situation in the boric acid pump rooms was improved substantially during the course of the inspection. It appears that accumulated boric acid crystals on floors and components in the boric acid pump rooms is the most significant housekeeping problem associated with the BIS and requires continued attention. Several indicator light bulbs on heat tracing panels inside the Unit 1 boric acid pump room were found to be inoperable and were promptly replaced by the licensee. The housekeeping adjacent to the boric acid batching tank was satisfactory and housekeeping in the charging pump rooms was very good. An excessive quantity of lube oil was noted under the charging pumps; the licensee has indicated that this situation will be corrected. The following electrical cabinets were opened by the licensee for inspection: (1) Charging Pump Breakers 52-1404, 52-1415, 52-1115, and (2) Motor Control Centers 52-11430, 52-11424, 52-11406, 52-10425, 52-10406. The above circuit breakers and motor control centers represent the principal BIS electrical components that actuate on SIAS. The inside area of these cabinets were in very good condition and appear to have benefited from preventive maintenance in the form of periodic cleaning. These electrical cabinets were notably free of foreign material; the electrical connections appeared clean and tight.

The BIS valve alignment was checked against the lineup contained in Attachment (1) of Procedure OI-2C, "Concentrated Boric Acid System Operations", Revision 8. Proper valve alignment was verified to the extent possible. In the process of checking valve alignment, it was noted that a number of equipment tags were missing. This situation was referred to the licensee for corrective action.

Control Room Indications

Control panels for the Unit 1 BIS appeared to be very clean and all controls and indications were clearly labeled. The display on panel 1C07 benefits from a mimic which clearly shows the flow path.

The control room alarm manual was reviewed and found to have clear instructions to operators for dealing with off-normal conditions. One error in the alarm manual was noted. Window F-35 on panel 1C07, "12 BA

Tank Level Low", is described in the alarm manual as initiated by device "1-LA-08". The correct device should be "1-LA-208". This matter was referred to the licensee and was promptly corrected.

Maintenance

Activities associated with preventative and corrective maintenance for the BIS were reviewed. All instruments and alarms which provide system states for the BIS (tank levels, temperatures, pressure, etc.) are the subject of preventive maintenance/calibration. Mechanical components (pumps and valve motors) and electrical components (heat tracing, circuit breakers, and load centers) are also in the preventive maintenance program. Preventive maintenance card EIN:201-41-ILIT/LIA/LA206 was found to be in error. Procedure Step C identified "1-LA-106" which should be "1-LA-206". This item was referred to the licensee for corrective action. A review of all outstanding maintenance for BIS (System 41) was undertaken. The licensee provided a computer printout of all outstanding maintenance requests. This printout showed a total of 29 mechanical, 5 instrumentation/control and 1 electrical maintenance requests outstanding for the Unit 1 BIS. All requests had been identified within the past year with most being outstanding for less than six months. Two "work packages", detailed directions for performing maintenance, were reviewed and found to contain good documentation (drawings, procedures, etc.) and were judged to be adequate for performance of the requested maintenance.

In conclusion, the preventive and identified maintenance system for the BIS appears to be operating efficiently. All major operating components receive preventive maintenance and identified maintenance appears to be undertaken within a reasonable time frame.

Compliance with Technical Specifications-Surveillance

The BIS is the subject of Technical Specification (TS 3/4.1.2, "Boration Systems") surveillance requirements to assure operation of pumps and valves on a SIAS signal, valve alignment, heat tracing operability, proper level and boron concentration in borated water sources, and pump readiness testing in accordance with the ASME Boiler and Pressure Vessel Code.

Surveillance test procedures and control room logs were reviewed to assure that the tests and observations were being performed in compliance with the requirements of TS 3/4.1.2. In addition, completed test procedure and control room logs were reviewed to assure proper test and observation frequencies. The inspectors concluded that the licensee is in compliance with TS 3/4.1.2 for testing of the Unit 1 BIS.

Control of Consumables (Boric Acid)

Large quantities of dry, boric acid are located on site for use in the BIS. The procurement and storage of this material was reviewed against commitments provided by BG&E. The following documents, which provided the basis for the review, are referenced in Section IB.2 of the Calvert Cliffs FSAR:

- Regulatory Guide 1.38 - QA Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, 05/77). This endorses ANSI N45.2.2 (12/20/72).
- ANSI N45.2.13 - QA Requirements for Control of Procurement of Equipment, Materials, and Services for Nuclear Power Plants; Draft 2, Revision 2, (10/73).

Although boric acid was not originally categorized as a "safety-related" item, the controls described in Section 6.11 of BG&E's Procurement and Storage Manual were and are in compliance with ANSI N45.2.2 and the FSAR.

The revised manual, dated 8/22/85, identified boric acid as a "safety-related" item, thus incorporating additional specific controls. The controls include independent laboratory testing, shipment inspection by an Operations Quality Assurance Section (OQAS) Inspector, and use of a segregated receipt inspection area until receipt inspection is completed. Copies of all test procedures used to perform required analyses, and purchase orders will be sent to the Supervisor, Quality Control, Surveillance/Receipt Inspection Unit.

On January 30, 1986, an inspection of the boric acid storage area was performed. During the inspection, it was noted that several of the storage bags were broken and sizable amounts of granular boric acid had leaked out the warehouse floor. The boric acid in storage was marked non-safety related because it had been purchased before the new manual classification. Therefore, an audit of chemistry records was performed. The audit showed that adequate steps had been taken to comply with the new manual requirements.

During further NRC inspection of the warehouse, it was observed that the boric acid was stored next to similarly packaged bags of urea. Because of the similarities of the packaging and the materials themselves, it is recommended that BG&E segregate the storage of these two materials.

In conclusion BG&E appears to be in compliance with regard to commitments for procurement and handling of dry, boric acid, at Calvert Cliffs. Future procurement of boric acid as "safety-related" and improvement of warehouse conditions will further improve this program.

Implications of PRA

A probabilistic risk assessment (Interim Reliability Evaluation Program: Analysis of Calvert Cliffs Unit 1 Nuclear Power Plant, NUREG/CR-3511) was used during the inspection of the BIS to identify critical components whose failure provide a significant contribution to the core melt probability. This effort was largely unsuccessful for the following reasons:

- The BIS, operated automatically on SIAS, is not credited for any significant accidents reflected in the PRA. Specifically, the small break LOCA for which BIS flow is credited, falls below the PRA cut-off at a probability of $7.8E-6$. Two other PRA sequences, T₄-173 and T₃-139, involve the failure to scram and the failure of ⁴BIS, in these cases, the BIS would be manually initiated. Analyses show that these sequences are dominated by failure of the reactor operator to initiate BIS rather than system failures.

One element of the PRA can be updated based upon this inspection of the BIS. Section 6.11.3 of NUREG/CR-3511 states that credit for use of the gravity feed lines (lines bypassing the boric acid pumps) was not provided since the associated MOVs-508 and 509 are not referenced in the emergency boration procedure EOP-13. A new procedure, AOP-1C, "Emergency Boration", was reviewed and found to contain instructions for use of the gravity feed lines. This procedure was discussed with several reactor operators and it appears that AOP-1C is well understood with regard to the gravity feed mode of the BIS.

Conclusion

The BIS appears to be in good condition and could be expected to fulfill its safety function on demand. The maintenance programs are efficiently run and encompass all key components of the BIS. The licensee is in compliance with applicable Technical Specification.

The licensee should continue its efforts in housekeeping in that use of concentrated boric acid represents chronic a problem with regard to minor leakage. In addition, the licensee should maintain vigilance with regard to the integrity of system insulation.

No violations were identified.

c. Biweekly Inspections

During plant tours, the inspector observed shift turnovers; boric acid tank samples and tank levels were compared to the Technical Specifications; and the use of radiation work permits and Health Physics procedures were reviewed. Area radiation and air monitor use and operational status was reviewed. Plant housekeeping and cleanliness were evaluated. Verification of the following tagouts indicated the action was properly conducted.

- Tagout 15770, Unit 2 SIT Isolation Valves checked on February 24, 1986.
- Tagout 15171, Unit 2 H₂ Purge System checked on February 24, 1986.

No violations were identified.

d. Other Inspections

Improperly Controlled Containment Isolation Valve

On January 24, 1986, the chemistry group checked out a key, administratively controlled by the operations group, for Containment isolation valve 1-PS-6529SV. This valve is allowed to be opened on an intermittent basis under administrative control in accordance with Technical Specification (TS) 3.6.4.1 and is located in the return line from the primary sample sink to the Unit 1 Reactor Coolant Drain Tank (RCDT). The chemistry group then conducted hands on training on use of the newly designed Post Accident Sampling System (PASS). At the same time, they were verifying the accuracy of the draft procedure for this system. The training was stopped near the end of the day shift. The chemists involved, believing that training would be continued on the next shift, neglected to finish the procedure and, as a result, left 1-PS-6529SV open without informing operations. The day shift Unit 1 Control Room Operator (CRO) had logged the fact that 1-PS-6529-SV was opened for chemistry at 9:55 a.m. The CRO failed, however, to pass this information on to the relieving CRO, and that second CRO did not notice the log entry concerning 1-PS-6529-SV during his log review. This was contrary to the requirements of General Supervisor, Operations (GSO) Instruction 82-6, "Logging Containment Isolation Valves", which states that "All containment isolation valves opened under administrative control per Technical Specification (TS) 3.6.4.1 shall be logged in the Control Room Operators log. These entries shall be carried over from the previous shift until the applicable valves have been returned to their normal positions."

About 3:00 a.m. on January 25, 1986 a Unit 2 Reactor Coolant System (RCS) leak rate determination showed higher than expected leakage. This caused the operations group to check the sample system lineup as part of an overall search for possible RCS leakage sources. It was then noted that 1-PS-6529-SV was open. In fact, the sample system was lined up in such a manner that it was allowing water to pass from the Unit 2 RCS to the Unit 1 RCDT. Valve 1-PS-6529-SV was then closed.

In Inspection Report 317/84-01; 318/84-01 (January, 1984), the licensee was cited for allowing 1-PS-6529-SV to be open without proper administrative controls. However, in that case the root cause was a false indication of valve position due to an improper switching of indicator light red (open indication)/green (closed indication) lens covers. At the same time two other Containment isolation valves, associated with the sample system and not allowed to be opened by TS, were opened for

a short period of time by chemistry personnel because they were doing an evolution at power which they had always previously done while shut-down. They did not realize that TS's did not allow those valves to be opened at power. While the root causes of the January 1984 events differ from the apparent causes of the present situation (i.e., chemistry technician failure to follow procedure and poor operator information turnover), one of the corrective actions from the January, 1984 events was intended to, and should have, prevented the present occurrence. Specifically, the above described GSO standing instruction was implemented as a corrective action to the January 1984 events. If the instruction had been properly followed through the shift changes, administrative control over the valve would not have been lost. Failure to maintain proper administrative control over 1-PS-6529-SV is a violation (317/86-03-01).

The licensee has taken the following corrective actions: (1) the responsible chemist was counseled; (2) the event was discussed with chemistry personnel; (3) a personnel incident report of the event was reviewed by and discussed with operations personnel; (4) reinforcing the GSO standing instructions, operations personnel were instructed to carry forward log entries regarding open containment isolation valves between shifts to serve as a reminder of valve status; and (5) to increase personnel responsibility/cognizance for valve control keys, keys are now being signed out to individuals (by name) instead of to organizational groups (e.g. "Chemistry Group"). Additionally, the licensee will add manual isolation valves to the sample return lines to the Volume Control Tanks (VCT). These valves will be accessible during all expected radiological conditions and will assure those flow paths can always be isolated. Sample returns will then normally be returned to the VCT's, and the PS-6529 valves will be kept shut. Minimizing the number of times the PS-6529 valves are opened will reduce the probability of recurrence of this event. This modification was scheduled for accomplishment during the week of March 10, 1986.

Because licensee corrective action appears to be appropriate and satisfactory, a written response to the above violation will not be required.

Auxiliary Feedwater Pump Journal Bearings

On January 29, 1986 operations personnel noted significant temperature increases on the turbine journal bearings for #11 Auxiliary Feedwater (AFW) pump. The pump was lined up for standby operation; however, a small amount of steam was leaking by the automatic (normally closed) steam supply valves (there are two supply valves), causing the pump to be rotating at a slow speed. The pump was removed from service and the bearings inspected. Both bearings had "wiped" and were replaced. The inspector discussed this event with the Manager, Nuclear Operations (MNO) and the Manager, Nuclear Maintenance (MNM) and pointed out that at slow rotational speeds a condition of imperfect or inadequate bearing lubrication may exist. (This condition could be set up by an insufficient

supply of oil to the bearing due to inadequate movement of the oil rings and/or inability to build up adequate fluid film lubrication due to the low speed. Reference Machine Design by P. Black and O. Adams, Jr., 1968, Chapter 21). The MNO stated he also suspected a possible inadequate lubrication problem. The pump was repaired and restored to the standby lineup. Again, it coasted at slow speed due to steam leakage and again bearing temperatures began to rise. A decision was made to increase the rotational speed by admitting more steam via a bypass valve around one of the automatic isolation valves. When pump speed was increased to about 800 RPM, bearing temperatures reduced to satisfactory levels. A change was initiated to operational instructions to maintain at least 800 RPM on AFW pumps coasting due to steam leakage.

Additionally, the bearings on #12 AFW pump were inspected and found satisfactory. Oil levels in all AFW pump bearings were checked.

Plant Operations and Safety Review Committee (POSRC)

The inspector attended a regular meeting of the POSRC on January 29, 1986. Technical Specification 6.5 membership and quorum requirements were met. Topics discussed included (1) through wall seepage discovered in one area of the salt water system piping (see Section 8 for further details), (2) low range pressurizer pressure transmitter drift problems, (3) inspection frequency for underground salt water system piping, (4) use of a temporary thermal relief in the High Pressure Safety Injection System, and (5) several procedural changes.

No unacceptable conditions were identified.

5. Events Requiring NRC Notification

The circumstances surrounding the following event requiring prompt NRC notification pursuant to 10CFR50.72 were reviewed. For events resulting in a plant trip, the inspectors reviewed plant parameters, chart recorders, logs, computer printouts and discussed the event with cognizant licensee personnel to ascertain that the cause of the event had been thoroughly investigated for root cause identification.

Unit 2 Reactor Trip

At 10:57 a.m. on February 4, 1986 Unit 2 tripped from 100% power. Plant parameters were quickly stabilized and safety systems functioned properly following the trip. The cause of the trip could not be identified, in part due to a failure of the computer sequence of events recorder. The licensee suspected a spurious high steam generator level signal led to a turbine and then reactor trip. The Channel C level transmitter for each steam generator (2-LIC-1113C and 2-LIC-1123C) had been deenergized for maintenance which provided one signal to a 2 out of 4 trip logic for high steam generator level trip (this is a non-safety related trip for plant equipment protection). The plant trip occurred about two minutes later. The inspector noted that recently an in-

intermittent ground (which had not been located) had occurred on the Channel D circuitry. He surmised that possibly this ground could have recurred and caused a second input signal to the 2 out of 4 logic. He informed the licensee of this possibility. They agreed to investigate this possibility. The licensee interviewed the technicians involved and an operator who had gone to the Engineered Safety Features (ESF) sensor and actuation cabinets after Channel C was deenergized and a resulting annunciator alarm (showing one signal to the 2 out of 4 logic) was received. No personnel error was apparent. The licensee attempted to duplicate the event, but this effort was unsuccessful. A post trip review was conducted and a POSRC meeting held. The inspector attended that meeting. Technicians reported that the sequence of events recorder had been tested and was not working properly (reason for its failure during the trip was unknown). No ground was currently evident on Channel D circuitry. A decision was made to perform additional testing prior to start up and, if no problems were found, restart the unit. The additional testing involved (1) deenergizing the Channel C circuitry again and checking for proper ESF cabinet operation, and (2) running an operational test of the high steam generator level trip circuitry. The POSRC also decided that, following start up, (1) on a once shift basis the ESF cabinets should be checked for indication of a partial trip signal (lock-in lights) to the high steam generator level circuitry, (2) further evaluation should be done by February 7 regarding whether an intermittent ground on Channel D could have caused the second input signal for unit trip and (3) restrictions should be placed on maintenance on the steam generator level circuitry until the ground evaluation was completed. The unit was restarted at 4:25 a.m. on February 5. Additionally, the inspector noted that the technicians involved had in their possession at the time of trip maintenance orders (approved for work by two different shift supervisors on two different days) for both the C and D channels. He expressed concern to the General Supervisor, Operations (GSO) that it may not be a good practice to allow concurrent authorizations for corrective maintenance on 2 separate channels (which provide inputs for a unit trip). The GSO acknowledged the inspectors concern. As of the end of the inspection period no system grounds had been found.

Reactor Trip on Unit 1 Due to Reactor Trip Breaker Problem

At 10:56 a.m. on January 23, 1986, Unit 1 tripped from 100% power due to the malfunctioning of a Reactor Trip Breaker (RTB) during surveillance testing. All safety systems functioned as designed, and plant conditions were quickly stabilized. The testing involved cycling the RTB's, as part of Surveillance Test Procedure STP 210B-1, Revision 25, Section VIII, RPS Logic Matrix Test.

After cycling for test, RTB2 did not fully close due to an internal mechanical (spring) failure. The failure was in the safe direction and did not prevent breaker opening. The control element assemblies (CEA's) were unaffected because power was still being supplied to them through other breakers which were to be subsequently tested. When the other breakers were then cycled open, power was lost to the CEA's. This resulted in a Reactor and Turbine trip. The failed spring's function is to hold the main contactors firmly together (for good electrical contact) while the breaker is closed. One end of the

spring had slipped off of its anchor point. Licensee personnel recalled seeing a similar problem in a breaker that had been received back from the vendor after modification. The licensee sent RTB2 back to the vendor for disassembly and inspection for root cause. The remaining installed RTB's were checked to ensure springs were properly attached to their anchor points. No further problems were identified. Licensee investigation of the root cause of failure will be followed by the NRC (IFI 317/86-03-02).

No violations were identified.

6. Observation of Physical Security

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, vehicle searches and personnel identification, access control, badging, and compensatory measures when required.

No violations were identified.

7. Review of Licensee Event Reports (LERs)

LERs submitted to NRC:RI were reviewed to verify that the details were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspector determined whether further information was required from the licensee, whether generic implications were indicated, and whether the event warranted on site followup. The following LER's were reviewed.

<u>LER No.</u>	<u>Event Date</u>	<u>Report Date</u>	<u>Subject</u>
<u>Unit 1</u>			
86-01	01/23/86	02/14/86	Reactor Trip Caused by Failure of TCB #2 During Surveillance Testing
<u>Unit 2</u>			
85-12	12/12/85	01/10/86	Reactor Trip on Low Steam Generator Water Level

8. Plant Maintenance

The inspector observed and reviewed maintenance and problem investigation activities to verify compliance with regulations, administrative and maintenance procedures, codes and standards, proper QA/QC involvement, safety tag use, equipment alignment, jumper use, fire protection, retest requirements, and reportability per Technical Specifications. The following activities were included.

- MO 205-246-186C, Install rebuilt water pump on #21 Diesel Generator, observed on January 28, 1986.
- PM 2-74-M-SA-3, Remove, Test, Replace Diesel Injectors, observed on January 28, 1986.
- Inspection of wiped bearings on #11 AFW pump turbine, observed January 30, 1986.
- MO 206-004-736A, #23 AFW pump repacking, observed on February 21, 1986.
- PM 2-36-M-2-3, Oil change on #23 AFW pump, observed on February 21, 1986.
- 4KV Breaker repair observed on February 25, 1986.

4KV Breaker Failure

Following maintenance (repacking) on #23 Auxiliary Feedwater (AFW) Pump on February 21, 1986, the pump breaker would not close on demand. The breaker was racked out and inspected. The cause of failures was found to be a failed drive pawl on the breaker lever assembly. That assembly charges the closing spring. The affected breaker, as part of an overall upgrade program for all 4KV breakers, had been recently sent to the manufacturer (GE in Philadelphia) for a number of vendor recommended modifications. One of those modifications was the installation of a new drive pawl. In this case, the drive pawl was only tack welded to the lever plate (final welds had not been done). The tack weld failed, preventing the drive pawl from performing its proper function.

An inspection program was initiated for the remaining safety related 4KV breakers in the plant (approximately 60 breakers). As of March 4, 1986 the licensee had inspected all but six of those breakers. No similar problems were found. Four of the remaining breakers supply vital 480 VAC buses and cannot be removed from service until the next unit outages. These breakers are normally closed and do not receive automatic load shed opening signals. Therefore, during accident situations they would not be expected to open and therefore not expected to have to be re-closed (i.e., require action of the closing springs). The closing springs were verified to be presently charged (by local indication). The remaining 2 breakers supply the #21 and 22 Salt Water Pumps. Those breakers were scheduled for inspection on March 5, 1986.

The failed breaker was a GE 4KV Magne Blast breaker, type AMH 4-76-250-1D. It was subsequently repaired by a vendor representative.

The vendor is evaluating the problem to determine if it may be reportable under 10 CFR21 requirements. Completion of the 4KV breaker inspections will be followed by the NRC 318/86-03-01.

Salt Water (SW) Piping Leakage

The licensee discovered through-wall seepage at one location in the #12 SW header. The seepage was in a welded spool piece section of concrete-lined carbon steel (between an elbow and a flange) downstream of the #12 Component Cooling Water (CCW) heat exchanger. The area was immediately downstream of a normally open valve and upstream of a normally throttled valve. Ultrasonic testing of the immediate area was performed. Wall thinning was evident in a triangular shaped area (8 inches wide) around the seepage. The original pipe thickness was 0.375 inches. Pipe thickness in this area ranged from 0.16 inches (very near the seepage) to 0.2 inches. As of January 29, the licensee was evaluating the minimum allowed wall thickness. As discussed in previous inspection reports, the licensee has experienced through-wall leakage at other locations in the SW system (e.g. SW pump discharge elbow and straight run of piping in discharge of #13 SW pump). It appears that once the concrete liner is lost in an area, as might be caused by flow induced erosion, the then exposed carbon steel piping is subject to severe corrosion attack. Previously, the licensee concluded that isolated small through-wall leaks did not pose a threat to the seismic (structural) integrity of the piping nor would they prevent adequate SW flow to vital heat exchangers. They also concluded, however, that the leaks would create potential water spray problems. The inspector expressed concern to the General Supervisor, Operations (GSO) (who was the Acting POSRC Chairman at the time) that no program was in place to perform periodic examinations (NDE) of SW carbon steel piping (with higher priority given to areas of known high susceptibility) which could provide earlier detection of wall thinning problems. The GSO agreed that such a program would be prudent. Further evaluation/development of such a program was added to the POSRC outstanding items list. Implementation of this program will be followed by the NRC (IF1 317/86-03-03). Later in the inspection period another small through-wall hole developed on the opposite side of this spool piece. This further emphasizes the need for periodic examinations.

No violations were identified.

9. Surveillance

The inspector observed parts of tests to assess performance in accordance with approved procedures and LCO's, test results (if completed), removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

- STP 210B, Unit 1 Reactor Protection System Functional Testing, observed on February 24, 1986.
- STP 210B, Unit 2 Reactor Protection System Functional Testing, observed on February 27, 1986.

No violations were identified.

10. Radiological Controls

Radiological controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10CFR Part 20 requirements were observed.

Radiological Event

At 8:14 a.m. on February 4, 1986 the Unit 1 Main Vent Gaseous Activity increased to 2E3 cpm (normal count rate was 50 cpm). In accordance with facility Emergency Plan Implementing Procedures a Radiological Event was declared at 8:20 a.m. (this release did not meet the criteria for an Unusual Event classification). The source of radioactive gas leakage was traced to a leaking swage-lock fitting in the Unit 1 primary sample sink hood area. A modification had just been completed which allowed sample returns to be directed to the Unit 1 Reactor Coolant Drain Tank (RCDT) inside Containment. There is only isolation valve (no check valves), 1-SV-6529, in that line between the sample sink and the RCDT. When a chemist, who was lining up the system to draw a primary sample, opened 1-SV-6529, gaseous activity residing in the RCDT passed from the tank, up to and out of the swage-lock fitting, through the sample hood and into the main vent. The sample lineup was secured and the fitting tightened. Activity level in the main vent dropped, and the licensee secured from the Radiological Event at 8:40 a.m. The licensee's off site dose projection (by "Midas" system computer) showed a maximum off site dose for the duration of the event of about 9.5 E-5 mrem whole body and 2.3 E-2 mrem thyroid.

No violations were identified.

11. Fire Protection Inspection

From January 28-31, 1986 an inspection of the licensee's fire protection program was conducted. Major areas covered during this inspection included a verification of the fire main, sprinkler system and halon systems lineups and a check of the condition of fire doors, fire barriers, hose stations, fire extinguishers, and fire hydrants. In addition, equipment contained within the fire pump house was examined and fire protection training given during general orientation training and requalification training was reviewed. A number of preventative maintenance requirements and the surveillance program were discussed in detail with the licensee's senior fire protection specialist. Results of this inspection are as follows:

- The fire pump house and its associated equipment appear capable of performing their intended function. The general area was exceptionally clean. However, the inspector noted that maintenance request Tag No. 006085, "Diesel Exhaust Pipe Rusted Off", has been outstanding since July 13, 1985. Although the rusted exhaust pipe appears intact up to the roof line, there is a potential for water to enter the diesel through this pipe rendering the diesel fire pump inoperable. This was discussed with the Manager, Nuclear Operations (MNO). The MNO stated he would take steps to ensure the exhaust was repaired.

- During welding on a fire door under Maintenance Order No. 206-008-810A, a spot check was made to determine whether adequate fire protection measures were taken. All precautions listed on the plant's cutting-welding-grinding permit were being followed including proper approval by a fire protection specialist. One discrepancy noted was that the permit itself was not initialed, signifying precaution review, by the workers involved in the repair.

When brought to the attention of the fire watch, this discrepancy was corrected. Requalification training lesson plans were reviewed and found to adequately address the proper handling of cutting-welding-grinding permits.

- General Orientation Training was reviewed and found to adequately address fire protection topics. The training booklet needs updating to reflect the newly issued cutting-welding-grinding permit.
- The licensee's senior fire protection specialist appeared extremely knowledgeable of the condition and status of plant fire protection equipment and requirements. He adequately addressed all inspector concerns.

No violations were identified.

12. Instrument Air Valve Installation Error

About 9:30 p.m. on February 17, 1986, the licensee discovered that a check valve (2IA 741) in the Unit 2 Salt Water Instrument Air System had been installed backwards. When properly configured, that check valve provides a path for backup Instrument Air (IA) from the #22 Salt Water Air Compressor (SWAC), to (1) Containment, (2) the atmospheric dump valves (fail closed on loss of IA and can be manually overridden open), (3) Reactor Coolant pump bleedoff Containment isolation valve (fails closed on loss of air), and (4) a reactor coolant sample Containment isolation valve (fails closed on loss of air).

A licensee evaluation a loss of IA to Containment showed the only significant consequence to be an inability to perform core flush (to prevent boron precipitation).

This procedure is a long term action and would not be used immediately following an event such as a loss of coolant accident. Because (1) supplied valves fail to a safe condition, (2) time would be available to provide an alternate supply of air to Containment following an accident, and (3) there are two other means of supplying backup IA to affected components (#21 SWAC and an accumulator), the potential safety consequences of this valve installation error do not appear to be severe.

The inspector walked down the Unit 2 Salt Water Instrument Air System in the Unit 2 Service Water Pump room to confirm no other installation errors existed. He noted one error in the system drawing (OM 454, Revision 9 and the corres-

ponding M drawing). Each SWAC carries two contradictory labels (e.g. "#22 SWAC" and "#21 Salt Water Air Compressor"). He pointed this out to the Tic-ensee for correction. No other problems were identified.

The licensee walked down the Salt Water IA system on both units to verify that the other check valves in SWAC supply lines were properly installed. A disagreement between the OM 454 print and the "as built" configuration was found in the location of one check valve physically installed in the supply line to the atmospheric dump valves. The licensee is confirming that this was a print error and not an installation error. No other deficiencies were identified.

The fact that valve IA 741 was incorrectly installed and not discovered calls into question the adequacy of the controls in effect, particularly operational testing, at the time of system installation (estimated to be just prior to Unit 2 startup). The SWAC system was a modification to the original system design. The inspector discussed this with the General Supervisor, Plant and Project Engineering (GSPPE) and asked if additional system testing may be warranted, particularly since significant portions of the air flow paths from the SWAC's are not checked by existing periodic testing procedures. The GSPPE felt additional testing was appropriate and stated that such testing would be planned and conducted. If possible that testing would be conducted during plant operation. If the testing can not be performed during operation, it will be performed at the next scheduled outage for each unit. Completion of this testing will be followed by the NRC (318/86-03-02). The licensee pointed out that their current procedures for testing modified systems are much more comprehensive and would, therefore, be more likely to detect such a problem.

Failure to adequately install and test check valve 2IA 741 is a licensee identified violation. Because the tests of 10 CFR2, Appendix C, Section V are met, a Notice of Violation will not be issued.

13. Review of Periodic and Special Reports

Periodic and special reports submitted to the NRC pursuant to Technical Specification 6.9.1 and 6.9.2 were reviewed. The review ascertained: inclusion of information required by the NRC; and validity of reported information. The following periodic report was reviewed:

-- December, 1985 Operations Status Reports for Calvert Cliffs No. 1 Unit and Calvert Cliffs No. 2 Unit, dated January 10, 1986.

14. Exit Interview

Meetings were periodically held with senior facility management to discuss the inspection scope and findings. A summary of findings was presented to the licensee at the end of the inspection.