U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report: 50-317/87-14 License: DPR-53
50-318/87-16 DPR-69

Licensee: Baltimore Gas and Electric Company

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

Inspection At: Lusby, Maryland

Dates: May 19 - June 30, 1987

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

Approved: L. E. Tripp 7/9/87
L. E. Tripp, Chief, Reactor Projects Section 3A date

Summary: May 19 - June 30, 1987: Inspection Report 50-317/87-14, 50-318/87-16

Areas Inspected: (1) Facility activities, (2) routine inspections, (3) operational events, (4) RCP vibration (5) maintenance, (6) surveillance, (7) follow-up of events, (8) Information Notice 87-23, (9) radiological controls, (10) physical security, (11) Licensee Event Reports, (12) Emergency Plan drill, (13) Temporary Instructions, (14) licensee action on previous inspection findings, (15) fuel integrity, and (16) reports to the NRC.

Inspection Hours totalled 312 hours.

Results: Additional management attention and resources appear to be needed regarding the tolerance of the root cause of housekeeping deficiencies and ineffective drains within the Auxiliary Building. Failed fuel and elevated fission product activity could exacerbate drain problems should a significant event occur. Some improvements are being seen with regard to system engineers participation in surveillance testing. Also, during a detailed assessment of operational problems with pressurizer spray valves, the licensee identified that there was insufficient weld penetration on the spray valve extension bonnet assembly (detail 7).

DETAILS

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.

Night shift inspections were conducted on June 10 and June 30, 1987. Weekend inspections were performed on Saturday, May 30 and Sunday, June 21, 1987.

1. Summary of Facility Activities

   At the beginning of the period Unit 1 was in Cold Shut Down (Mode 5) finishing Environmental Qualification (EQ) upgrades and continuing review of ASME Maintenance Orders as a result of Mechanical Commercial Quality (MCQ) replacement part deficiencies as detailed in Inspection Report 317/87-10;318/87-11. Unit 2 was completing refueling outage (Mode 6) activities while simultaneously working EQ upgrades and reviewing MCQ deficiencies.

   On May 27 Unit 1 entered Mode 1 after completing surveillance, MCQ and EQ post maintenance testing. The unit paralleled to the grid, escalated in power on May 28 and remained at 100% power throughout the remainder of the period.

   By June 4 Unit 2 had completed most surveillance testing and all MCQ and EQ modifications. Start up testing identified problems with (1) a pin hole leak on #22 salt water header; (2) a bearing failure on #22 Containment Spray Pump; (3) #12 Emergency Diesel Generator voltage regulator problems; (4) a leak on AFW 130 check valve flange; (5) a pin hole leak on service water piping to #22 containment cooler; and (6) excessive vibration on #22B Reactor Coolant Pump (RCP). The licensee isolated the affected salt water piping and completed all other repairs except the vibration on #22B RCP. After numerous preliminary checks, on June 9 a determination was made that an inspection of RCP #22B pump intervals would be required. On June 15 disassembly revealed that the suction deflector plate of #22B RCP had separated from the pump shaft and lodged itself within the main impeller. Simultaneously, the licensee was procuring a replacement rotating assembly from Northeast Utilities. By June 25 the new (Millstone) pump was installed and alignment and RCP checks had begun.

   By June 30, at the end of the period, all RCPs had been balanced satisfactorily, and the unit was heated up to normal operating temperature and pressure to complete pre-start up testing.

2. Review of Plant Operation - Routine Inspections

   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, effluent monitoring, emergency power source operability, control room logs, shift supervisor logs, tag out logs, and operating orders.

No unacceptable conditions were noted.

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 was performed. Operability of instruments essential to system performance was assessed. The following systems were checked:

-- Emergency Diesel Generator (Electrical)*

-- Unit 2 Shut Down Cooling

*For this system, the following items were reviewed: The licensee's system lineup procedure(s); equipment conditions/items that might degrade system performance (hangers, supports, housekeeping, etc.); instrumentation lineup and operability; valve position/locking (where required) and position indication; and availability of valve operator power supply.

During the period the inspector did a walk down inspection of accessible components of the generator and electrical connection subsystems associated with the emergency diesel generators (DG's). This included the generator, generator output breakers, disconnect switches, DG motor control centers, DG MCC transfer switches, and DG control panels. Associated surveillance test procedures were reviewed to ensure Technical Specification Surveillance requirements were addressed. Information from the plant system description manual and the inspector's walk down were compared against the FSAR description. The inspector witnessed an overspeed test (PE 2-24-5-O-R) performed on DG #21. The governing Operating Instruction (OI-21) was reviewed for adequacy. The inspector selected a type of component (DG MCC transfer switches) used to supply power to DG auxiliaries and verified, by review of the purchase specification, that those components were purchased as Class 1E equipment. Additionally, he verified that they are included in the newly revised plant Q-list. He reviewed the PM program to ensure these components were adequately maintained. No deficiencies were noted in the above reviews.

Finally, the inspector reviewed electrical protective relay set point generation methodology with one of the licensee's senior electrical engineers. He discussed preventive maintenance/calibration of these relays with personnel of the licensee's electrical test department. He reviewed a recently issued Functional Test Procedure (FT-E-59 dated January 14, 1987) for GE and Westinghouse relays. During this review the inspector and licensee personnel realized the electric test department was not forwarding documentation of all test results to the on site relay engineer in the Nuclear Engineering Services Department (NESD). This was a new requirement, and the relay department was mistakenly forwarding this documentation to another company relay engineer (off site). The Relay Engineer, NESD, stated this deficiency would be resolved by achieving compliance with the procedure or, if after further discussion, they determine a more appropriate documentation program, then a procedure change would be initiated.

One of the goals of the licensee is to have results of surveillance tests and preventive maintenance (PM) forwarded to systems engineers for trending, and where appropriate, readjustment of PM periodicities and content. The inspector noted that although this is planned, it had not been implemented at the time of inspection. In fact no trending of PM's was being done.

The DG operating instruction carries a strong precaution not to allow diesels to run unloaded for greater than 5 minutes due to possibility of blower rotor blade to casing rub due to high differential temperature conditions. The inspector noted that the DG's are automatically started on certain accident conditions but may not be immediately paralleled to a bus and loaded (e.g., safety injection actuation with off site power available and, for DG #12, loss of off site power without the existence of a safety injection signal). He discussed this with the systems engineer. The precaution was issued by the vendor and has not been superseded by subsequent correspondence. However, DG's #11 and #12 now have newer design blowers with greater clearances that should eliminate the problem. Additionally, the engineer pointed out that on one occasion, the DG's had run unloaded, with the old blower design, for two weeks without failure indicating that the precaution is probably overly restrictive. The engineer assured the inspector that they are working closely with the vendor to ensure resolution of blower concerns and elimination of associated operational restrictions. Final resolution of this issue will be followed by the resident inspector.

Finally, the inspector noted during his review of the Q-list that the pressurizer power operated relief valves (PORV's) are qualified mechanically in accordance with safety grade requirements; however, they are not qualified electrically. He discussed this with the chairman of the Q-list committee and asked why this situation existed considering the fact that the PORV's are referenced and utilized by the Emergency Operating Procedures for Reactor Coolant System "Bleed and Feed" operation during a Feed Line Break event. The committee chairman stated this was the subject of discussions between a utility group and the NRC. The probable outcome of these discussions will be for full inclusion of the PORV's in the Q-list if the feed line break accident is a design basis accident discussed in Chapter 14 of the Final Safety Analysis Report. Currently, feed line break is not discussed in Chapter 14 and there is disagreement between the NRC staff and the licensee regarding whether feed line break is a design basis accident for this plant. Resolution of this question will determine how the Q-list will address the PORV's.

No unacceptable conditions were noted.

c. Biweekly and Other 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.

Auxiliary Building Inadequacies

On May 22, the inspector participated in a tour of the site with the Region I Regional Administrator and other NRC staff members. On June 1, the inspector performed a tour of Units 1 and 2 Auxiliary Buildings. During these tours, and numerous previous tours, the inspector noted an apparently declining trend in housekeeping, an increasing trend in the number of contaminated areas, and apparently declining demonstration of pride in workmanship below the standards normally evidenced by the work force. This was evidenced by failure to maintain good cleanliness controls during and after maintenance; inattention while removing protective clothing resulting in step off pad areas being cluttered with contaminated clothes and waste; and an apparent increase in visible boron crystals indicative of leaks on valves, stems, pump shafts, and the surrounding areas.

The inspector recognizes that much of this deterioration of standards has resulted from the Unit 1 and 2 ten year In Service Inspection outages, exascerbated by the extensions due to Environmental Qualification and the Material Qualification issues and RCP problems. Further investigation however, revealed that the Facilities Support group (the personnel responsible for decontaminating the Radiological Controlled Areas) have been overloaded with additional areas to be decontaminated due to Auxiliary building drains backing up and recontaminating numerous previously clean areas. This problem of contaminated drains backing up has been occurring frequently during the past several years. The supervisor of Facilities Support maintains records of these "events" indicating that the drains backed up thirty-three times during the first six months of 1986 and periodically since. These events range from one floor drain to all the floor drains on the minus ten foot and minus fifteen foot levels of the Auxiliary Building overflowing simultaneously.

Review of logs and discussions with technicians reveal that attempts to clean the drains have been unsuccessful. Placing rubber floats in the drains to block the drain when the water level becomes high and routing the drains have also been unsuccessful.

An Engineering Support Request dated March 10, 1986 was submitted to fix the problem. However, the problem was assigned as a very low priority even though it has been known to be a problem for several years. Each time the problem occurs it inhibits access, (requires use of full PC's and often respirators), causes additional contaminated areas, facilitating the spread of contamination, enhances the probability of personal contaminations (a weakness at Calvert Cliffs), generates additional rad waste, places additional work load on Facility Support resources (rework), degrades working conditions for maintenance, operations, and surveillance personnel, and contributes to the degradation of the appearance of the controlled area.

Problem Description

The Auxiliary Building drain system drains to the Miscellaneous Waste Receiver Tank (MWRT) along with several other systems including the Emergency Core Cooling Pump Room sump which is fed from the containment sump. The ECCS pump room sump is the only system which is automatically pumped to the MWRT. Each system is separated from the MWRT by check valves. These are designed to prevent back flow to the adjacent systems.

The problems appear to be:

(1) The drain piping appears to have a reduced capacity due to material accumulations in the drains, i.e., cement dust from boring holes in walls, Black Beauty sand from sand blasting, and other miscellaneous debris.

(2) The check valve design utilizes a heavy weighted disc which often requires an excessive head of liquid before the disc will open.

(3) If the check valve disc sticks shut, liquids in the drain accumulate and back up.

(4) If the check valve disc sticks open when the ECCS pump sump automatically operates or when drains are utilized from higher elevations, the -15 drains over flow with liquid rad waste from the ECCS pump room sump or elsewhere.

(5) During the performance of certain surveillance test procedures which require cross connecting head tanks, oscillations in level occur which cause the tanks to over flow into the drain system which exceed the capacity of the drain system and over flow the drains on the lower levels.

This appears to be a problem which the licensee has been tolerating for several years and one which the engineering department has been unsuccessful at resolving, thereby causing rework for facilities support and unnecessary rad waste.

The inspector requested the licensee to evaluate the consequences of a small break LOCA or Reactor Coolant Pump seal failure coincident with significant fission products in the coolant with respect to what controls are in place to prevent the fission products from eventually being spread throughout the Auxiliary Building, and whether these controls are adequate. The concern stems from the fact that fission products in the containment sump would be drained to the ECCS pump room sump. In order to determine a leak rate, subsequently the ECCS sump would be automatically pumped to the MWRT however, because of drain problems, the drains would overflow in the rooms and passageways of the Auxiliary Building.

The licensee stated they would consider cleaning the drain system again after both units have resumed operation. Additional management attention and resources appear to be needed in this area.

3. <u>Operational Events</u>

<u>Inadvertent Boron Dilution (2 Events)</u>

Two inadvertent boron dilutions of the Unit 2 Reactor Coolant System (RCS) occurred during the period. Both were the result of an improperly positioned valve in the Chemical and Volume Control System (CVCS). Neither dilution caused RCS boron concentration to decrease below the value needed to meet Technical Specification shutdown margin requirements. Following is a description of the events.

During the day on June 2, 1987, with Unit 2 in Cold Shutdown condition (Mode 5), Reactor Coolant System (RCS) boron concentration slowly decreased from 2478 ppm (6:20 a.m.) to 2344 (5:00 p.m.). The Technical Specification (TS 3.1.1.2) required shutdown margin was greater than 3% Delta K/K (2100 ppm boron). The RCS letdown/charging system had just recently been returned to service and the boronometer was in operation. That instrumentation was showing the decrease in boron concentration; however, the rate of decrease was slow, and the slope of the trace on the recorder was small enough not to attract the operator's attention. During this time the operations group was filling the Refueling Water Tank (RWT) with a blended boric acid/demineralized water mixture utilizing the make up system. Day shift operators had noticed that, during the additions to the RWT, Volume Control Tank (VCT) level was also increasing, indicating that the make up isolation valve to the VCT (2-CV-512) was leaking. This would not be expected to be of an immediate concern because it should have resulted in a high concentration boric acid solution entering the VCT.

The oncoming evening shift noted the boron decrease on the boronometer, ordered an immediate RCS chemistry sample to verify boron concentration, added 690 gallons of boric acid to the RCS to increase shutdown margin, had the Electrical and Controls group adjust the regulator for 2-CV-512 to ensure it was fully closed, and had the chemistry group check RCS boron concentration on a once per hour basis until they confirmed there was no further boron dilution ongoing. The dilution had stopped because the RWT fill operation had been ended earlier and not because of the above actions.

The operations group believed that leakage by 2-CV-512 was the root cause. Even with the RC make up pumps secured, the demineralized water header is maintained slightly pressurized. They felt that with the VCT at very low pressure conditions (typical for Mode 5 operation), demineralized water could leak by 2-CV-512 into the VCT.

The following day, June 3, day shift personnel confirmed that the 2-CV-512 leakage problem had been resolved and then again began filling the RWT. As a result of another operation in progress, pressurizer pressure reduction, very little flow was passing through the boronometer thereby making it an unreliable indicator of RCS boron concentration. About 4:00 p.m. normal flow was returned through the boronometer and operators observed a rapid 100 ppm decrease in boron concentration. The RWT fill operation and charging flow were secured. An RCS sample showed 2330 ppm boron. A VCT sample showed 2451 ppm. Again, a search for a dilution source was initiated, resulting in the finding of 2-CVC-308 to be three turns open.

As part of an effort to identify the cause of the dilution the operations group intended that valve 2-CV-308 be checked shut. When that valve is opened a flow path is established for demineralized water to pass from the discharge of the RC make up pumps directly to the charging pump suction header (bypassing the demineralized water blending isolation valve). However, apparently due to a communications breakdown that valve position was never checked. Later (after a second event) it was learned that mispositioning of 2-CVC-308 was the cause of the problem. A valve lineup had recently been performed (June 2, 1987) on the Chemical and Volume Control System by a single individual. The valve lineup indicated that 2-CV-308 had been checked shut.

The inspector asked the licensee to consider making 2-CVC-308 a "locked closed" valve. The General Supervisor, Operations (GSO) stated that currently he was considering this action. On June 12 a procedure change (CCOM 87-210) was issued making the corresponding 308 valves on both units locked closed valves. The GSO reported that the operator who performed the June 2 CVCS lineup recalled that he did check 2-CVC-308 shut. Operations personnel cycled the valve and found that it was not difficult to operate.

The event was discussed with the two shifts involved. A Performance Improvement Report (PIR) will be drafted by personnel involved to determine lessons learned from the event to help prevent recurrence.

Reactor Coolant Pump Problems

On June 4, 1987, with the licensee heating up the Unit 2 reactor coolant system following completion of a refueling outage, high vibrations were experienced on the #22B Reactor Coolant Pump (RCP). During the first minute of pump run, vibration readings were normal (about 7-8 mils). Then a step jump in vibration (up to 15 mils) was experienced. The pump was shut down after 14 minutes of operation. Later on June 4, a second run of the pump was attempted. Again vibrations were normal for the first minute of operation, and then a step increase occurred to 50 mils. Vibration then increased more slowly up to 72 mils over the next 13 minutes of operation. The pump was then shut down. The licensee began checking all equipment associated with the pump/motor assembly that was accessible without disassembly of the pump itself, e.g., motor bearing, coupling, supports, motor hold down bolt torque, vibration instrumentation. A locking nut associated with the coupling was reported to have a low break away torque; however, this did not turn out to be the root cause of the vibration problem. On June 9, 1987, #22B RCP was started and vibrations were still high.

The licensee concluded that the root problem must be internal to the pump and began preparations for disassembly. As a part of these preparations, an ultrasonic examination (UT) was performed on the pump shaft to look for indications of a crack. The licensee had obtained assistance from the pump vibration monitoring equipment manufacturer in analyzing the vibration data they had obtained. That vendor advised them that the shaft might be cracked. The UT examination was principally intended to see if a crack was present which would interfere with or prevent the lifting operation of the pump internals. The licensee recognized that this type of UT examination of the shaft was difficult and would utilize an unproven technique. Similar examinations at another plant had given false indications of cracks. The UT exam did show an indication that possibly extended 25% around the circumference of the shaft in the ACME thread area. Subsequent PT and UT examinations, performed in the local ACME thread area after pump disassembly, showed that the above indication was false and no circumferential crack existed.

When the rotating assembly was removed from the pump casing, the suction deflector ring ("donut"), which normally attaches to and covers a ring of eight 1-1/4 inch diameter bolts (that fasten the impeller to the shaft) and two drive pins (which transmit the torsional load between shaft and impeller), was found to have broken loose in the eye of the impeller, moved radially outward and jammed between two blades of the impeller. The deflector ring is 13-1/4 inches in outer diameter, weighs about 40 pounds, and is attached to the impeller by two 3/4 inch cap screws (ASTM type A286). The cap screws are torqued to 50 ft-lbs. and secured in place by means of a 1/4 inch x 1/4 inch x 2 inch stainless steel locking bar welded on either end to the body of the donut. Closer inspection revealed that the locking bar welds had fractured on one of the cap screws and that the cap screw had apparently fully backed out of its hole. The cap end of the second screw was intact in the donut; however, the screw had broken in the threaded portion leaving half of the screw in the impeller hole and half in the donut. The mechanisms of locking bar weld and screw failures were not positively known at the close of the inspection period, but it is suspected that the screw broke due to fatigue failure after the first screw backed out. Further metallurgical examination of the failed screw is planned. The vendor (Byron-Jackson) made a wooden template the size of the deflector ring and verified, using a spare impeller that a ring could not pass all the way through the impeller. In fact, it can not pass a sufficient distance through the impeller to contact the casing and cause possible pump seizure. A search, using a camera probe, was made of the pump suction piping to locate the missing cap screw/locking bar. Additionally, the #22 steam generator cold leg plenum was opened and inspected for the same reason. The bolt/locking bar could not be found. These items probably passed through the impeller. At the close of the report period an analysis of the possible effects of these loose parts was being conducted by Combustion Engineering and the licensee. Preliminary conclusions indicated that only the locking bar had the potential for entering the fuel region. If this happened, it would not cause flow blockage, but could lead to fretting damage to a small number of fuel pins.

The licensee believes that the mass shift associated with the breaking of the donut was sufficient to have been the root cause of the pump vibration.

Examination of other areas of the pump determined the following. The rotating and stationary elements of the hydrostatic bearing had come into contact and worn sufficiently to require replacement of the stationary element. The lower part of the rotating assembly had contacted and worn the casing wear ring; however, an analysis of the post-rub wear ring clearances showed that the ring could be reused (additional pump bypass flow acceptable). Axial thermal cracking, similar to that found on the Unit 1 #12A pump shaft (see section 4 of Inspection Report 50-317/86-19;50-318/86-19) was found on the shaft at the top of the ACME threaded region (where temperatures fluctuate significantly due to the mixture of hotter reactor coolant bleed off flow and cooler RCP seal recirculation flow). The cracks were in a 360 degree band extending from the 3rd thread from the top down to the 10th thread. The maximum crack length was 0.9 inch. The maximum crack depth was 70 mils below the thread root area. A previous study done on the thermal crack problem by the pump vendor (Byron-Jackson) and B&W and later confirmed by an independent study by another consultant (MPR) firm showed that the maximum depth predicted for thermal cracks to extend before arresting was 250 mils. The report concluded that this type of cracking was not a concern. Thermal cracking was also noted on the corresponding area on the pump cover. Those cracks were axial and extend for a 1 inch, 360 degree band around the cover bore. The longest crack was 0.85 inches long and extended from the chamfered area immediately above the threads down to the 6th thread from the top. A Byron-Jackson report also had been done for the cover which predicted thermal cracks which would arrest before a depth of 85 mils. The licensee tried unsuccessfully to measure the depth of the cover cracks by Eddy current testing. The complex geometry (with component cooling water flow holes) and composition (3 types of materials) of the cover prevented accurate measurement of these cracks by Eddy current, radiography, or UT. Since there was uncertainty on the depth of the cover cracks and since the cover wall separating Reactor Coolant bleed off flow (2250 psig) and Component Cooling System (150 psig design maximum pressure) flow channels is relatively thin (400 mils) the licensee performed a safety analysis regarding the likelihood of crack progression and possible consequences. The analysis concluded that it would be very unlikely for a crack to propagate through-wall without arresting and that, even if this should occur, the cover material is very tough and not subject to a brittle or catastrophic mode of failure. It concluded that there would be a leak before break situation, that any leakage would be small, and that mechanisms existed to provide early warning of this leakage. The analysis was reviewed in a joint meeting of the plant and the off site safety committees on Sunday, June 21. The resident inspectors attended this meeting. The conclusions of the analysis and a description of the cracking were presented by the licensee by telephone on June 22 to NRC Region I materials specialist personnel. Copies of the Byron-Jackson cover study and the MPR shaft study will be provided by the licensee to the NRC for review.

The licensee obtained a new rotating assembly and installed it in the #22B RCP. This is the first failure of the donut assembly seen in a Byron-Jackson pump.

No unacceptable conditions were noted.

5. 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, personnel qualifications, radiological controls for worker protection, fire protection, retest requirements, and reportability per Technical Specifications. The following activities were included:

-- No. 21 Emergency Diesel Generator Overspeed Test (PE 2-24-5-0-R)

-- No. 22B RCP Rotating Element Disassembly

-- No. 22 LPSI Pump High Vibration Correction

-- No. 22 Containment Spray Pump Bearing Repair

-- No. 2-RV-439 Low Pressure Safety Injection Relief Valve

No unacceptable conditions were noted.

6. 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 test was reviewed:

-- ISI Inspection of Unit 2 Steam Generator Hydrostatic Test

-- STP-04, Unit 1 Integrated Emergency Safety Features Test

-- STP-05, Unit 1 Auxiliary Feed Water System Test

-- STP-07, Unit 1 Engineering Safety Features Monthly Logic Test

-- STP-08-A, Unit 1 12 DG and 4KV Bus 21 LOCI Sequence Test

-- STP-086, Unit 1 Boration Flow Path Temperature Determination

-- STP-087, Unit 1 Borated Water Source Operability Verification

-- STP-M-3-2, Main Steam Safety Valve Testing

-- PTP-2-RCS-10Y-1, Reactor Coolant System Hydrostatic Test

-- PSTP-2, Low Power Physics Testing

-- NET-3, Nuclear Engineering Procedure-3, Internal Vibration Monitoring

During the performance of STP-M-3-2 Main Steam Safety Valve testing it was noted that the system engineer coordinated the evolution. Several mechanics, quality control inspectors and the system engineer demonstrated a well orchestrated job characterized by efficiency. The testing was performed by test procedure using calibrated instruments, correct tools, appropriate test controls, and conducted in a professional manner. The system engineer appeared suitably knowledgeable of the test procedure and the Main Steam Safety valves.

No unacceptable conditions were noted.

7. <u>On Site Follow Up of Events</u>

-- On May 25, while preparing to heat up Unit 1 to normal operating temperature and pressure from cold shutdown, operators signed the prerequisites of OP-1 "Plant Startup from Cold Shutdown" stating that two boric acid water sources were available as required by Technical Specification 3.1.2.2. Subsequently, plant heat up commenced at 7:25 p.m.

At 9:00 p.m. the heat up continued and the plant entered Mode 4 operation (greater than 200 degrees F). At 9:50 p.m. a surveillance test procedure STP-086-1, Boron Flow Path operability check, was completed satisfactory.

At 1:15 a.m. on May 26 operators commenced dilution of the RCS in order to approach the critical boron concentration. This was performed by the "direct" method which requires isolating the Refueling Water Tanks (RWT) from the charging flow path. The RWT is also considered one source of boric acid water.

At 2:55 a.m. operators pumped No. 11 Boric Acid Storage Tank (BAST) to No. 12 BAST in order to top off No. 12 and commence the regeneration of No. 11; this caused No. 11 BAST to be inoperable (i.e., without sufficient capacity/level or proper concentration).

At 4:46 a.m. the RCS temperature increased above 300 degrees F (entered Mode 3). Twenty minutes later the operators recognized that they had changed modes with less than the required (2) borated water sources available. They then secured the dilution and reopened the RWT to charging pump isolation valve. At 6:35 a.m operators restored No. 11 BAST to proper level and concentration.

This event was determined by the Shift Supervisor to be potentially in violation of Technical Specification 3.0.4 which prohibits entry into an operational mode unless the conditions of the limiting conditions for operations are met without reliance on the action statement, since only one source of boric acid was immediately available. This was reviewed by the Plant Operations Safety Review Committee as a possible reportable event pursuant to 10 CFR 50.73. The committee established that since the evolution was performed by procedure, in a controlled manner and the control room operators were aware of the status of the isolation valve, the RWT was available and operable and that, in any event requiring RWT water to the suction of the charging pumps, operator action would be required under normal or the current conditions.

This was discussed with the inspectors who reviewed the event. The inspectors determined that no inadequate conditions were noted.

-- On April 27, 1987 both units were in Mode 5. Maintenance and engineering personnel were assessing operational problems with pressurizer spray valves. Engineering personnel noted on the valve drawings that the extension bonnet assembly to the pressurizer spray valve may not have sufficient weld penetration to meet the ASME code requirements. Follow up of this perceived inadequacy with the vendor, ITT Corporation and Hammel Dahl, Inc., determined that the design of the welds was inadequate, and that the welds did not meet the code specifications.

BG&E engineering then performed a fatigue analysis which predicted crack initiation at 270 pressurization cycles. At this time, Unit 1 and 2 had experienced 54 and 37 full cycles, respectively. Therefore, based on no crack initiation, the stress analysis was re-performed to determine an adequate weld size to meet primary, secondary, and cyclic stress limits. Using the values calculated in this new analysis, all the deficient welds were repaired by adding 1/4 inch thickness to the existing weld with a 3:1 taper into the base material. As of May 22, 1987, all repairs were completed and the welds are now in compliance with the design specifications and construction code. This is considered an isolated incident due to the uniqueness of the design modification in that the extension bonnet assembly was an addition to the pressurizer spray valves specifically for Calvert Cliffs. The licensee reported this event to the NRC in Licensee Event Report No. 87-10.

No unacceptable conditions were noted.

8. NRC Information Notice No. 87-23

Loss of Decay Heat Removal During Low Reactor Coolant Level Operation

A copy of this Information Notice was delivered to the licensee on May 26, 1987. This information was discussed with the licensee. Because of previous losses of shut down cooling and other concerns specifically (IEB 84-03, Refueling Water Cavity Seal) the licensee has installed refuel pool cavity level instrumentation equipped with alarms and variable set points. These instruments are portable and are set up in the control room during conditions which require draining the RCS water level. Operators are provided detailed values of where water level is at and where it should be. Operators are required by procedures to set tolerances for alarms which occur when water level deviates from the set point.

The licensee has evaluated when vortexing might occur and incorporated into the operating instruction a caution to monitor for fluctuating pump amps and flow instabilities when starting a second pump in the shut down cooling system. An operations maintenance coordinator has been established to minimize the potential of work proceeding which might affect the shut down cooling system operation. Operators receive training on loss of shut down cooling events during the annual requalification process. This includes recovery actions from a loss of shut down cooling.

Based on the above the licensee appears to be adequately addressing controls for shut down cooling, however, this Information Notice was sent to the Plant Operations Event Assessment Committee for further assessment.

9. 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 10 CFR Part 20 requirements were observed. Independent surveys of radiological boundaries and random surveys of nonradiological points throughout the facility were taken by the inspector.

No unacceptable conditions were identified.

10. 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 unacceptable conditions were noted.

11. 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 follow up. The following LER's were reviewed:

LER No.	Event Date	Report Date	Subject
Unit 1			
87-09	04/23/87	05/21/87	Use of Fasteners (Bolts, Studs, Threaded Rods & Nuts) in ASME Class 1, 2, & 3 Systems Without Proper Certification, Special NDE, or Special Marking
87-10*	05/22/87	06/19/87	Pressurizer Spray Valve Bonnet Weld Design Deficiency
Unit 2			
87-04	05/07/87	06/05/87	Failure of Inlet Piping to Relief Valve (2-RV-439) (Details of this event are described in paragraph 3 of Inspection Report 317/87-10; 318/87-11)

*Detailed examination of this event is documented in paragraph 7 of this inspection report.

No unacceptable conditions were noted.

12. Emergency Plan Drill

On June 18, 1987, the licensee conducted an Emergency Plan Practice Drill. The inspector observed drill activities from the Control Room, Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). Communication with state and local authorities were established during the drill. The inspector noted that the licensee has strengthened the staffing of the dose assessment center at the EOF by a second individual to assist the Radiological Assessment Director (RAD).

The "MIDAS" off site does projection computer was used effectively. This was an improvement over the previous annual exercise where, although available, the computer was not fully utilized and heavy reliance was placed on hand calculation, rule of thumb estimates. A third party consultant, with expertise in emergency planning/off site dose projection, was asked by the licensee to observe and comment on the drill. These actions showed licensee responsiveness to NRC concerns as well as initiative in seeking other potential improvement areas.

The drill scenario appeared somewhat weak in that it provided minimal challenges to dose assessment personnel at the EOF. This made it difficult for the inspector to assess the degree of training and capabilities of this group.

No unacceptable conditions were noted.

13. Status of High Priority Temporary Instructions

In a memorandum from the Office of Inspection and Enforcement to each regional office, a list of ten Temporary Instructions was provided as high priority Temporary Instructions (TIs). The following provides a status of each of these items:

TI Number	Subject	Status
2500/19	RX Vessel Overpressure Protection	Awaiting Region I management to coordinate with NRR and provide specific direction to inspectors as specified by this TI.
2500/20	Implementation of ATWS Rule 50.62	Documented in Inspection Report 317/86-10.
2515/84	Event V - Interfacing LOCA	Not Applicable to CCNPP.
2515/85	Mk I Containment Program	Not Applicable to CCNPP.
2515/86	Natural Circulation Cooldown	Scheduled to be completed during a future planned inspection.
2515/87	Inst. to Follow Course of Accident (RG 1.97)	Scheduled to be completed during a future planned inspection.
2515/88	Flooding of Equipment Important to Safety	Awaiting Region I management to coordinate with NRR and provide specific direction to inspectors as specified by this TI.
2515/89	Inspection of BWR SS Piping	Not Applicable to CCNPP.
2515/90	SDV Capability	Not Applicable to CCNPP.
2515/91	Salem ATWS Item 4.1	Previously documented in Inspection Reports 317/84-31, 85-30, and 86-10.

14. Licensee Action on Previous Inspection Findings

(Closed) 317/84-31-02, Refueling Cavity Water Seal. The licensee has remained with the mechanical steel ring plate and two "M" shaped silicone rubber seals each enveloped by nine steel channel seal covers placed end to end around the circumference of the seal assembly. This design has demonstrated to be an effective seal. Additional level instrumentation equipped with alarm functions have been incorporated in the control room to alert operators of decreases in cavity water level. This item is closed.

15. Status of Fuel Integrity

The existence of some number of failed pins had been indicated throughout the operating cycle by an elevated level of fission product activity in the Reactor Coolant System (RCS). Coolant levels of fission product isotopes Iodine-131, -133, and -134, and Cesium-134 and -137 were previously projected to determine the numbers of failures and some characteristics of those failures.

During both the Unit 1 and Unit 2 refueling/10 year In Service Inspection (ISI) outages, the entire core was off loaded to the Spent Fuel Pool to facilitate the ISI work. During this time extensive examinations were conducted on the fuel in search of fuel pin defects such as failed cladding, blisters, end caps loose or missing or other defects as evidenced by NDE examinations.

An ultrasonic examination technique was employed to identify failed pins. Each fuel assembly (there are 217 in the core) contains 176 fuel pins in a 14x14 array (20 spaces are taken up by Control Rod Guide Tubes). Other techniques (visual examination and Eddy current) were employed to confirm failure prior to pin replacement and to identify the cause of failure.

The results of confirmatory Eddy current testing and visual examination showed that of the approximately 38,000 pins in the core, 3 pins scheduled for use in Unit 1 Cycle 9 and 6 pins scheduled for use in Unit 2 Cycle 8 were failed.

The licensee determined that of the 9 failed pins described above (3 from Unit 1 and 6 from Unit 2) at least 6 failures were caused by debris-induced fretting (loose material in the RCS became lodged at a fuel assembly spacer grid and vibrated due to RCS flow until it worked through the fuel cladding).

The licensee discussed the effects of the above in the Calvert Cliffs news letter and emphasized the importance of maintaining cleanliness controls and material accountability in the RCS and associated piping.

No unacceptable conditions were identified.

16. 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; test results and/or supporting information;

consistency with design predictions and performance specifications; adequacy of planned corrective action for resolution of problems; determination whether any information should be classified as an abnormal occurrence; and validity of reported information. The following periodic report was reviewed:

-- Operating Data Reports for Calvert Cliffs Units 1 and 2, dated June 12, 1987.

No unacceptable conditions were identified.

17. 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.