U.S. NUCLEAR REGULATORY COMMISSION
Region 1

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Docket Nos.:	50-317; 50-318
Report Nos.:	50-317/99-07; 50-318/99-07
Licensee	Baltimore Gas and Electric Company Post Office Box 1475 Baltimore, Maryland 21203
Facility:	Calvert Cliffs Nuclear Power Plant Units 1 and 2
Location:	Lusby, MD
Dates:	August 15, 1999 to October 2, 1999
Inspectors:	Scott Stewart, Senior Resident Inspector Fred Bower, Resident Inspector Tim Hoeg, Resident Inspector
Approved By:	Michele G. Evans, Chief Projects Branch 1 Division or Reactor Projects

Executive Summary
Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Inspection Report Nos. 50-317/99-07 and 50-318/99-07

This routine inspection report summarizes aspects of BGE operations, maintenance, engineering and plant support. The report covers a seven-week period of resident inspection.

Plant Operations

BGE effectively prepared for the tropical storm weather associated with Hurricane Floyd by augmenting plant staff and delaying planned maintenance. Safety systems remained operable during the storm. Following the storm, BGE promptly initiated a number of departmental self assessments to verify the adequacy of the site severe weather response plans and to gather the lessons learned. (O1.1)

Unit 1 was manually shutdown in anticipation of an automatic low steam generator level trip signal following a loss of main feedwater on September 22, 1999. The plant was effectively stabilized in hot standby in a timely manner. BGE determined that the proximate cause for the event was human error during preparations for maintenance on non-vital switchgear that powered main feed pump support systems. Appropriate interim corrective actions were implemented pending the completion of the final root cause determination by the Significant Issues Findings Team. The reactor was returned to full power operation without complication. (O1.2)

Maintenance

The 1A emergency diesel generator (EDG) failed its monthly surveillance test when a failed component in the governor electronic control system caused erratic speed control after the engine started. Although this event was considered a maintenance rule functional failure, the threshold for placing the 1A EDG in (a)1 status was not met. (M1.1)

The containment radiation signal (CRS) B train logic module failed. BGE performed a risk assessment and determined that replacing the logic module while at power would increase plant risk to approximately 15 times normal. Since the CRS is not required while the plant is at power, BGE deferred the performance of the work to a shutdown period. This was an excellent example of the use of individual plant examination results to assess and manage risk, and avoid placing the plant in a risk significant configuration. (M1.1)

Maintenance and quality verification personnel failed to follow procedures that require all work instructions to be included in the work package and used at the work site during compression fitting maintenance. This failure to follow station procedures was treated as a non-cited violation. The safety related compression fitting maintenance on the 12 steam generator auxiliary feedwater flow control valve positioner was performed successfully and the valve returned to its standby configuration. (M1.3)

Executive Summary (cont'd)

Engineering

The inspectors concluded that an operability determination regarding the corrosion behavior of the cladding for high duty fuel was appropriate in scope and detail. BGE concluded that the Unit 1 and Unit 2 reactor cores were operable for the remainder of their current operating cycles, since the fuel was expected to remain within all the applicable design limits. (E1.1)

Plant Support

BGE's fitness for duty program for random drug testing is well defined and proceduralized. Interviewed fitness for duty personnel were knowledgeable and demonstrated a sensitivity for employees' rights and the disclosure of private information. BGE's fitness for duty program, policies, and procedures reviewed during this inspection were satisfactory. (S1.1)

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Report Details

Summary of Plant Status

During this inspection period, Unit 1 operated at or near 100 percent reactor power, with three noteworthy exceptions. On September 17, 1999, reactor power was reduced to 89 percent to support main condenser waterbox cleaning during tropical storm conditions associated with Hurricane Floyd. The reactor was returned to full power on September 19. On September 21, a near fully withdrawn control element assembly (control rod) dropped, causing a power reduction to approximately 95 percent. The control rod was recovered and the reactor was returned to full power later the same day. On September 22, the reactor was manually tripped in anticipation of an automatic trip signal from low steam generator water level due to a loss of main feedwater flow. The reactor remained in hot standby until September 23. Unit 1 was reconnected to the grid and returned to full power on September 24, 1999. Additional minor power reductions were conducted to support routine maintenance and testing.

With two exceptions, Unit 2 operated at full power this inspection period. On September 16, reactor power was reduced to 85 percent to support main condenser waterbox cleaning during tropical storm conditions associated with Hurricane Floyd. The unit was returned to full power on September 17. On September 18, reactor power was reduced to 93 percent to again clean the main condenser water boxes. Unit 2 returned to full power on September 19.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Plant operations were conducted safely with a proper focus on nuclear safety. The inspectors conducted daily tours of the control room to observe the conduct of activities and verify safety system alignments. Control room operators were aware of plant conditions, remained attentive, and conducted shift turnovers in a thorough manner. Communications were formal and in accordance with BGE policies. Operators demonstrated effective implementation of self-checking and peer-checking techniques. Good alarm response, including the use of alarm response procedures, was noted. Control room equipment status was clearly identified. During the inspection period, management expectations concerning minimizing control room distractions were reinforced and control room personnel appropriately implemented these expectations.

The inspectors accompanied the Unit 1 auxiliary building, Unit 2 auxiliary building, and Unit 1 turbine building plant operators on tours during deep backshift hours on August 29, September 5, and September 26, respectively. The plant operators conducted the rounds and recorded logs in accordance with BGE procedures.

During the daylight hours of September 16, 1999, the site experienced tropical storm weather associated with Hurricane Floyd. Heavy rains and high tropical storm winds impacted the site. Storm debris, which included small fish and grasses, were entrained with the cooling water intake for the main turbine condensers on both units. A power reduction to 85 percent power on Unit 1 was accomplished as traveling screen clogging necessitated the stopping of a cooling water pump. Unit 2 operators conducted a preemptive power reduction to 90 percent, in the event that a cooling pump on that unit needed to be stopped. BGE had effectively prepared for the storm by maintaining augmented plant staffing and by delaying planned maintenance to ensure maximum safety system availability during the storm. All safety systems remained operable during the storm, although operator actions were necessary to flush the saltwater/service water heat exchangers on Unit 1 to maintain system flows within the allowable limits. Following debris removal from the main condenser waterboxes, reactor power was restored to 100 percent at both units. The inspectors observed that BGE promptly initiated a number of departmental self assessments to verify the adequacy of the site severe weather response plans and to gather the lessons learned.

On September 21, following control element assembly (CEA) manipulations for reactivity and power control, CEA 37 became unlatched and dropped into the core reducing power to approximately 95 percent. No control rod manipulations were ongoing at the time of this rod drop. CEA 37 was recovered and the reactor was returned to full power later in the day. On September 23, troubleshooting was performed on CEA 37. The troubleshooting included withdrawing and inserting the rod while recording control element drive mechanism gripper coil currents. BGE determined that the traces were normal with no anomalies noted. The vendor also reviewed these traces and confirmed BGE's conclusions.

O1.2 Unit 1 Manual Reactor Trip and Recovery to Full Power

a. Inspection Scope

The inspectors observed activities associated with a manual reactor shutdown of Unit 1 on September 22. The inspectors were onsite during the event and went to the control room to observe operator response to the transient and to assess the post-trip review activities. The inspectors also observed portions of the recovery of Unit 1 from hot shutdown to full power during normal and backshift hours.

b. Findings and Observations

On September 22, at 9:08 a.m., Unit 1 was manually shutdown (tripped) from 100 percent power. The manual shutdown was performed in anticipation of an automatic shutdown signal on low steam generator water level due to a loss of main feedwater flow. Feedwater flow was lost as a result of an overcurrent condition trip of the feeder breaker supplying two non-vital 480 volt motor control centers (MCCs 106 and 116). De-energization of these MCCs resulted in the loss of the main and auxiliary oil pumps for both main feed pumps.

All control rods fully inserted and all safety equipment operated as designed during and following the automatic shutdown. Loss of MCCs 106 and 116 also resulted in a loss of condenser vacuum which caused the control room operators to close the main steam isolation valves. Reactor decay heat removal was established using auxiliary feedwater and the atmospheric dump valves. A reactor engineer and additional licensed operators responded to the control room. Plant operators quickly and effectively stabilized the reactor in Mode 3 (Hot Standby). Plant management and quality assurance personnel provided appropriate oversight during the event. BGE properly notified the NRC of the trip in accordance with 10 CFR 50.72. Later, the inspectors reviewed the BGE post-trip review and found no discrepancies.

BGE performed insulation testing to verify the integrity of MCCs 106 and 116 and the associated loads. The feeder breaker was tested to confirm the accuracy of the overcurrent trip settings. The MCCs 106 and 116 were subsequently re-energized and returned to service without complications. The inspectors observed routine post-trip activities, which included evaluation of the transient, confirmation of fuel and reactor coolant pump seal integrity, and recovery planning. A BGE Significant Issues Findings Team (SIFT) was assembled to determine the root cause(s) of the trip and to recommend corrective actions to prevent recurrence.

A loss of condenser vacuum and condensate flow following the Unit 1 shutdown, resulted in several large water hammers in the condensate system. Systems and design engineering personnel responded to observe the transient and to walkdown the system, observing pipe hangers, piping supports, and high stress area configurations. Design engineering evaluated their walkdown results and directed nondestructive examinations (NDE) be performed on specific pipe sections to assess for stress cracking. The engineering staff presented the results of their water hammer evaluation to the Calvert Cliffs Plant Operational Safety Review Committee (POSRC) on September 23. Based upon their evaluation, the engineering staff recommended restart of Unit 1. The POSRC accepted the engineering staff's recommendation.

On September 23, the SIFT reported to the POSRC their preliminary assessment of the plant trip. The team determined that the MCC 106 feeder breaker responded to an actual current condition that was above the initiation setpoint of the long-time delay overcurrent trip. The SIFT determined that the proximate cause for the event was that the current exceeded procedure limits because a technician incorrectly measured the current with a clamp-on amp meter. The technician read only one-half of the actual current through the breaker because the current was measured on only one of the two conductors per phase feeding the motor control centers. Compensatory actions initiated by BGE included placing all line breaker maintenance requiring cross ties on hold pending risk assessment and providing training on the event to operations and maintenance personnel. The SIFT recommended restart of Unit 1and continuing the investigation of the human performance issues surrounding the event. The POSRC voted to accept the SIFT recommendations.

The forced outage repairs and testing were completed on September 23. On September 24, the reactor was taken critical and Unit 1 was returned to power operation without complication. The inspectors observed selected portions of the reactor restart, generator paralleling, and power escalation. Access control to the control room was limited in order to prevent distractions to the operators during the start-up and power ascension evolutions. Reactor engineering personnel were observed in the control room making confirmatory observations of the reactor restart. Criticality was achieved at the estimated critical condition (ECC). The inspector reviewed the ECC calculation and considered it complete. A pre-evolution power ascension brief was conducted by the Unit 1 senior reactor operator (SRO) reviewing special precautions, procedural requirements, watchstander responsibilities, and past experiences. The inspector considered the pre-evolution brief to be detailed and timely. Management and quality assurance oversight personnel were observed in the control room monitoring the power ascension. The inspector observed proper repeat back communications between the licensed operators during the power ascension process. The SRO exhibited good command and control functions during power escalation, clearly taking the leadership role. The reactor plant was returned to full power on September 25.

At the end of the inspection period, the root cause analysis by the SIFT was ongoing. An issue report (IR3-021-325) had been initiated to enter the event into the BGE corrective action program. The areas of focus by the team were: (1) training and qualification; (2) risk assessment; (3) human performance, and, (4) instrumentation design deficiencies.

c. Conclusions

Unit 1 was manually shutdown in anticipation of an automatic low steam generator level trip signal following a loss of main feedwater on September 22, 1999. The plant was effectively stabilized in hot standby in a timely manner. BGE determined that the proximate cause for the event was human error during preparations for maintenance on non-vital switchgear that powered main feed pump support systems. Appropriate interim corrective actions were implemented pending the completion of the final root cause determination by the Significant Issues Findings Team. The reactor was returned to full power operation without complication.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Licensee Event Report (LER) 50-317/99-04: Reactor Trip Due to Main Transformer Bushing Flashover

The LER described the July 24, 1999 reactor trip of Unit 1 during harsh weather. The event was previously inspected and reported in NRC Inspection Report 50-317&318/99-06, Section O.2. There were no issues identified in the inspector's in-office review of this LER that were not previously inspected and reported. The LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

On September 13, the 1A emergency diesel generator (EDG) failed its monthly surveillance test when speed control was erratic after the engine had been started. An issue report was written to document the problem and technical specification action statement 3.8.1.B was entered for an inoperable diesel generator. BGE maintenance and engineering personnel diagnosed the problem as a failed component in the governor electronic control system. The control system was replaced requiring an engineering test (ETP 99-012) to assure that the diesel control system was capable of its design function. On September 15, the engine was tested satisfactorily, the action statement was exited, and the engine was returned to service. This event was considered a maintenance rule functional failure; however, the threshold for placing the 1A EDG in (a)1 status was not met.

On September 24, the containment radiation signal (CRS) B train logic module failed. The CRS is an engineered safety features actuation signal (ESFAS) provided to limit the release of radioactive fission products during refueling and maintenance periods when containment integrity is breached. The CRS isolates and secures the containment purge system and is not required during power operations. BGE initially identified the replacement of the logic module as a high priority maintenance item. BGE assessed risk associated with de-powering the ESFAS B train logic cabinet to replace the CRS logic module. BGE determined that performing this maintenance while at power would increase plant risk to approximately 15 times normal. BGE reduced the priority of the logic module replacement and deferred the performance of the work to a shutdown period. The inspectors considered this an excellent example of using individual plant examination results to assess and manage risk, and avoid placing the plant in a risk significant configuration.

M1.2 Routine Maintenance Observations

a. <u>Inspection Scope (62707)</u>

The inspectors reviewed maintenance activities and focused on the status of work that involved systems and components important to safety. Component failures or system problems that affected systems included in the BGE maintenance rule program were assessed to determine if the maintenance was effective. Also, the inspectors directly observed all or portions of the following work activities:

MO 1199903996	Troubleshooting and Repair of 11 Feed Regulating Bypass Valve
MO 1199903954	CEA 37 Troubleshooting
MO 2199803117	22 High Pressure Safety Injection Pump Motor Replacement
MO 1199903518	1-AFW-4512-CV Corrective Maintenance
MO 1199902890	24 Battery Charger Clean and Inspect

b. <u>Observations and Findings</u>

During the maintenance activities, the inspectors observed that technicians were experienced and knowledgeable of their assigned duties. Maintenance personnel practiced peer checking and self-verification while doing work. The pre-job briefings included the important aspects of each maintenance task and were effective in ensuring the work was conducted in accordance with BGE requirements. Supervisory oversight was appropriate.

c. Conclusions

During the selected maintenance activities, the inspectors observed that the technicians were experienced and knowledgeable of their assigned duties. Maintenance personnel practiced peer checking and self-verification while performing work.

M1.3 Auxiliary Feedwater Control Valve Maintenance

a. Inspection Scope (62707)

The inspector observed maintenance activities associated with the replacement of the positioner for the 12 steam generator flow control valve, 1-AFW-4512-CV. The inspector reviewed the applicable maintenance order and quality assessment of the maintenance. This maintenance was chosen for observation because the auxiliary feedwater (AFW) system is the most important risk reduction system at the plant.

b. Observations and Findings

On August 20, 1999, AFW control valve 1-AFW-4512-CV was time tested in the open direction per surveillance procedure STP-O-5A-1, AFW System Quarterly Surveillance Test. The valve opened in 24.1 seconds vice the nominal design time of 41 seconds. An immediate retest was performed with a time of 28.5 seconds. The valve was considered operable because the test results indicated that the valve was still capable of performing it's safety function. Control room personnel contacted the system engineer in a timely manner and requested an engineering evaluation. The valve's condition was entered as a potential limiting condition for operation (PLCO) in the control room logs and an Issue Report (IR3-058-110) was written.

BGE performed a visual inspection of the subject valve and identified a worn positioner assembly. A maintenance order (MO) was written to replace the positioner. On August 23, the inspector observed maintenance being performed on 1-AFW-4512-CV in accordance with MO No. 1199903518. The inspector observed two instrument and controls (I&C) technicians and a Quality Verification (QV) assessor at the job site. The MO required the valve positioner to be replaced in accordance with BGE technical procedure TUBE-01, Use of Compression Type Fittings. TUBE-1 provides directions to ensure proper tightening of compression type fittings associated with valve positioner air lines. The inspector identified that a copy of TUBE-01 was not included with the MO package and was not at the job site. The inspector mentioned this to one of the I&C technicians and they had one delivered in a timely manner. The valve positioner was subsequently replaced, tested satisfactorily, and returned to service within the limiting condition for operation (LCO) allowed outage time.

Inspector follow-up of the failure to include procedure TUBE-1 in the valve positioner replacement MO identified that Calvert Cliffs administrative procedure MN-1-101, Control of Maintenance Activities, Section 5.6.E.1, Maintenance Pre-work Review, requires the responsible maintenance group supervisor(s) review the MO package for completeness and accuracy, including having the required procedures available prior to performing maintenance. Further, Section 5.8.D.5.a, Maintenance Order Performance, requires that the responsible technicians ensure appropriate technical references are present at the job site when performing maintenance. BGE procedure PR-1-103, Use of Procedures, Section 5.2.D.1, Technical Procedure Usage, requires referral use procedures to be at the work site. In this particular case, the QV assessor present at the job site had also overlooked the absence of the work instruction, which is contrary to Quality Inspection Process Guideline NPADG-17 Section 5.10.A.1.f, Conducting the Inspection, which instructs the quality verification assessor to ensure that the appropriate work instructions are physically at the job site. Through follow-up discussions with the licensee, the inspector learned that having TUBE-1 at the job site for work on compression fittings was particularly important because of a history of maintenance-related failures of compression fittings due to technician error.

This severity level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee corrective action program as documented in Issue Report Nos. IR3-033-073, IR3-028-361, and IR3-028-362. **(NCV 50-337&338/99-07-01)**.

c. <u>Conclusions</u>

Maintenance and quality verification personnel failed to follow procedures that require all work instructions to be included in the work package and used at the work site during compression fitting maintenance. This failure to follow station procedures was treated as a non-cited violation. The safety related compression fitting maintenance on the 12 steam generator auxiliary feedwater flow control valve positioner was subsequently performed successfully and the valve returned to its standby configuration.

M1.4 Routine Surveillance Observations

a. <u>Inspection Scope (61726)</u>

The inspectors observed all or portions of the following surveillance tests:

STP-O-5A-1	Unit Auxiliary Feedwater Operability Test
STP-M-573-1	HPSI Piping System Leakage
STP-O-73D-1	Charging Pump Performance Test
ETP 99-012	1A Emergency Diesel Generator

b. <u>Observations and Findings</u>

The inspectors found that the selected surveillance activities were performed safely and in accordance with approved procedures. Test details were discussed at a pre-test briefing attended by all test participants. A question and answer session followed and all involved test participants left the brief with clear expectations. The test participants were knowledgeable of their assigned responsibilities. Supervisory and engineering personnel participation was clearly observed in the conduct of the surveillance tests. Minor test discrepancies were documented in the BGE corrective action program for correction.

c. Conclusions

Surveillance testing was thorough and consistent with industry standards. The inspectors observed that minor discrepancies noted during the tests were properly entered into the corrective action system.

III. Engineering

E1 Conduct of Engineering

E1.1 Corrosion Behavior Of High Duty Fuel

a. Inspection Scope

The inspectors reviewed the operability recommendation for the corrosion behavior of high duty fuel.

b. Findings and Observations

NRC Inspection Report 50-317&318/99-06 documented that BGE initiated a design verification inspection for spent reactor fuel assemblies to check that critical performance expectations were met for clad oxide layer thickness and the absence of crud or unexpected corrosion. The initial eddy current and visual examinations identified some blistering (spalling) of the clad oxide layer on some fuel pins where none was expected. The inspections also found a generally thicker oxide layer than expected for the given fuel burn-up. Additional inspections and engineering analysis to fully characterize these indications and to evaluate potential causes were implemented by BGE. The preliminary evaluation included a study of the effects of higher burn-up and burn-up rates for selected fuel pins. The unexpected oxide layer thickness was correlated with higher fuel duty cycles (temperature and local power). The oxide layer thickness could be an indication of hydrogen embrittlement in the zirconium based cladding. BGE engineering personnel told the inspectors that a potential consequence of thicker oxide layer and spallation could include fuel exceeding its design basis in certain postulated core upset scenarios. At the end of the inspection period for NRC Inspection Report 50-317&318/99-06, the effects of the observations on the specific core analyses for Calvert Cliffs were being evaluated.

Nuclear fuel oxide layer spalling and its impact on fuel cladding integrity are not currently considered in the Calvert Cliffs fuel performance or safety analyses; therefore, BGE considered the observed oxide blistering/spallation as an unanalyzed condition. The inspectors reviewed Functional Evaluation/Operability Determination 99-010, "Fuel Cladding." The operability determination reviewed the high burn-up fuel behavior against the fuel cladding design basis, cladding design limits (mechanical properties), and all applicable design basis events. The operability determination was supported by an engineering evaluation that was performed by the fuel vendor. BGE determined that the fuel cladding oxidation is expected to remain within the design limit of 120 microns; and therefore, the maximum fuel cladding stress limits remain bounded during steady-state operation and design basis events. BGE determined that the cladding strain capability was expected to remain above the one percent design strain limit. BGE also determined that the expected localized oxide blistering and spallation would have no significant impact on the fuel temperature during steady-state operation or design basis events. BGE's operability determination concluded that Unit 1 and Unit 2 are safe at full power for the remainder of their respective current fuel cycle, since all applicable design limits are met.

A root cause determination for this issue (IR3-020-203) is ongoing. BGE determined that an evaluation and response to this issue is warranted to support licensing of the follow-on reload cycles.

c. Conclusions

The inspectors concluded that an operability determination regarding the corrosion behavior of the cladding for high duty fuel was appropriate in scope and detail. BGE concluded that the Unit 1 and Unit 2 reactor cores were operable for the remainder of their current operating cycles, since the fuel was expected to remain within all the applicable design limits.

IV. Plant Support

S1 Conduct of Security and Safeguards Activities

S1.1 Fitness For Duty Program

a. Inspection Scope (81502)

The inspector reviewed BGE's fitness for duty (FFD) program associated with random drug testing and confirmation of positive urinalysis results. The inspector reviewed applicable BGE procedures, policies, and interviewed key program personnel.

b. Observations and Findings

The Calvert Cliffs fitness for duty program is outlined in BGE procedure SE-1-100, "Fitness For Duty Program." The medical review officer's roles and responsibilities are defined in BGE procedure SE-1-300, "Access Authorization Manual." The inspector reviewed the subject procedures and found them to be adequately written, properly approved, and administratively controlled. The procedures effectively outlined the roles and responsibilities of the fitness for duty program staff including the medical review officer (MRO), FFD program manager, drug screening coordinators, collectors, and access requesters. The procedures provided clear and concise actions to be taken when responding to confirmation of positive urinalysis results, appeals processes, and confidentiality and disclosure of information. The inspector verified the subject procedures complied with 10 CFR 26.

The FFD program manager and a BGE MRO were interviewed and found to be professional and knowledgeable in regards to their responsibilities and authorities. BGE has a zero tolerance for positive drug tests. BGE and contractor employees with confirmed positive tests are denied access authorization to the protected area and are subject to disciplinary action up to and including discharge. The FFD program manager and MRO were sensitive to employee's rights for appeals, requests for sample retests, and the confidentiality and disclosure of sample information.

c. Conclusions

BGE's fitness for duty program for random drug testing is well defined and proceduralized. Interviewed fitness for duty personnel were knowledgeable and demonstrated a sensitivity for employees' rights and the disclosure of private information. BGE's fitness for duty program, policies, and procedures reviewed during this inspection were satisfactory.

V. Management Meetings

X1 Exit Meeting Summary

At the conclusion of the inspection, on October 20, 1999, the inspectors presented the inspection results to Mr. Katz and others of BGE management. BGE acknowledged the findings presented.

X2 Management Meeting Summary

On September 13, 1999, NRC Chairman Greta Dicus toured the Calvert Cliffs site and met with BGE managers and staff. The discussions included a review of recent site activities and were general in nature. Chairman Dicus was accompanied by members of her staff. Mr. Hubert Miller, Region I Administrator, represented Region I during the visit.

On September 22, 1999, sixteen members of the staff of the NRC Chief Financial Officer toured the Calvert Cliffs site with BGE staff and met with Mr. Katz, Plant General Manager. The discussions were general in nature.

ATTACHMENT 1

Partial List of Persons Contacted

BGE
C. Cruse, Vice President, Nuclear Energy Division
P. Katz, Plant General Manager
K. Cellars, Manager, Nuclear Engineering
L. Wechbaugh, Superintendent, Nuclear Maintenance
M. Navin, Superintendent, Nuclear Operations
B. Montgomery, Director, Nuclear Regulatory Matters
S. Sanders, General Supervisor, Radiation Safety
T. Sydnor, General Supervisor, Plant Engineering
D. Holm, General Supervisor, Plant Operations
T. Pritchett, Superintendent, Technical Support
L. Smialek, Radiation Protection Manager
C. Earls, General Supervisor, Radiological/Chemistry

NRC

W. Cook, Project Engineer, Division of Reactor Projects Branch 1
M. Evans, Chief, Division of Reactor Projects Branch 1

INSPECTION PROCEDURES USED

IP 71707	Plant Operations
IP 62707	Maintenance Observation
IP 61726	Surveillance Observation
IP 37551	Onsite Engineering
IP 71750	Plant Support Activities
IP 81502	Fitness for Duty Program

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
BGE	Baltimore Gas and Electric Company
CEA	Control Element Assembly
CRS	Containment Radiation Signal
ECC	Estimated Critical Condition
EDG	Emergency Diesel Generator
ESFAS	Engineered Safety Features Actuation Signal
ETP	Engineering Test Procedure
FFD	Fitness for Duty
I&C	Instrument and Controls
IR	Issue Report
LCO	Limiting Condition Operation
LER	Licensee Event Report
MCCs	Motor Control Centers
MO	Maintenance Order
MRO	Medical Review Officer
NCVs	Non-Cited Violations
NDE	Nondestructive Examination
PLCO	Potential Limiting Condition for Operation
POSRC	Plant Operational Safety Review Committee
QV	Quality Verification
SIFT	Significant Issues Findings Team
SRO	Senior Reactor Operator

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened / Closed

50-337&338/99-07-01	NCV	Procedure Non-Compliance Associated with Maintenance on AFW Flow Control Valve to the 12 Steam Generator

Closed

50-317/99-04	LER	Reactor Trip Due to Main Transformer Bushing Flashover