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REGION I

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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, Maryland
Dates:	March 2, 1997 through April 12, 1997
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EXECUTIVE SUMMARY

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Inspection Report Nos. 50-317/97-02 and 50-318/97-02

This integrated inspection report includes aspects of BGE operations, maintenance, engineering, and plant support. The report covers a seven week period of resident inspection and includes the results of announced inspections by engineering and health physics specialists.

Plant Operations

The inspectors observed that the Unit 2 shutdown was well coordinated through the use of a detailed pre-evolution brief which included communications, self-checking techniques, and peer verifications of control manipulations. Fuels and feedwater system engineers provided good support during the shutdown.

BGE responded appropriately to the discovery of missing/broken lockwires on four Unit 2 safety relief valves (SRV). An operability evaluation was thorough and reasonable and the followup SRV testing during unit shutdown for refueling validated the BGE conclusions.

On March 28, the fuel transfer carriage with an irradiated fuel assembly became stuck in the fuel transfer tube due to a poor material condition of the carriage. The inspectors found that BGE had either not identified or had not corrected a number of deficient conditions and problems related to fuel handling activities. Specifically, the inspectors found that contrary to 10 CFR 50, Appendix B, Criterion V, appropriate drawings were not used during the troubleshooting of the stuck carriage; contrary to 10 CFR 50, Appendix B, Criterion XVI, adequate corrective actions were not taken to ensure fuel handling could be accomplished without incident; and, contrary to procedures, the spent fuel pool ventilation system was not properly aligned during fuel handling operation. These were considered apparent violations. Additionally, BGE identified two apparent violations with technical specification requirements related to fuel handling and a failure to follow fuel handling procedures. Based on these findings, the inspectors considered that management involvement in the handling of irradiated fuel was poor.

Maintenance

The inspector found that the BGE inservice inspection program was well implemented, and the steam generator tube eddy current inspection was very well planned and executed. The steam generator inspection indicated an emphasis on safety, preplanning, and training of personnel for an effective inspection and data analysis. BGE determined both steam generators to be Technical Specification Category C-3 due to greater than one percent of the tubes being defective. All of the defective tubes were removed from service by plugging.

Engineering

In general, BGE's assumptions of valve factor, load sensitive behavior, and stem friction coefficient were considered conservative and acceptable for GL 89-10 closure. However,

due to the small amount of in-plant dynamic test data obtained, BGE's basis for these assumptions for non-tested valves was considered weak and could be improved by obtaining additional information as part of the long term motor operated valve (MOV) program.

BGE had adequately addressed MOV design-basis capability for the majority of valves in the Calvert Cliffs Nuclear Power Plant GL 89-10 program. However, design basis capability concerns were identified regarding the power operated relief valve (PORV) block valves, and a violation was cited due to inadequate corrective action. Also, several issues to be addressed during the long term MOV program were identified regarding the steam generator feedwater isolation valves and the safety injection pump to refueling water tank minimum flow isolation valves.

BGE's actions to address pressure locking and thermal binding concerns regarding motor-operated gate valves were generally acceptable.

BGE had developed an adequate tracking and trending program for addressing MOV performance problems. However, the inspectors concluded that the program was not effective in fully evaluating the performance of a recent Unit 1 power operated relief valve block valve leakage problem.

Plant Support

BGE Nuclear Security retained an individual to provide special investigative skills to the security department during potential tampering events. The inspectors considered this a noteworthy enhancement in BGE's ability to appropriately respond to these events. Also, BGE informed the inspectors that an interfacing procedure between Operations and Security was being developed to formalize when and how security assistance would be provided when tampering was suspected.

Overall outage planning from a radiological controls perspective was very good and overall ALARA controls were very good.

BGE implemented a generally effective internal exposure control program.

Although no unplanned exposures were identified, BGE's radiation work permit program had no controls to preclude workers from signing in on invalid radiation work permits or permits that had been revised since last read by the workers.

BGE installed electronic, camera surveilled, doors to allow access to high radiation areas and reduce aggregate exposure to radiation protection personnel. However, there was no specific guidance for verification of personnel entering the areas to ensure those entering were in fact so authorized.

High radiation area key control procedures did not provide guidance for modifying key inventory. Examples of keys not on the inventory were noted.

Executive Summary (cont'd)

BGE's failure to control access to the Unit 2 containment, a high radiation area, via the emergency airlock, and the failure to document the problem in the corrective action system were contrary to procedure and an apparent violation.

Inspector observations indicated weaknesses in oversight and control of on-going non-routine radiological work activities. BGE lost control of a diver allowing the individual to enter previously unsurveyed areas in the spent fuel pool. The inspector determined a failure to provide adequate instructions, as required by 10 CFR Part 19, to the diver who unknowingly entered an unsurveyed portion of the spent fuel pool; the failure to ensure an individual is not able to gain unauthorized or inadvertent access to areas where radiation levels could be 500 rads or more in an hour as required by 10 CFR Part 20; and, the failure to perform reasonable surveys to ensure compliance with 10 CFR Part 20 were apparent violations.

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ATTACHMENTS

Report Details

Summary of Plant Status

Unit 1 remained at full power throughout the inspection period.

Unit 2 started the inspection period at 90 percent power and continued a planned power reduction until March 14, when the unit was shutdown for a refueling outage. The outage extended through end of the inspection period.

I. Operations

O1 Conduct of Operations [1]

O1.1 General Comments (71707)

Overall, the plant was operated safely with a focus on continued nuclear safety. On March 14 and 15, the inspectors observed portions of the plant shutdown for the 1997 refueling outage. Nuclear Fuels Management (NFM) engineers had determined the optimum control rod positions to minimize the axial symmetry index transient during the shutdown. The operating crew completed training for the shutdown using the NFM strategy. The inspectors observed that the shutdown was well coordinated by operations personnel through the use of a detailed pre-evolution brief, three-point communications, self-checking techniques, and peer verifications of control manipulations. An NFM engineer and the feedwater system engineer provided good support during the shutdown. The General Supervisor-Nuclear Plant Operations provided management oversight. The inspectors concluded that the shutdown for the Unit 2 refueling outage was very well planned and conducted with a strong regard for nuclear safety.

On April 7, a Unit 1 control element assembly (CEA) dropped from a middle position to the full in position. At the time of the drop, actions were in progress to restore the CEA to the full out position following a rod movement test. Operators appropriately implemented the abnormal operating procedure and the rod was quickly restored to the normal withdrawn position. Since the particular CEA had a low reactivity worth, the power transient was less than one-half percent of full power. The inspectors considered the operator actions effective in managing the transient. Troubleshooting, initiated to determine the cause of the CEA drop continued at the end of the inspection period.

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[1] Topical headings such as O1, M1, etc., are used in accordance with the NRC standardized reactor inspection report outline found in MC 0610. Individual reports are not expected to address all outline topics.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdown (71707)

The inspectors used Inspection Procedure 71707 to walk down accessible portions of the 1A emergency diesel generator (EDG) and related support systems. The inspectors determined that the 1A EDG was properly aligned for automatic operation upon the receipt of a start demand. In general, the material condition of the systems and the EDG building was very good, with trash and transient combustibles kept to a minimum and in appropriate containers. The inspectors noted several minor operating procedure discrepancies involving component location which were brought to BGE operations management attention. The inspectors did not identify any substantive concerns as a result of the walkdown.

O2.2 Unit 2 Reactor Defueling

a. Inspection Scope

The inspectors observed and reviewed reactor defueling activities on Unit 2.

b. Findings and Observations

To support an inservice inspection of reactor components, BGE planned to defuel Unit 2, placing the reactor core in the spent fuel pool. Defueling was started on March 27.

On March 28, while moving a spent fuel assembly to the spent fuel pool, an overload occurred on the transfer carriage, stopping the carriage in a mid position between the spent fuel pool and the fuel transfer tube. An initial attempt to move the carriage by hand cranking using an installed winch indicated that the carriage was stuck. In the position, the carriage blocked the transfer tube isolation valve preventing isolation of the spent fuel pool from the refueling pool. During the event, there was no need to shut the transfer isolation valve. A notification to the NRC was made in accordance with 10 CFR 50.72 b(2)(iii)(D), for inability to close the isolation valve. BGE stated that the safety significance of the inability to close the gate valve was low because a passive steel seal was used to isolate the reactor pressure vessel annulus from the refueling pool and because the stuck assembly would remained covered with water if the seal failed.

An initial visual inspection by system engineering personnel through the roughly forty feet of water in the pool identified that a limit switch actuating magnet attached to the outside of the fuel transfer carriage was missing a cap screw, had become dislodged, and was contacting the spent fuel pool upender assembly, preventing full movement of the carriage into the spent fuel pool. At that time, engineering personnel, based on the visual inspection, incorrectly stated that the magnet assembly was comprised of three magnets, each connected to the carriage by two capscrews. Additionally, it was incorrectly identified that a second magnet was out of position and that a second cap screw was missing.

Based on the initial engineering assessment, BGE prepared a contingency plan that provided for physical removal of the magnet assembly if plant operators needed to shut the transfer tube isolation valve which could occur if inventory was being rapidly lost in either the spent fuel pool or the refueling pool.

BGE also prepared a procedure for attempting to push the dislocated magnet into place using a customized tool. According to the plan, as the magnet was pushed into place, the carriage would be hand cranked into the spent fuel pool upender. The fuel assembly could then be placed in the spent fuel pool and additional action could be taken to repair the carriage. The plan when tried did not work, because the customized tool was the wrong size and because the controlling distance from the top of the pool to the repair area, roughly 40 feet, was too far to accurately manipulate the tool.

Following the attempted procedure, a camera inspection of the carriage was completed which identified that only one cap screw was missing and the dislodged magnet was attached to the carriage by two in-place lockwired capscrews. The second magnet suspected to be out of position was not part of the limit switch assembly and was correctly installed. Subsequently, the carriage was moved to the refueling pool, the fuel assembly was removed, and the carriage was returned to the spent fuel pool for repairs.

The inspectors noted that the initial troubleshooting plan and initial engineering assessment were done without consulting engineering drawings of the carriage limit switch magnet assembly and were based on faulty information. The discrepancies included the incorrect assumption that only one, instead of two, cap screws would have to be broken to remove the dislodged magnet and that a second magnet was out of position. The inspectors considered these discrepancies significant because the engineering assessment, contingency plans, and management approvals were not based on the best available engineering information, but rather a faulty assessment. The incorrect information was used in the contingency plan, in the BGE assessment of the problem, and in the 10 CFR 50.72 report to the NRC. 10 CFR 50, Appendix B, Criterion V requires activities affecting quality be accomplished in accordance with instructions, procedures, or drawings appropriate to the circumstances. The failure to implement appropriate drawings of the carriage assembly during the development of the troubleshooting plan was considered an apparent violation of NRC requirements. **(EEI 30-317&318/97-02-01)**

Subsequently, a number of camera and diver inspections were completed to fully inspect the refueling equipment and complete repairs. Also, a number of tests were done and additional deficiencies with the equipment were identified and corrected. During one of the dives, a potential radiation controls event occurred (See Section R1.5 of this report). Significant equipment deficiencies and problems identified during the defueling included:

- Prior to starting defueling, the refueling machine camera had failed. BGE had written continued fuel movement without the camera into the refueling

procedures. BGE informed the inspectors that the cameras were generally unreliable and were not needed for fuel movement activities.

- During an empty carriage transfer test, the spent fuel pool upender vertical switch grounded and tripped due to a faulty switch. The switch was replaced.

- Prior to starting defueling, the auxiliary hoist, used in control element assembly swaps, had continued to the full out position after the control switch had been placed in stop. The switch had failed and was replaced. BGE informed the inspectors that had this uncontrolled movement occurred during a control element assembly swap, the operators could have stopped the hoist by opening a power supply breaker on the refueling machine.

- During the initial fuel moves, a control element drive assembly cable was caught by the moving refueling bridge and damaged.

- During initial fuel moves in the spent fuel pool, a grapple closed light remained lit when the grapple open light was lit. Fuel movement continued as allowed by the procedure using a fuel spotter to determine if the fuel assemblies were ungrappled. At the time, the problem had been assessed and engineering personnel were preparing a design change to correct the issue after defueling.

- A relief valve stuck open on the refueling upender, preventing lowering the upender. The cause was sludge and other debris in the hydraulic system for the upender due to wear and age related degradation. BGE corrected the condition and wrote a preventive maintenance procedure to clean the upender hydraulics prior use.

- After the stuck fuel assembly was removed from the carriage, but prior to completing defueling, metallic debris was identified in the carriage and noted in the refueling log. An evaluation of the debris was not done, and system engineering was not aware of the problem. After the log entry was made detailing the issue, and after questioning by the NRC, engineering personnel could not identify if there was metallic debris or why the log entry was made.

- Following resumption of defueling, the hoist latch light would not light and a bridge and trolley lockout occurred. Troubleshooting identified that a loose screw came out of a spare terminal in the logic circuit for the refueling bridge and contacted the hoist latch contacts, stopping the machine.

- After defueling, the proximity switches on the spent fuel pool upender and transfer machine were replaced because some of the cable jacketing had fallen off exposing the conductors.

- A number of cable overloads had occurred on the refueling and spent fuel transfer machines due to loose cables. Following defueling, the cables were tightened.

- Following defueling, solenoid valves for the upenders were replaced because they were leaking by. Also, numerous air and water hoses on the refueling machine were replaced because they were brittle and leaking.

NRC Inspection Report 50-317&318/96-03 reported the failure of welds on the refueling pool upender during fuel movement activities on Unit 1 in 1996. The inspectors considered that the number and significance of deficiencies and problems during fuel handling had escalated and were concerned that BGE had not taken proper corrective actions for the identified fuel movement problems to ensure that fuel handling could be accomplished without incident. 10 CFR Part 50, Appendix B, Criterion XVI, stated that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and nonconformances are promptly identified and corrected. The failure to ensure adequate corrective actions for the problems as they were identified, and to preclude repetition following the initial problems was an apparent violation of NRC requirements. **(EEI 50-317&318/97-02-02)**

BGE identified during their review of the spent fuel pool ventilation problem (See NRC Inspection Report 50-317&318/96-10) that a surveillance test normally performed to demonstrate the operability of the spent fuel pool ventilation system had not been performed since September of 1994. The issue was identified to the NRC in Licensee Event Report 50-317/97-001. The missed surveillance was a requirement of Technical Specification 4.9.12.d.2. The failure to comply with the technical specification was an apparent violation of NRC requirements. **(EEI 50-317&318/97-02-03)**

During the troubleshooting of the refueling equipment and prior to the restart of defueling, BGE determined that they had not been in compliance with Technical Specification 3.9.6.a.2. The technical specification stated that during movement of fuel assemblies within the reactor pressure vessel, an overload cutoff limit of less than or equal to 3000 pounds would be in effect for the refueling machine. BGE identified that for six inches of refueling machine hoist travel, the overload limit was bypassed by design of the machine. The bypass occurred while the fuel assembly was above the core, but within the boundaries of the pressure vessel. The cutoff was used to ensure breakaway of the refueling machine hoist box from its landed position during fuel assembly withdrawal. Refueling personnel informed the inspector that brief loads (spikes) in excess of 3000 pounds had been observed during past refueling activities while in the bypass zone. The inspectors considered the failure to implement the technical specification as an apparent violation of NRC requirements. **(EEI 50-317&318/97-02-04)**

To reestablish compliance with the technical specification, BGE modified the refueling machine instituting the 3000 pound cutoff and installing a reset switch to reset the machine should a cutoff occur. Fuel movement was restarted on April 5

at 8:39 p.m.. Operators were cautious to slowly transition from fuel movement to fuel and hoist box movement, to minimize the possibility of 3000 pound overloads and no 3000 pound overloads occurred after defueling was restarted.

The inspectors also questioned BGE whether the refueling equipment had been included in the 10 CFR 50.65 (maintenance rule) program. BGE informed the inspectors that the system was considered for inclusion, but was not because the system had no specific safety related function. BGE wrote an issue report to re-evaluate inclusion of refueling equipment within the scope of the maintenance rule program.

On April 23, BGE started refueling the Unit 2 core. On April 24, during a shift turnover, plant operators noted that the spent fuel area ventilation system charcoal filters were not aligned for fuel handling activities. Proper alignment of the ventilation system required that charcoal filters be in service during fuel handling. Use of the charcoal filters during fuel handling was included in the updated final safety analysis report, the technical specification bases, and the safety evaluation for a fuel handling event. The charcoal filters were required to be aligned for fuel handling by Calvert Cliffs procedure Operating Instruction OI-25, Spent Fuel Handling Machine. Although this event occurred after the end of the inspection period, it was included in this report by the inspectors because of its proximity and relevance to other fuel handling problems. NRC Inspection Report 50-317&318/96-10 included a Notice of Violation for incomplete procedures and failure to follow procedures during fuel handling operations in the spent fuel pool. At that time, BGE had not taken actions to ensure that the spent fuel pool ventilation system was appropriately aligned for fuel handling. The failure to follow fuel handling procedures in not aligning the spent fuel pool ventilation system during fuel handling was an apparent violation of NRC requirements. **(EEI 50-317&318/97-02-05)**

c. Conclusions

The inspectors found that BGE had either not identified or had not corrected a number of deficient conditions and problems related to fuel handling activities. On March 28, the fuel transfer carriage while carrying an irradiated fuel assembly, became stuck in the fuel transfer tube due to a poor material condition of the carriage. Additionally, BGE had identified two apparent noncompliances with technical specification requirements related to fuel handling and a failure to follow fuel handling procedures. Based on these findings, the inspectors considered that management involvement in the control of irradiated fuel handling was poor.

O8 Miscellaneous Operations Issues

O8.1 Missing Main Steam Safety Valve Lockwires

a. Inspection Scope (71707)

On March 11, during rounds of the auxiliary building, a BGE non-licensed auxiliary building operator noticed a lead seal and lockwire on the floor in the Unit 2 main

steam isolation valve (MSIV) room. The lockwire and seal were determined to have come from one of the main steam safety (pressure relief) valves. The inspectors reviewed the BGE evaluation of the issue and corrective actions taken.

b. Observations and Findings

Upon discovery of the lead seal and lockwire, the auxiliary operator reported the finding to the control room. Operators inspected all the main steam safety/relief valves (SRVs) in the room (the lead seal and locking wire had been found under one of them) and found that, of the eight valves, the seals and lockwires were either missing or broken on four (2RV-3993, 2RV-3994, 2RV-3995, and 2RV-3996). An inspection of the Unit 1 MSIV room revealed no discrepancies with the seals or lockwires on those SRVs.

Each main steam safety valve had a protective cap over the lift/reseat setpoint adjustment mechanism. The cap was screwed in place and secured with a cotter pin. The lockwire with lead seal was threaded through the cap and attached to the valve body after the setpoint was set/adjusted to provide a tamper seal.

BGE system engineers began an immediate investigation to determine the operability of the four SRVs and requested the standby assistance of security personnel should tampering be indicated. One of the missing lockwire/lead seals was located and sent out for metallurgical analysis. Based on the following, BGE concluded that the four SRVs were operable and tampering was not a causal factor.

- The protective cap was fully seated on the valve and the attaching cotter pin appeared to be the original and showed no evidence of double bending;
- The cotter pin on the top nut appeared to be the original and showed no evidence of double bending;
- The gap between the manual lifting device and the top nut was consistent on the four valves with intact seals;
- The special tools needed to adjust the lift setpoint were found intact in their locked tool box, and;
- The results of the metallurgical analysis indicated that the most probable failure mechanism was fatigue of the lockwire due to vibration.

Unit 2 began a refueling outage on March 14 and BGE scheduled two of the four suspect SRVs for testing during the shutdown. BGE informed the inspectors that should either of the two tested SRVs perform unexpectedly, the other two would also be tested. On March 15, both SRVs tested satisfactorily.

BGE's long term corrective actions were being developed when the inspection period ended, and consideration was being given to using a different material for the lockwire.

The inspectors noted that BGE Nuclear Security had recently hired an individual to provide special investigative skills to the site. Although not used in this instance, the inspectors considered this a noteworthy enhancement in BGE's ability to appropriately respond to potential tampering events. Also, BGE informed the inspectors that an interfacing procedure between Operations and Security was being developed to formalize when and how security assistance would be provided.

c. Conclusions

BGE responded appropriately to the discovery of missing/broken lockwires on four Unit 2 safety relief valves. The operability evaluation was thorough and reasonable and the followup SRV testing during unit shutdown for refueling validated the BGE conclusions.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Inservice Inspection Activities

a. Inspection Scope (73753)

The inspectors observed the preparation and conduct of the inservice inspection (ISI) of the Unit 2 steam generators and other pressure components.

b. Observations and Findings

The BGE inservice inspection program plan for the second ten-year interval was approved by the NRC in a January 11, 1990 letter. BGE adopted the American Society of Mechanical Engineers (ASME) Code, Section XI, 1983 Edition with Addenda through summer of 1983, as the basis of the program for both units of the plant. The inspector noted that all the changes in, and modifications to the program, were properly documented and incorporated. The latest change to the plan was "Change G," incorporated in November 1996.

The personnel engaged in non-destructive examinations were properly qualified and were certified to proper levels of inspection/examinations in the different examination methods, including Liquid Penetrant, Magnetic Particle, and Ultrasonic Tests. There were four BGE inspectors, and seven contractor (NES) inspectors engaged in ISI work. The qualification and certification of VT-inspectors was in accordance with the requirements of SNT-TC-1A, as mandated by ANSI/ASME N45.2.6-1978, which is accepted by Code case N-424. All other personnel, except eddy current test analysts, were qualified/certified in accordance with the requirements of the Code. The Eddy Current Test (ECT) analysts were qualified and certified according to BGE procedure MN-3-105, "Qualification of NDE Personnel and Procedures," which required that personnel analyzing steam generator (S/G) eddy-current data be certified in accordance with the current revision of

EPRI TR-106589, PWR Steam Generator Examination Guidelines, Appendix G (commonly referred to as QDA) document.

The inspector reviewed the nondestructive examination procedures used by BGE and found the procedures clearly written with adequate explanation of the technical basis and work control.

BGE had contracted Framatome to acquire and analyze eddy current data for the Unit 2 Steam Generators. The vendor's procedures used for data acquisition and analysis were reviewed and approved by BGE. The data acquisition equipment used at the site was a Zetec system, consisting of hardware and computers with software for data acquisition and analysis. The inspection method consisted of "Bobbin Coil" and "Rotating Coil" (Plus Point) inspections. A summary of the inspection plan was provided to the NRC in a November 13, 1996 meeting which was summarized in a December 11, 1996 letter from the NRC to BGE. The inspector observed the data gathering and analysis activities and found them effective.

The BGE ECT procedure provided for an independent review of the eddy current data. This independent review was performed by a separate and independent contractor (Zetec). During the inspection there were no disagreements between the production and independent data reviews.

At the conclusion of the steam generator inspection, 277 tubes in 21 steam generator and 214 tubes in 22 steam generator required plugs. This made a total of 704 tubes plugged (8.3%) in 21 steam generator and 443 tubes plugged (5.2%) in 22 steam generator. Both steam generators were Category C-3 according to Technical Specification 3.4.5, due to greater than one percent defective tubes. In accordance with the technical specifications, BGE intended to notify the Region I NRC Administrator of the results. On April 9, 1997, BGE briefed the NRR staff regarding the status of the SG eddy current inspections.

c. Conclusions

The inspector found that the BGE inservice inspection program was well implemented, and the steam generator tube eddy current inspection was very well planned and executed. Specifically, the steam generator inspection indicated an emphasis on safety, preplanning, and training of personnel for an effective inspection and data analysis. BGE determined both steam generators to be Technical Specification Category C-3 due to greater than one percent of the tubes being defective.

M1.2 Routine Surveillance Observations

The inspectors witnessed and reviewed selected surveillance tests to determine whether approved procedures were in use, details were adequate, test instrumentation was properly calibrated and used, technical specifications were

satisfied, testing was performed by qualified personnel, and test results satisfied acceptance criteria or were properly dispositioned.

The inspectors noted that surveillance testing was performed safely and in accordance with proper procedures. The inspectors also noted that an appropriate level of supervisory attention was given to the testing depending on its sensitivity and difficulty. Surveillance testing activities that were reviewed are listed below:

STP-0-92-2 Refueling Machine Auxiliary Hoist Functional Test
STP-O-29-1 Monthly CEA Test

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) URI 50-317&318/96-03-01: Disposition of Unexplanded Steam Generator Tubes

The unresolved item involved the design acceptability of a number of steam generator tubes that were not fully rolled (explanded) into the steam generator tubesheet. The condition was due to errors in the manufacturing process. The tubes had been identified by BGE during past steam generator tube inspections and BGE maintained a list of the affected tubes. Each was inspected through the tubesheet region during the steam generator inspections. BGE provided the inspectors a March 1994 evaluation of the condition. The evaluation stated that only the weld between the tube and the tubesheet was credited as the code pressure boundary and in the tube stress analysis. The tube expansion was not credited but was preferable only to minimize tube corrosion. Based on the design evaluation, the inspector considered the unresolved item closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Generic Letter 89-10 Motor-Operated Valve Program Review (T/I 2515/109)

Introduction and Purpose

On June 28, 1989, the NRC issued Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," requesting licensees to establish a program to ensure that switch settings for safety-related motor-operated valves (MOVs) were selected, set, and maintained properly. Seven supplements to the generic letter have been issued to provide additional guidance and clarification. NRC inspections of licensee actions implementing the provisions of the generic letter and its supplements have been conducted based on the guidance provided in NRC Temporary Instruction 2515/109, "Inspection Requirements for Generic Letter 89-10," which is divided into Part 1, "Program Review," Part 2, "Verification of Program Implementation," and Part 3, "Verification of Program Completion."

The NRC conducted the Part 1 inspection at Calvert Cliffs Nuclear Power Plant (CCNPP) in August 1991 as documented in NRC Inspection Report (IR) 91-81. A Part 2 inspection, conducted in April 1994 was documented in NRC IR 94-17. The purpose of this inspection was to verify completion of the Generic Letter 89-10 program at CCNPP in accordance with Part 3.

E1.2 Summary Status of Generic Letter 89-10 MOVs

a. Inspection Scope

Generic Letter 89-10 requested that licensees notify the NRC in writing within 30 days after the MOV design-basis reviews, analyses, verifications, tests, and inspections have been completed. In a letter dated June 2, 1994, Baltimore Gas and Electric (BGE) clarified their commitments to complete the generic letter program at CCNPP. Specifically, for Unit 1, actions would be completed by the end of Refueling Outage 12 (Spring 1996). For Unit 2, actions would be completed by the end of Refueling Outage 11 (Spring 1997). In a letter dated September 3, 1996, BGE provided their final response to the generic letter. In addition to these BGE documents, the inspectors reviewed Plant Engineering Section Guideline PEG-15, "Motor Operated Valve Program," dated February 2, 1997, and other related documents associated with all MOVs in the Generic Letter 89-10 program. Using these documents, a valve sample was selected that included examples of all methods used by BGE in the generic letter program to demonstrate design-basis capability.

b. Observations and Findings

The BGE generic letter program scope consisted of 92 MOVs of which 38 valves (14 gates and 24 globes) were dynamically tested. BGE methods for demonstrating MOV design-basis capability included verification by: 1) valve-specific dynamic test, at or near, design-basis conditions, 2) valve-specific test, linearly extrapolated to design-basis conditions, 3) in-plant or industry information obtained from dynamic tests on similar MOVs, and 4) Electric Power Research Institute (EPRI) performance prediction model (PPM) data applied to MOVs that were not practicable to test. The inspectors reviewed special test packages and engineering evaluations for the following selected MOVs.

1/2-MOV-0403	Power Operated Relief Valve Block Valve
1/2-MOV-0405	Power Operated Relief Valve Block Valve
1/2-MOV-4516	Steam Generator Feedwater Isolation Valve
1/2-MOV-4517	Steam Generator Feedwater Isolation Valve
1-MOV-0627	High Pressure Safety Injection Loop Isolation Valve, Alternate Header 11B
2-MOV-0660	Safety Injection to Refueling Water Tank Minimum Flow Return Isolation Valve

The inspector reviewed changes to the MOV program scope that had been made since NRC Inspection Report 50-317&318/94-17. The safety injection tank

isolation MOVs were removed from the BGE generic letter program on the basis that they were not required to change position to support a system safety function. These MOVs are administratively controlled by deenergizing them with their feeder breakers open when the safety injection tanks are required to be operable for a design basis event. Based on the guidance in Generic Letter 89-10, Supplement 1, this is an acceptable basis for excluding these valves from the MOV program.

c. Conclusions

BGE dynamically tested 38 of 92 motor operated valves in the Generic Letter 89-10 program. The inspectors concluded that the exclusion of the safety injection tank isolation MOVs from the Calvert Cliffs program was acceptable.

E1.3 MOV Sizing and Switch Settings

a. Inspection Scope

The inspectors reviewed documents and calculations that established the thrust requirements for MOVs in BGE Generic Letter 89-10 program. The purpose of this review was to assess BGE justifications for assumptions used in MOV thrust calculations which form the basis for determining the design-basis requirements. The documents that were reviewed included thrust calculations and test evaluation packages associated with the selected MOVs. BGE methods for determining minimum thrust requirements were documented in Attachment 19, Calculation No. CA03474, "Unit 1 Generic Letter 89-10 Motor Operated Valve Thrust Calculation," Rev. 0, dated February 21, 1997. White Paper "Differential Pressure Test Statistical Analysis for Valve Factor, Rate of Loading and Coefficient of Friction," dated March 21, 1996, contained the basis for BGE program assumptions.

b. Observations and Findings

BGE thrust calculations typically utilized the standard industry equations. Orifice seat diameter was used to calculate valve seat area. Valve factors were based on in-plant test results or from other industry sources as specified by BGE methodology. A stem friction coefficient of 0.20 was used for determination of actuator output thrust capability. BGE applied margin to account for diagnostic equipment uncertainty, torque switch repeatability, load sensitive behavior, and potential valve degradations.

Valve Factor and Grouping

BGE had divided MOVs into 22 valve groups based on manufacturer, type, size, and ANSI pressure class rating. BGE attempted to use in-plant data for justification of valve factors for non-dynamically tested MOVs. However, BGE did not have sufficient in-plant test results to adequately cover many gate valve groups. To address this, the BGE White Paper included analyses of the valve factor results from 8 gate valve tests. After reviewing these data, BGE determined that a 0.7 valve factor was justified based on a 95% confidence band of the available valve factor

data. The inspectors noted that BGE analysis included some gate valves where they were unable to obtain adequate valve factor results. For these valves the analysis assumed a 0.0 valve factor. Since the quality of the test results was poor and produced invalid 0.0 valve factors, the inspectors did not consider the use of this information to be acceptable. When the 0.0 valve factors were removed, the analysis mean increased but the variation in the data decreased and resulted in the same 0.7 valve factor. The inspectors concluded that the BGE general assumption of a 0.7 valve factor for gate valves at Calvert Cliffs was reasonable based on the industry dynamic testing experience. However, BGE analysis did not establish specific valve factors for the many different size and pressure class gate valves which was especially important for non-dynamically tested Generic Letter 89-10 valves. The inspectors noted that this area could be improved by continuing to gather in-plant and industry information as part of the BGE MOV program to establish an applicable valve factor basis for the non-tested valves.

BGE applied the EPRI PPM to provide thrust requirements for the containment sump suction valves which are Velan 24 inch flex-wedge gate valves. BGE's calculation evaluated the applicability of using the EPRI PPM based on criteria contained in Section 4 of the calculation. A review of this section revealed that valve size was not a consideration. After review of the NRC Safety Evaluation by the Office of Nuclear Reactor Regulation of Electric Power Research Institute Topical Report TR-103237, "EPRI Motor-Operated Valve Performance Prediction Program," dated March 15, 1996, the inspectors noted that EPRI considers the gate valve model limited to valves that are a maximum of 18 inches. However, due to the lack of industry information for large Velan gate valves, the inspectors considered the PPM results to be the best available data at this time. Based on the available margin and the use of PPM-based thrust requirements, the inspectors considered the current settings to be adequate for Generic Letter 89-10 program closure. However, the inspectors noted that BGE had not established an adequate long-term basis for these MOVs which could be achieved by obtaining additional information (e.g., justify use of the EPRI PPM or apply other applicable industry data), as part of the BGE long term MOV program.

Although BGE personnel had informally reviewed and were familiar with the NRC safety evaluation of Electric Power Research Institute Topical Report TR-103237, "EPRI Motor-Operated Valve Performance Prediction Program", they had not documented this review of the safety evaluation for information that may affect BGE use of the EPRI PPM. BGE agreed to officially document this review. The lack of review of industry information important to the use of EPRI PPM was identified as a weakness in the BGE generic letter program.

BGE thrust calculations for globe valves used a standard gate valve equation to determine minimum required thrust. BGE explained that the gate valve equation was chosen for simplicity and for conservatism. Given the BGE method of accounting for stem rejection, irrespective of the globe valve flow direction or stroke direction, the inspectors agreed that use of this equation would result in conservative minimum thrust requirements. However, the inspectors also noted that the gate valve equation was used to back calculate valve factors from globe

valve dynamic test results. In these cases, incorrectly subtracting out stem rejection forces resulted in nonconservative valve factor determinations. BGE revised their globe valve dynamic test determinations using a standard globe valve equation and agreed that the revised valve factors fell within the range of expected values. BGE personnel were considering use of the standard globe valve equation for the purpose of evaluating globe valve testing.

Load Sensitive Behavior

The BGE White Paper documented an analysis of eight gate valve data points which determined that a 15% load sensitive behavior margin was justified based on a 95% confidence band of the available load sensitive behavior data. Given the small amount of in-plant data, the inspectors concluded that BGE load sensitive behavior justification for non-dynamically tested MOVs could be improved by obtaining additional information as part of their long term MOV program.

Stem Friction Coefficient

The BGE White Paper documented an analysis of eight gate valve data points which determined that a 0.20 stem friction coefficient assumption was justified based on a 95% confidence band of the available stem friction coefficient data (under dynamic test conditions). Given the small amount of in-plant dynamic test data, the inspectors concluded that the stem friction coefficient justification for non-dynamically tested MOVs could be improved by obtaining additional information to increase the confidence in this assumption as part of their long term MOV program.

Degradation Margin

BGE included a 5% margin in the valve minimum required thrust to address age-related valve degradations. Results from the BGE long term MOV program will be used to revise this 5% margin if necessary. The inspectors found this approach to be acceptable for Generic Letter 89-10 program closure.

c. Conclusions

In general, the BGE assumptions of valve factor, load sensitive behavior, and stem friction coefficient were considered conservative and acceptable for Generic Letter 89-10 program closure. However, due to the small amount of in-plant dynamic test data obtained, the inspectors concluded that the BGE basis for these assumptions for non-tested valves could be improved by obtaining additional information as part of their long term MOV program. BGE efforts in this area was an inspector followup item **(IFI 50-317&318/97-02-07)**.

E1.4 Design-Basis Capability

a. Inspection Scope

The inspectors reviewed capability assessment packages and associated test reports for the selected MOVs. The purpose of this review was to assess BGE efforts to establish design-basis capability for all MOVs in their Generic Letter 89-10 program.

b. Observations and Findings

The NRC review of the BGE MOV program focused on low margin valves. Of particular concern were the power operated relief valve (PORV) block valves. The inspectors also had comments regarding the steam generator feedwater isolation valves and the safety injection pump to refueling water tank minimum flow return isolation valves.

PORV Block Valves - 1/2-MOV-403 & 405

The PORV block valves are 2.5 inch Velan 1500# solid wedge gate valves equipped with a Limitorque SMB-000 actuator using a five foot-pound motor. The inspector noted that the UFSAR information in Table 4-17 indicated that the PORV block valves were 2500 pound class valves. BGE stated that the UFSAR information was incorrect. Issue report IR1-011-376 was initiated on March 6, 1997, to correct the UFSAR. Since BGE considered that the PORV block valves were very similar to EPRI valve number 13, the valve materials used at CCNPP were reviewed and clarified as follows:

Valve body	=	316 Stainless Steel/ASTM A479
Valve seats	=	316 Stainless Steel/ASTM A479 with Stellite 6 hardfacing
Wedge	=	Stainless Steel Alloy XM19 with Stellite 6 hardfacing at seating surfaces
Wedge guide	=	Stainless Steel Alloy XM19 - no hardfacing
Guide rail	=	316 Stainless Steel/ASTM A479 - no hardfacing

The original valve design included gearing to produce a 10-second stroke time. However, as a result of several modifications, the current stroke time was approximately 45 seconds . Also, the original valve design was a flex wedge gate but it was modified several years ago to a solid wedge design to increase the valve's open loading capability.

Design-Basis Differential Pressure

BGE had identified 2098 psid as the design-basis differential pressure for the PORV block valves. This was based on an analysis that used 2300 psid as a starting point, and then using operator response time and valve stroke time as a basis for determining the worst-case differential pressure for these valves. BGE assumed a 15-second operator response time and a 45-second valve stroke time. Further, the

analysis determined the plant pressure when the PORV block valve was half way through the 45-second valve stroke (22 seconds). Therefore, the plant pressure was determined at 37 seconds after plant pressure decreased below the analysis starting point of 2300 psid. Given that the PORV reset pressure was approximately 2365 psig, the inspectors were concerned that the use of this approach was inadequate because it did not consider the possibility of an operator error resulting in PORV block valve closure at pressures above 2098 psid.

Valve Factor

BGE assumed the use of a 0.3 valve factor based on the apparent valve factor at flow isolation obtained from EPRI valve number 13 that was tested as part of the Performance Prediction Program (PPP). Valve 13 was a 2.5 inch Velan 1500# flex-wedge gate valve that was of a new design not yet used by the nuclear industry. The inspectors noted that the Calvert Cliffs PORV block valves were of a solid wedge design. Therefore, the EPRI valve was not an exact match to Calvert Cliff's valves. Further, BGE did not select a conservative valve factor from the EPRI test. The correct valve factor from the EPRI test would be at least 0.34 based on the seating portion of the thrust trace. The inspectors noted that selecting valve performance parameters from a different valve at flow isolation is highly subjective and valve specific. Also, EPRI tested valve 13 under blowdown conditions only one time. This single data point does not provide any insights into the potential for variation in MOV performance under high temperature blowdown flow conditions. Therefore, this single test was not considered to be adequate for generic letter program closure.

BGE was applying the mean-seat based EPRI valve factor obtained from valve 13 to PORV block valves which had thrust requirements that were based on the use of the valves' orifice diameters. Due to BGE use of a smaller disc area term, this resulted in the nonconservative application of the EPRI valve factor data.

The NRC reviewed the EPRI PPM software as documented in the safety evaluation that was referenced in Section E1.3 of this report. The NRC endorsement of the PPM (with the conditions stated in the safety evaluation) only covers the use of the PPM software. The safety evaluation does not accept use of the PPP individual valve factors. Because of the above concerns, the inspectors did not consider BGE's use of EPRI's valve 13 test results to be appropriate for justification of a 0.30 valve factor for the PORV block valves.

Available Margin

BGE contacted the valve vendor and obtained correct seat dimensions so that a mean-seat diameter could be used to determine the valve disk area for the thrust calculations. This allowed direct usage of valve factor data from the EPRI valve 13 test. Assuming 15% margin for load sensitive behavior, 10.3% for diagnostic equipment uncertainty and torque switch repeatability, and actual packing loads, the available valve factors for the PORV block valves are provided below. The values are slightly different due to individual valve performance including variations in the

existing torque switch settings. (Note: BGE will take credit for full motor capability after modifications are implemented to bypass the torque switch for the entire stroke.) These values do not include any margin for degradation.

Available Valve Factor at 2098 psid		Available Valve Factor at 2300 psid	
1-MOV-403	0.58	1-MOV-403	0.51
1-MOV-405	0.54	1-MOV-405	0.46
2-MOV-403	0.46	2-MOV-403	0.38
2-MOV-405	0.40	2-MOV-405	0.34

Conference Calls of March 7 and 19, 1997

Conference calls occurred on March 7 and 19, 1997, during which the NRC and BGE personnel discussed the technical aspects of the BGE operability determination of March 6, 1997, where they considered the PORV block valves to be operable. BGE also included in these discussions the short and long term plans being taken to improve the design basis capability of these valves. The lead inspector requested these calls to address the following issues: (1) the BGE basis for using a design basis differential pressure of 2098 psid; (2) the BGE technical basis for applying the specific valve factor information of EPRI valve 13 test to the PORV block valves; (3) the BGE efforts regarding the use of other industry data, including the use of the EPRI developed PPM software, to support valve setup and design basis capability.

During the call of March 7, 1997, it was apparent to NRC personnel that BGE had developed a systematic approach for defining the differential pressure to be used as a basis for operability. However, upon review of past correspondence in light of this effort, it was also evident that BGE had decreased the differential pressure design requirement without substantially changing the equipment capability. For example, in 1991 (see NRC Inspection Report 50-317&318/91-81) the NRC reviewed BGE's design-basis differential pressure determination for the PORV block valves and accepted the basis of 2300 psid. This was considered a reasonable differential pressure given the guidance in emergency operating procedure EOP-0 that specified operator action to ensure that the PORV was closed as pressure decreased below 2300 psi. More recently (see BGE "Final Response to NRC Generic Letter 89-10; Safety-Related Motor-Operated Valve Testing and Surveillance, dated September 3, 1996) BGE identified the PORV block valve design-basis differential pressure as 2282 psid. This value was based on the maximum expected normal operating pressure of the reactor coolant system.

At the conclusion of the call of March 7, 1997, BGE agreed to (1) actively pursue the possibility of using the EPRI PPM software to predict the PORV block valve thrust requirements; (2) contact another nuclear plant which recently used the EPRI PPM software concerning its PORV block valves; and (3) verify the various materials utilized in the PORV block valves. BGE responded to these issues during the second week of the inspection and the conference call of March 19, 1997. As documented

in an internal memorandum, dated March 13, 1997, to the Plant Manager, BGE agreed to the following planned actions regarding 1/2-MOV-403&405:

- The PPM will be run on 2-MOV-403&405 during the upcoming refueling outage. Actual internal dimensions will be obtained for both valves prior to running PPM to help in the evaluation.

- The torque switch bypass modification will be accomplished during the current Unit 2 refueling outage.

- Pending the results of the PPM, BGE will continue to evaluate the results to ensure that they have acceptable margin. If the results do not provide significant margin for the long-term, BGE will investigate modifications which may include replacement of the operators for Unit 2 in 1999 and Unit 1 in 1998.

- For Unit 1, BGE will implement the torque switch bypass modification in 1998. This work was verified to be on the confirmation work list for any shutdown. All of this was contingent on BGE not obtaining something from the Unit 2 evaluation that would cause them to take more prompt action.

Toward the end of the inspection, the inspectors identified an issue which involved BGE actions in response to a past leakage problem of 1-MOV-403. The leakage problem was experienced in April 1994 and was corrected in April 1996. The inspector had the following concerns: (1) this problem had not been mentioned by BGE in any of the earlier discussions concerning PORV block valve operability; and (2) it was unclear how BGE had evaluated the degraded conditions of the valve and concluded that the other 3 PORV block valves were not affected. The inspector's concerns were discussed during the conference call of March 19, 1997.

BGE had performed a root cause analysis in August 1996 regarding the leakage problem of 1-MOV-403 and concluded that improperly machined travel guides, that is poor maintenance work practices, contributed to the valve leakage. Upon review of the leakage experienced (approximately 8 gallons per hour), BGE also concluded that the safety implications of the problem were minimal. In light of this minor leakage, BGE considered the other three PORV block valves to be acceptable without further evaluation. BGE also concluded that the leakage problem encountered in 1994 and the repairs made in 1996 did not impact the functionality of 1-MOV-403 or the other PORV block valves. However, BGE initiated an Issue Report on March 20, 1997, to further evaluate the maintenance performed on 1-MOV-403 for its implications.

During the conference call of March 19, 1997, the inspector requested from BGE a copy of the 1-MOV-403 maintenance work records completed during April 1996 under Maintenance Order (MO) #1199402257. Upon review of this documentation, the inspector observed the following remarks made by maintenance personnel on April 13, 1996: "The sides of the guide slots are hitting the guide rails." After lapping the valve seats, the maintenance personnel entered the following remarks on April 22, 1996: "Wedge seemed to tilt when it got near the bottom. Inspected

guides and found that weld on lower part of guide was protruding into guide area. Wedge is hitting weld and pulling it out of the seat". This interference was eventually corrected by a combination of machining and lapping the valve seats and wedge. In light of these valve conditions, the inspectors questioned the BGE assumption that this valve's performance was comparable to EPRI valve 13. Furthermore, it was questionable whether 1-MOV-403 would have functioned under design basis conditions if it had been required to do so subsequent to April 1994 and prior to the repairs of April 1996. BGE had not performed an operability evaluation in April 1994 or April 1996 to assess the valve's functionality in its degraded state. The inspectors also noted that BGE MOV project, design, or component engineering personnel made no specific attempts during the April 1996 maintenance activity to confirm that the valve condition matched favorably to EPRI valve #13. Ample opportunity was available for such a review as evidenced in MO #1199402257 where extensive machining was required to accomplish proper seating of the valve internals. Also, as evidenced by the need for clarifying the valve materials and understanding the solid versus flex wedge designs during the inspection, it was apparent that BGE had not completed a detailed mechanical comparison review of EPRI valve 13 and the PORV block valves. Yet several months later, BGE in the Generic Letter 89-10 closure report used EPRI valve 13 as the basis for justifying the design basis capability of the PORV block valves.

Given these technical concerns and the low thrust margins available, the inspectors concluded that the PORV block valves' current switch settings were not adequate for Generic Letter 89-10 program closure. In light of the information available during the last several years regarding the marginal capability of the PORV block valves, the inspectors also concluded that BGE corrective action measures taken regarding these risk significant valves during the last several years have been poor and untimely. As discussed in section E7 of this report, BGE MOV Project Self-assessment Report 93-02 identified in 1993 potential design basis capability concerns for gate valves with low (0.3) assumed valve factors. Also, BGE performed corrective maintenance in April 1996 on 1-MOV-403 without fully evaluating the impact of these repairs on past operability or the extent of these degraded conditions as applied to the other PORV block valves. This is a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. **(VIO 50-317&318/97-02-08)**

<u>Steam Generator Feedwater Isolation Valves</u> - <u>1/2-MOV-4516 & 4517</u>

These valves are 16 inch Velan flex-wedge gates with Limitorque SMB-2 actuators. BGE used the EPRI PPM to determine the valve minimum required thrust. The inspectors observed that valves 1-MOV-4516 and 2-MOV-4517 had low available thrust margins of 0.6% and 2.9% respectively. The inspectors also identified the following comments during the review of the switch settings for these valves.

The PPM calculations used a design-basis differential pressure of 275 psid. This was obtained from BGE Calculation M-93-182, "Max. Line and Differential Pressure Valves 1 and 2-MOV-4516 and 4517 May Experience During Operation," dated December 8, 1993. It was noted that this calculation evaluated the closing

conditions for the feedwater isolation valves when a feed line break occurs without a steam generator isolation signal. Under this scenario, the maximum differential pressure would be 636 psid. The analysis concluded that this differential pressure could be lowered if the operating procedures were revised to deenergize the condensate booster pumps and condensate pumps prior to manually closing the feedwater isolation valves. The inspectors reviewed Abnormal Operating Procedure 3G, "Response to a Condensate or Feedwater Rupture" and verified that all pumps are deenergized prior to closing the feedwater isolation valves. Therefore this issue was resolved.

The design-basis differential pressure of 275 psid for the feedwater isolation valves was based on the condensate pump discharge pressure while assuming no contribution from the feedwater pump which was stopped within 30 seconds (approximately one half of the feedwater isolation valves' stroke time). The inspectors requested confirmation of feedwater pump coast down time. In response, BGE provided a plot of pump pressure and speed over time that was measured during an operational trip of the pumps. A review of this data indicated that the pump was still providing pressure beyond 30 seconds after the pump was secured.

The inspectors recognized that BGE has planned electrical modifications to substantially improve the design basis capability for these MOVs. However, given the uncertainty related to the identified design-basis differential pressure and the low margins involved, the inspectors requested BGE to reassess the acceptability of the use of the design basis differential pressure value of 275 psid and the current torque switch settings for these valves. The issue will be followed up by inspector followup item **(IFI 50-317&318/97-02-09)**.

<u>Safety Injection Pump to Refueling Water Tank Minimum Flow Return Isolation Valves</u>

The available thrust margins for these valves ranged between 3.3% to 9%. Also the inspectors noted that the thrust calculations used valve factors in the range of 0.20 to 0.30 for these MOVs. BGE personnel explained that these valves were dynamically tested under near design-basis conditions. However, a restricting orifice located between the safety injection pump discharge and the min-flow return isolation valves reduced the volumetric flow rate to such a low level that personnel were unable to determine a valve factor from the dynamic test results. Further, BGE stated that the effect of the orifice was not accounted for in the original design-basis review for these valves. Therefore, the design-basis differential pressure has been specified as overly conservative for the pressure that actually exists at the valves when they are stroked close during pump operation. While the inspectors had reasonable assurance that the valves would function based on the dynamic tests that were performed, BGE was requested to establish an acceptable valve factor basis and to resolve any margin concerns for these valves. BGE agreed to review this matter which will be followed by inspector followup item **(IFI 50-317&318/97-02-10)**.

c. Conclusions

The inspectors found that BGE had adequately addressed MOV design-basis capability for the majority of the valves in the Generic Letter 89-10 program. However, the inspectors identified design basis capability issues regarding several MOVs. Also, some minor issues regarding the steam generator feedwater isolation valves and safety injection pump to refueling water tank minimum flow isolation valves require further BGE evaluation.

E1.5 (Closed) URI 317&318/94-17-02 Pressure Locking and Thermal Binding

a. Inspection Scope

The inspectors reviewed the evaluation of gate valves susceptible to pressure locking and/or thermal binding which BGE had completed in response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." In its letters dated February 13 and July 25, 1996, BGE identified valves that were susceptible to pressure locking and/or thermal binding and specified corrective actions.

b. Observations and Findings

BGE's generic letter submittals stated that the Grand Gulf pressure locking analytical method was utilized to demonstrate that the actuators on the following Units 1 and 2 valves could develop adequate thrust to overcome pressure locking:

MOV-504	Charging pump suction from refueling water tank
MOV-514	Boric acid pump discharge
MOV-4144&4145	Emergency sump recirculation
MOV-4516&4517	Steam generator feedwater isolation
MOV-6900, 6901 & 6903	Containment hydrogen purge
MOV-2080	Instrument air containment isolation

The inspectors reviewed actuator capability and Grand Gulf analytical calculations for these valves and identified the following concerns:

1. BGE did not use Generic Letter 89-10 disk coefficient of friction values in pressure locking thrust calculations.

2. BGE used an increase of 0.4 psi/°F for thermally-induced pressure locking calculations. Industry testing has revealed that this is not conservative in that testing has identified that bonnet pressure increases at a higher rate than that used by BGE.

3. BGE developed a method to calculate maximum closing force for limit-switch seated valves to use in pressure locking calculations. This calculational method for determining maximum closing thrust was not validated by a test program.

4. Pressure locking calculations for MOV-4144 and 4145 specified that the emergency sump recirculation piping was dry. The margin between required thrust and actuator output thrust was minimal. BGE now maintains this piping full of water. Pressure locking calculations for this condition should be revised to reflect this change and the margin between required thrust and actuator thrust evaluated for long term pressure locking corrective action. Also, the effects of leakage from the safety injection system into the bonnets of MOV-4144 and 4145 through check valves SI-4148 and 4149 and manual isolation valves SI-132 and 144 during shutdown cooling, surveillance or other modes of operation should be evaluated.

5. The margins between required thrust and actuator output thrust for valves MOV-4516, 4517 and MOV-6900, 6901, 6903 were minimal for long-term pressure locking corrective action.

6. The BGE analytical method to determine pressure locking thrust for MOV 2080 was not validated by a test program.

The BGE Generic Letter 95-07 submittals stated that the shutdown cooling isolation valves (1&2-MOV-651&652) and the PORV block valves (1&2MOV-403&405) were susceptible to thermal binding. The inspectors reviewed BGE corrective actions to preclude thermal binding and found them to be adequate. This involved testing to verify that seating/unseating forces were minimal or testing during thermal binding conditions to evaluate any increased opening forces.

The adequacy of the Grand Gulf pressure locking analytical method and BGE actions to address pressure locking remain under NRC evaluation. During the current inspection, the inspectors raised concerns regarding pressure locking analytical parameters and margins for long term pressure locking corrective action. These issues must be resolved in order for the NRC staff to issue a safety evaluation for BGE's response to Generic Letter 95-07. This was identified as inspector followup item **(IFI 50-317/318/97-02-11)**.

c. Conclusions

The inspectors concluded that thermal binding corrective actions were thorough. While no discrepancies were identified in actuator capability calculations, an inspector follow item was opened to identify additional tasks needed for the final resolution of BGE pressure locking corrective actions. Previous URI 50-317&318/94-17-02 is closed.

E1.6 MOV Failures, Corrective Actions, and Performance Trending

a. Inspection Scope

Item (h) of Generic Letter 89-10 requested licensees to include a monitoring and feedback effort in the MOV program to establish trends in MOV operability. The inspectors reviewed BGE's trending practices as described in Appendix TR, MOV

Trending Program, of Plant Engineering Section Guideline PEG-15, "Motor Operated Valve Program."

b. Observations and Findings

PEG-15 requires that the MOV Component Engineer provide a periodic report which documents the review, evaluation, and determination of any adverse trends in the performance of MOVs at Calvert Cliffs. The inspectors reviewed the first report issued for the time period of January 1994 to August 1996. The information in this report met the general intent of Generic Letter 89-10 regarding tracking and trending the performance of MOVs. The inspectors specifically noted that one of the significant MOV issues included in this report was an inservice failure of 1-MOV-4516 which was attributed to a motor pinion key problem.

In addition to reviewing the trending report, the inspectors requested information on those MOVs where corrective maintenance had been performed in 1996 and 1997. During the review of several of these maintenance activities, the inspectors identified some concerns regarding maintenance performed on PORV block valve 1-MOV-403 which was discussed in Section E1.4 above. With specific regard to the trending program, the inspectors noted that BGE did not fully evaluate the impact of these maintenance activities and leakage problems on the valve performance at design-basis conditions.

c. Conclusions

BGE established an adequate MOV trending program as recommended by Generic Letter 89-10. However, the inspectors concluded that the program was not effective in fully evaluating the performance of 1-MOV-403 in light of a recent leakage problem.

E7 Quality Assurance in Engineering Activities

a. Inspection Scope

The inspectors reviewed the following independent and self assessments conducted of the BGE MOV program.

- MOV Project Self Assessment Reports 93-02 dated June 17, 1993 and 94-01 dated March 21, 1994.

- Quality Assurance Audit 96-03 of MOV Activities.

b. Observations and Findings

The inspectors noted that the earlier self assessments, more so than the recent Quality Assurance Audit 96-03, assessed the major elements of the MOV program. These self assessments were led by the cognizant MOV project manager. The inspectors also noted that the BGE MOV program did not have the benefit of a

critical independent assessment of the MOV program by knowledgeable personnel who were not permanently associated with BGE.

BGE Self Assessment Report 93-02 identified potential problem areas regarding (1) the use of low (0.3) valve factors; (2) the need for validating the assumed valve factors on non-tested valves; and (3) evaluation of current MOV setups and accommodation of higher valve factors. The inspectors noted that the PORV block valve design basis capability issue, which was extensively reviewed during this inspection, was a specific example of these previously identified licensee concerns and was not yet fully resolved.

c. Conclusions

BGE performed adequate self-assessments in developing the MOV program but had not yet fully resolved previously identified design basis capability concerns regarding the PORV block valves.

E8 Miscellaneous Engineering Issues

The inspectors updated or closed the following items which had been identified in past MOV program inspections.

E8.1 (Closed) URI 50-317&318/94-17-01: Design-basis differential pressure for 1/2-MOV-653 and 1/2-MOV-655. The Part 2 inspection noted that 0 psid was identified as the design-basis differential pressure for valves 1/2-MOV-653 and 1/2-MOV-655. This pressure was used instead of a higher differential pressure that was based on a valve mispositioning scenario. Subsequent to the Part 2 inspection, Supplement 7 to Generic Letter 89-10 was published which removed the need for pressurized water reactors licensees to consider mispositioning scenarios. The inspectors reviewed the design-basis reviews for these valves and verified that the worst-case non-mispositioning differential pressure for these valves was 0 psid. Therefore, the inspectors considered this issue closed.

E8.2 (Update) Inspection Report 50-317&318/94-17: Develop procedures for extrapolation of test results. The Part 2 inspection report noted that the BGE test procedures did not specifically require the use of extrapolation techniques when dynamic test conditions were less than design-basis conditions. BGE revised diagnostic test procedure MOV-009 to include a requirement to perform a linear extrapolation when test conditions are less than design-basis conditions. However, the inspectors noted that MOV-009 did not provide any guidance for performing the extrapolation and did not provide a location in the procedure where the calculation could be documented and reviewed. BGE personnel agreed to revise MOV-009 to allow proper documentation of test extrapolations. Inspector followup item **50-317&318/97-02-12** will track this item to closure.

E8.3 (Closed) Inspection Report 50-317&318/94-17: Install limiter plates on Unit 2 MOVs. The Part 2 inspection report noted that limiter plates had not been installed on Unit 2 MOVs. Subsequent to the Part 2 inspection, BGE installed limiter plates

to prevent exceeding actuator torque limits. Therefore, the inspectors considered this issue closed.

E8.4 <u>(Closed) Inspection Report 50-317&318/94-17</u>: Identify design-basis flow rates. The Part 2 inspection report determined that BGE design-basis reviews did not determine the flow rates that correspond to the identified design-basis differential pressures. In response to this concern, BGE personnel reviewed their design-basis documents and determined the worst-case flow rates for all MOVs in the BGE Generic Letter 89-10 program. Therefore, the inspectors considered this issue closed.

E8.5 <u>(Closed) Inspection Report 50-317&318/94-17</u>: Use of hydrostatic pump test results. The Part 2 inspection report questioned the use of hydrostatic pumps as a pressure source for dynamic tests. BGE reviewed the issue and decided to not take credit for dynamic tests that were performed using hydrostatic pumps. Therefore, the inspectors considered this issue closed.

E8.6 <u>(Closed) Inspection Report 50-317&318/94-17</u>: Develop dynamic test procedures for Unit 2 MOVs. The Part 2 inspection determined that dynamic test procedures had not been developed for Unit 2 MOVs. Subsequent to the Part 2 inspection, these procedures were completed. Therefore, the inspectors considered this issue closed.

E8.7 <u>(Closed) Inspection Report 50-317&318/94-17</u>: Periodic actuator refurbishments. The Part 2 inspection report questioned the BGE intent to not perform periodic actuator refurbishments. BGE has subsequently decided to perform periodic actuator refurbishments on four MOVs in high temperature applications (Main Steam Isolation Bypass Valves). Further, additional MOVs may receive periodic actuator refurbishments based on results from the long term MOV program. The inspectors considered this issue closed.

E8.8 <u>(Closed) Inspection Report 50-317&318/94-17</u>: MOV trending program development. The Part 2 inspection report identified this area as an MOV program weakness and was left open pending the development and review of a performance trending program. Based on the review discussed in Section E1.6 above, this issue is closed.

E8.9 <u>Improved Technical Specifications</u>

BGE submitted their proposed Improved Technical Specifications to the NRC on December 4, 1996, and requested approval by June 30, 1997. BGE has established a schedule to implement the improved technical specifications on August 12, 1997. The inspectors reviewed the BGE Improved Technical Specifications Implementation Project Plan and attended an Implementation Task Force Meeting. The task force consisted of a representatives from the key groups that were participating in the plans to implement the improved technical specifications. The task force was appropriately focused on coordinating

implementation issues such as procedure revisions, training, and self-assessment of the implementation processes.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Unit 2 Outage Radiological Controls (Program Changes)

a. Inspection Scope (83750)

The inspector reviewed selected radiological controls program changes since the previous inspection in this area. Areas reviewed included organization and staffing, facilities and equipment, and procedure changes.

b. Observations and Findings

BGE changed its practices for control of worker access to outage radiological work areas (e.g., containment areas) since the previous outage inspection (April 1996). The principal change involved the removal of health physics control points from the containment to reduce personnel exposure. The inspector made the following observations relative to this change.

- Although BGE implemented enhanced camera coverage and use of remote control access doors to high radiation areas, the inspector noted that workers could sign in on their radiation work permit (RWP) and readily enter the containment with little contact with radiation protection personnel. Further, it was readily possible for workers to sign-in on invalid or revised radiation work permits and re-enter the radiological controlled area without realizing that their RWP requirements may have changed.

- The inspector observed radiation protection personnel authorize access of an individual to a high radiation area (by use of a remote controlled door monitored by a camera) without a clear identification of the individual. Subsequent inspector review indicated no clear expectations, regarding identity verification prior to entry authorizations, were promulgated to personnel acting as remote high radiation area access controllers.

- The inspector was concerned that an individual could enter a radiation area without proper dosimetry because there was no independent verification.

BGE acknowledged these concerns and committed to re-evaluate this change to ensure that radiological control of personnel is not diminished or compromised.

c. Conclusion

BGE removed health physics control points from the containment. Although BGE expects this to reduce overall radiation protection personnel aggregate exposure,

this action appears to have reduced the overall effectiveness of worker access controls to radiological controlled areas.

R1.2 Unit 2 Outage Radiological Controls (Planning and Preparation)

a. Inspection Scope (83750)

The inspector selectively reviewed the planning and preparation for the Unit 2 refueling outage. The inspector reviewed records, discussed outage planning with licensee representatives, and observed activities to verify necessary preparations and management support for radiation protection planning. The inspector selectively reviewed work that had the potential for creating radiological hazards (e.g. reactor coolant pump work, refueling, and steam generator work activities).

b. Observations and Findings

BGE provided overall very good planning and preparation for outage radiological controls work activities and implemented new outage planning programs to improve work planning and control. Radiological controls personnel were well aware of planned work activities, including emergent work. Planning programs provided for various reviews of planned work including a one day and three day look ahead. Planned upcoming work was brought to the attention of in-field radiological controls personnel. Of particular note was the excellent planning and preparation for addition of peroxide to the reactor coolant system to enhance system crud removal and the excellent water movement planning to reduce personnel radiation exposure to as low as is reasonably achievable (ALARA). BGE's chemistry staff indicated between about 600 - 800 curies of radioactivity was removed from the reactor coolant system (RCS). BGE installed additional high radiation area controls and real-time radiation monitors to monitor and control access to piping systems expected to exhibit elevated radiation levels. Further the planning effectively minimized unnecessary discharge of waste water containing elevated radioactivity concentrations.

BGE issued a Unit 2 1997 pre-outage report which discussed numerous ALARA actions/initiatives for the outage and demonstrated a very good understanding of planned work and its radiation exposure consequences for the worker population. Lessons learned from previous outages were effectively incorporated into outage planning. BGE effectively used engineering controls to minimize the need to use respiratory protection equipment.

During in-field observations of on-going work, the inspector noted workers to be sensitive to unplanned radiation exposure. For example the inspector noted actions by workers to effectively coordinate reactor cavity and bridge work during cavity draindown to minimize unnecessary radiation exposure.

At the time of the inspection, the inspector noted that BGE had recently joined an international organization for the collection and review of worldwide radiation exposure data for nuclear power stations. BGE was performing benchmarking of its

typical aggregate radiation exposure for various work tasks against that presented for similar facilities. BGE had initiated actions to visit selected similar facilities which exhibited clearly lower aggregate exposure in the work task/function of interest. Further, as a result of the benchmarking efforts, BGE planned ALARA initiatives for activities involving the refueling path, steam generator work, snubber work, scaffolding, shielding, and RCS chemistry.

c. Conclusions

BGE implemented overall effective ALARA planning and implementation of ALARA plans for the Unit 2 refueling outage. BGE initiated actions to benchmark its ALARA program performance against similar facilities.

R1.3 Unit 1 Outage Radiological Controls (Internal Exposure Controls)

a. Inspection Scope (83750)

The inspector selectively examined the internal exposure control program. The inspector reviewed records, discussed the program with cognizant licensee personnel and observed internal exposure control practices during observation of work activities and tours of the RCA.

b. Observations and Findings

The inspector noted air sampling to be representative of air in workers' breathing zones. Engineering controls were used to reduce ambient airborne radioactivity. Radiation protection personnel closely monitored activities with the potential to generate airborne radioactivity (e.g., reactor cavity draindown).

As a result of concerns generated during self-assessment activities and previous NRC questions in the area of calibration and checking of air sample counting instruments used for analysis of potential alpha airborne radioactivity, BGE generated a report, dated February 24, 1997, entitled 1997 Radiological Air Sampling Issues. Based on the inspector's preliminary review of this document, the inspector questioned the bases for BGE's weighted alpha derived air concentration (DAC) and BGE's conclusions relative to alpha attenuation attributable to filter loading. The inspector noted that the use of the weighted DAC did not appear to be consistent with the requirement of 10 CFR 20.1204. Further, the use of inappropriate filter loading factors could impact the filter analysis results. The inspector will review these matters during a future inspection. The adequacy of BGE's air sampling and analysis program is an unresolved item **(UNR 50-318/97-02-13)**

The inspector noted that BGE provided whole body counting for divers. The inspector noted that tritium bioassays were typically not performed. The license indicated this matter would be reviewed.

c. Conclusions

BGE implemented a generally effective internal exposure control program.

R1.4 Unit 2 Outage Radiological Controls (External Exposure Controls)

a. Inspection Scope (83750)

The inspector selectively examined BGE's general external exposure control program. The inspector reviewed records, discussed the program with cognizant licensee personnel and observed exposure control practices during observation of work activities and tours of the RCA. The inspector reviewed high radiation area controls and general radiological posting, reviewed implementation of the radiation work permit program, and reviewed implementation of the dosimetry program.

b. Observations and Findings

External Exposure Controls (General)

The inspector noted that in general, radiological areas (e.g., high radiation areas, radiation areas) were properly posted and locked. BGE increased the health physics staff to support outage work and used BGE radiation protection technicians in lead capacities over contracted technicians to provide coverage of work activities.

The inspector's review of a caged and locked high radiation area on the Unit 2 refueling floor (reactor vessel head laydown area) and a caged and locked high radiation area in the lower annulus area of the Unit 2 containment (fuel transfer tube area) indicated personnel may be able to access the areas by circumventing the enclosures. The inspector noted that subsequent to construction of the enclosure area at the vessel head laydown area, scaffolding and a scaffolding ladder were positioned near the reactor head lay down area enclosure. Subsequent BGE review indicated access would be unlikely but the ladder was repositioned and the under vessel area itself was locked thereby precluding unauthorized entry. Inspector review indicated the design of the fuel transfer area enclosure allowed for the enclosure to be easily scaled by climbing over the area or walking into the area via overhead piping. BGE evaluated the area and determined that radiation dose rates in excess of 1000 millirem/hr did not exist. Consequently, the area was not required to be locked. BGE reviewed all enclosures for locked high radiation areas, including postings, found no similar concerns, and initiated guidance regarding acceptable standards for enclosures/barricades for locked high radiation areas.

In general workers were provided briefings as required by applicable radiation work permits and 10 CFR 19.12, and were observed to be generally wearing dosimetry as prescribed. Some workers were wearing their electronic dosimeters (ELD) away from the thermoluminiscent dosimeter (TLD) which could result in a mismatch between TLD and ELD dose values. BGE's radiation protection staff counseled the individuals.

The inspector observed preparation for entry into the Unit 2 reactor cavity following drain down of the cavity. Licensee personnel performed comprehensive surveys to support planned entries, identified a 30 R/hr hot spot, and took action to vacuum up the spot prior to worker entry.

The inspector observed exposure control methods for Unit 2 steam generator inspection personnel. The inspector noted that the platform areas were continuously monitored by television cameras when personnel were in the areas, and access to the areas was controlled through locked doors. Further, BGE provided real-time teledosimeters to monitor radiation dose rates and integrated exposures received. The teledosimetry readouts were continuously monitored by personnel. The inspector observed that the dose rate alarm was set at 5000 millirem/hr. This was considered high considering the generally low radiation dose rates present on the platforms. BGE initiated a review of this matter.

The inspector noted weaknesses in the high radiation area key control procedures. Specifically, BGE updated the key inventory multiple times during a week when changing temporary locks and chains. There was no apparent procedure guidance for updating the key inventory log. Further, the inspector's review identified two instances where the key log did not reflect the status and location of temporary locks.

Although overall external exposure controls were considered good, exceptions were noted as discussed below.

High Radiation Area Access Controls for Unit 2 Containment Emergency Airlocks (EALs)

On February 16, 1997, a licensee radiation protection technician identified, via review of the radiation protection log book, that the lock for the Unit 2 Containment Emergency Airlock (EAL) may not have been transferred from the outer handwheel of the Unit 2 containment outer EAL handwheel to the outer access door of a small building enclosing the outer EAL door. The lock is normally transferred in preparation for a personnel entry into containment to allow personnel egress from containment, via the EAL, in an emergency. Rather the lock was left on the floor of the small building enclosing the outer EAL. Further, the lock serves as the control for high radiation area access control purposes.

The inspector's review indicated that, based on inspector discussions with individuals involved and review of applicable security logs, the high radiation area lock was not in place on the door of the small building door enclosing the outer Unit 2 EAL for approximately four hours on February 16, 1997. The matter was identified by a radiation protection technician during review of a log book entry and the lock was subsequently applied to the EAL door. The inspector noted that the outer door of the small building was locked by a security lock, no personnel had made unauthorized entries into the containment, and no unplanned radiation exposures occurred.

The following observations were made.

- The shift radiation protection technician identified on February 16, 1997, that the Unit 2 EAL lock may be mis-positioned based on a review of a questionable log entry for that day.

- The shift technician dispatched an in-field radiation protection technician to the area. The technician found that the lock was on the floor.

- The lock was not in place on the outer door from 11:30 a.m. to 3:00 p.m. on February 16, 1997.

- The technician subsequently placed the lock back on EAL door. (Inspector Note: As discussed below, the lock was to have been placed on the door of the small building. Since the lock was placed back on the EAL door, this action may have prevented egress of personnel from the containment in the event of an emergency.)

- There was no formal guidance regarding movement of the high radiation area lock from the outer EAL door to the door of the small building indicating weaknesses in high radiation area access control.

- A guidance memorandum on the proper procedure for locking the area was distributed on February 25, 1997. Personnel were trained on the guidance memorandum.

- A special tag was added to the Unit 1 and Unit 2 EAL high radiation area door keys to alert personnel to review the guidance memorandum when using the keys to ensure proper movement of locks from the outer EAL doors to the door of the small buildings as appropriate.

- The access to the small building was continuously locked by security; however, personnel could request security personnel for access.

The inspector noted that Technical Specification 6.4.1, "Procedures", requires, in part, that BGE establish, implement, and maintain the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33 recommends, in Section 7.e., "Radiation Protection Procedures", that procedures be established for access control to radiation areas. Licensee radiation protection procedure RSP 1-104, "Area Posting and Barricading", Revision 10, requires in Section 6.5, "Locked High Radiation Area", that areas exhibiting radiation levels in excess of 1000 millirem at 30 centimeters from the radiation source be provided with a locked barrier or ensure that the area is provided with continuous direct or electronic surveillance that is capable of preventing unauthorized entry. The inspector noted that areas inside the Unit 2 Containment exhibited radiation levels greater than 1000 millirem at 30 centimeters and that the access door was not provided with the above specified access controls.

The following additional observations were made.

- The radiation protection technician who had not properly installed the lock normally worked in the radioactive material handling group and was not familiar with the expectations regarding movement of the lock. Consequently it appeared possible that personnel outside the radiation protection operations group (e.g., radwaste handling personnel) could perform activities normally assigned to operational radiation protection personnel, without the benefit of training or understanding of all applicable instructions and guidance (e.g., those promulgated by the Radiation Protection Supervisor) prior to performing activities associated with those instructions and guidance. The inspector noted there were approximately 23 such memoranda or special instructions written for a variety of topics. Further, the inspector questioned if such memoranda were appropriate in consideration of the need for appropriate station approved procedures as required by Technical Specification 6.4.1. BGE initiated a self-assessment of its training program for personnel who do not normally work in the field operations group and who may be called upon to act as a member of that group.

- Radiation protection personnel apparently initiated two issue reports following identification of the failure to install a high radiation area locks. The first involved a safety matter associated with ensuring that the chain lock was removed from the outer EAL door prior to personnel entering into the containment. The second involved the matter discussed above involving access control to high radiation areas. The inspector was able to view a copy of the issue report involving the safety matter but could not locate any copy of the incident report for the high radiation area issue despite radiation protection assurance that such an incident report was initiated. The inspector noted that as of April 9, 1997, both original incident reports were lost and had not been properly entered into the BGE corrective action program. As a result, BGE initiated actions to re-issue the Issue Reports and also filed an Issue Report regarding the loss of the two original Issue Reports.

Regarding the issue associated with the safety matter, the inspector noted that, based on discussions with radiation protection personnel and review of the copy of the issue report for the safety matter, the high radiation area lock for the Unit 2 outer EAL door was re-installed on the Unit 2 EAL airlock instead of the small building enclosing the Unit 2 EAL. Consequently, in the event an individual needed to egress the containment, the EAL would not function. The lock was apparently on the handwheel of the outer Unit 2 EAL from about 3:00 p.m. on the February 16, 1997 until about 9:00 a.m. on February 18, 1997, when a technician placed the lock on the door of the small building enclosing the EAL.

The inspector noted that no entries were made into the Unit 2 containment during the time period. The inspector noted that BGE's Procedure No. 1-104, "Containment Access", Revision 3, contained a pre-entry checklist but the checklist did not contain a check to verify the status of the locks on the EALs prior to entry into containment. This was considered a weakness. As discussed above, BGE did

issue a radiation control guideline regarding proper positioning of the locks based on the status of personnel entries into containment. In addition appropriate personnel were trained on the guideline including radwaste radiation protection technicians who may act as operational radiation protection personnel.

BGE established Procedure No. QL-2-100, Revision 5, Issue Reporting and Assessment, to provide guidance for issuance of Issue Report (IRs). The procedure indicated that an issue report was used to document an actual or suspected condition adverse to quality or a significant condition adverse to quality and provides a method for notifying affected groups and initiating corrective actions. The inspector noted that the procedure required in Section 4.7, that reviewing supervisors ensure issue reports are received by the issues assessment unit group within three working days of being initiated. The inspector noted as discussed above, that issue reports were written in February 1997 for the high radiation area access control concern and the safety concern associated with locking and control of Unit 2 containment emergency airlocks. The reviewing supervisor was aware of the issue reports; however, neither report was provided to the Issues Assessment Group for cause and corrective action determination.

The failure to control access to the Unit 2 containment, a high radiation area, and the failure to document the problems in the corrective action system were in the aggregate, an apparent violation of NRC requirements. **(EEI 50-318/97-02-14).**

c. Conclusions

The inspector's infield reviews indicated BGE implemented a generally good external exposure control program. However, as discussed above, problems in the guidance and training of personnel on use and positioning of high radiation area locks was identified. Further, when the problems were identified, issue reports were not completed.

R1.5 Unit 2 Outage Radiological Controls (Control of Diving Activities)

a. Inspection Scope (83750)

The inspector selectively reviewed BGE's diving activities associated with the malfunction of the Unit 2 fuel transfer system on April 3, 1997. The malfunction involved inability to transfer fuel from the Unit 2 reactor to its spent fuel pool. (See Section O2.2 of this report for particulars on the malfunction). The inspector noted that as a result of the malfunction, a total of five dives were conducted in the Unit 2 refueling cavity and spent fuel pool to inspect and repair the fuel transfer equipment. During the fourth dive activity on April 3, 1997, and unknown to three radiation protection technicians who were to be providing continuous coverage of the diver, the diver (Diver A) left the presurveyed dive location at the south end of the Unit 2 spent fuel pool and traveled about 15 feet to the unsurveyed north end of the pool to inspect the drive cable for the fuel transfer system. This movement resulted in the diver entering elevated radiation fields emanating from irradiated reactor spent fuel and receiving an unplanned radiation exposure. The inspector

reviewed the diving activities, in particular the fourth dive activity, relative to guidance provided in Technical Specifications, 10 CFR 20, and the following NRC information and guidance documents.

- NRC Information Notice No. 82-31, "Overexposure of Diver During Work in Fuel Storage Pool", dated July 28, 1982

- NRC Information Notice No. 84-61, "Overexposure of Diver in Pressurized Water Reactor (PWR) Refueling Cavity", dated August 8, 1984

- NRC Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants", dated June 1993, Appendix A, Procedure for Diving Operations in High and Very High Radiation Areas

b. Observations and Findings

The inspector noted that three dive activities (some involving multiple water entries) were conducted on March 31 and April 2, 1997. Dive 1 and 2 were conducted in the spent fuel storage pool and dive 3 was conducted in the reactor cavity. One diver (Diver A) completed these three dive activities, was continuously monitored by a television camera and was provided continuous coverage by radiation protection personnel. The diver did not sustain any unusual radiation exposures based on TLD results.

The diver was provided with multiple radiation protection dosimetry including multiple thermoluminescent dosimeters (TLDs) for the wrists, head, chest, back, and thighs (above the knee) and feet. Real-time radiation survey meters (telemeters) for monitoring the wrists, thighs (above knee), chest, and back were also provided. The meters provided real-time radiation dose rates and integrated radiation doses at their specific locations. The detectors were set to alarm at an integrated radiation dose of 100 millirem and a real-time radiation dose rate of 895 millirem/hr. The diver was also provided with two radiation survey meters attached to an approximately 30 inch shaft. The diver surveyed various locations with the meters. The diving activities were controlled by RWP No. 97-2316 and a job standard (No. 18, Diving Maintenance in the Refuel Pool and Spent Fuel Pool).

All radiation survey meters read out on the surface and were monitored by a radiation protection technician. The diver did not receive any direct readout of the radiation survey data except that provided over an intercom relayed to dive support personnel in communication with the diver by intercom.

The inspector noted that the fourth dive activity was planned for April 3, 1997, and Diver A was to complete that dive activity also. In preparation for this dive activity a pre-job meeting was held early in the morning of April 3, 1997. All dive support personnel attended this activity, including engineering personnel, the diver and dive support personnel, and the three radiation protection technicians that were to provide coverage of this activity. The dive work activity was discussed including the applicable radiation work permit and job coverage standard. Task 8 of the dive

activity task list indicated that a kinked cable was to be visually inspected near the stinger end of the fuel transfer mechanism located at the north end of the Unit 2 spent fuel pool for indications of damage if dose rates permit. The kink was located at the north end of canal when the carriage is in the spent fuel pool between the stinger and the fuel rack. At the main pre-job meeting, radiation protection supervisory personnel indicated a check would need to be performed to ensure that radiation survey data had been collected to support dive Task 8.

Subsequent to completion of the main pre-job meeting (which included radiation protection personnel responsible for dive operations), the radiation protection supervisor informed the night shift dive activity engineering support individual that Task 8 was not to be performed because radiation surveys had not been made to support this activity in the north end of the pool. The inspector noted that this information was passed on by the night shift engineering support personnel to the day shift dive engineering support personnel (who provided engineering support for dive four). However, the day shift dive activity engineering support individual did not inform dive personnel of the fact that Task 8 was not to be performed due to lack of radiation surveys. Consequently, the diver and dive support personnel (including the dive tender) were not informed that diver movement to the north end of the spent fuel transfer system, within the Unit 2 spent fuel storage pool, was not approved.

(Inspector note: To support inspection of the cable at the north end ("stinger end") of the fuel transfer system, the diver would need to traverse approximately 20 - 30 feet outside the surveyed dive location at the south end of Unit 2 spent fuel pool, pass within approximately 4 to 8 feet of recently discharged irradiated fuel, traverse unsurveyed portions of the spent fuel pool floor, and come within close proximity of unsurveyed portions of the spent fuel transfer system which had recently transferred irradiated fuel elements. Further, the floor of the pool had not been vacuumed recently in the area to be traversed. The inspector noted that the diver did not have real time radiation survey instruments for the lower extremities (i.e., feet).)

The inspector noted that in preparation for the dive activity, the lead radiation protection technician briefed the diver (Diver A) as to the expected radiation doses rates within the dive location. Due to significant communications problems between the diver and the lead radiation protection technician, including misunderstanding by the diver of the radiation survey documents, the diver believed he was being provided a complete briefing of the entire Unit 2 spent fuel pool fuel transfer area, including the north end of the transfer system. The inspector determined that the diver was actually provided a briefing only on the limited survey area at the work area in the south end of the pool as indicated by the radiation survey documentation (Unit 2 SFP upender area radiation survey, dated April 3, 1997, at 4:00 a.m.). Because the survey data sheet was enlarged, it gave the impression to the diver as being the entire pool. The inspector further noted that the radiation protection technician did not show the diver an accompanying map that clearly depicted that the radiation survey information presented was only for the south end of the pool.

The inspector noted that the diver entered the spent fuel pool at about 9:00 a.m. that day and commenced work on a limit switch on the upender at the south end of the pool. No problems were encountered and the diver requested some tie wraps for the work activity. While waiting for the tie wraps the diver (Diver A) felt he had time to inspect the kinked cable at the north end of the spent fuel pool unaware that he was prohibited from traveling to the north end of the pool. As indicated previously, the diver was unaware that a comprehensive radiation survey of the area had not been performed. The diver informed personnel on the surface that he was going to inspect the cable. The radiation protection personnel on the surface believed the diver was referring to the cable in his immediate work area. The diver support personnel were also not aware that the individual was prohibited from performing the work activity at the north end of the pool. Consequently, the dive communicator acknowledged the diver, and the diver proceeded to the north end of the pool. The dive tender provided additional cable and hose, as needed.

As the diver proceeded to the north end of the pool, out of the comprehensively surveyed area, he performed general scanning measurements with his duel probes (0-2 R/hr and 0-200 R/hr) which were taped together as a 30 inch wand believing that the area he was traversing had already been surveyed. The diver scanned the floor area, surveyed to his right, left, above and also in front of him. The diver stopped at grid location approximately 85, placed his survey meters on the floor of the pool and ascended about 15 feet, without his meters, to check a cable. The diver was about three feet from the cable. The diver submerged again, retrieved his survey meters, and continued to proceeded to the north end of the pool. The diver spent an estimated 30 seconds performing the inspection. The dive tender, believing the activity was authorized, continued to provide cable and breathing air line to the diver.

The radiation protection technician monitoring the teledosimetry systems and hand held survey meter results, noted that the diver's right wrist radiation monitor alarmed and indicated 3R/hr then increased to 9 R/hr. The diver had stopped to survey a pipe (later determined to be the spent fuel pool cooling and clean-up suction line) at the north end of the spent fuel pool. The left wrist monitor also alarmed and indicated 2.3 R/hr. The technician directed, through communication with the dive communicator, that the diver retreat to a low dose rate area. The technician did not note a high reading on the hand-held probe and believed it read about 250 millirem/hr.

The inspector noted that the technician monitoring the real time radiation dose rate and dose integrating instruments at the surface believed the diver had encountered a hot spot at the south end of the pool at his specified work location. The technician was unaware that the diver was actually at the north end of the pool in relatively close proximity to irradiated fuel. The technician conferred with the lead radiation protection technician for the task and the shift lead radiation protection technician who was observing the activity outside the area. The shift lead radiation protection technician authorized the diver to re-enter the area and re-survey the location for hot particles unaware that the diver was actually at the north end of the pool and had entered underwater radiation fields generated by irradiated fuel. The

diver was given permission to survey the location of interest. The diver entered the area keeping his survey meter in front of him when the 0 - 2 R/hr probe went offscale and the 0-200 R/hr probe indicated about 3 R/hr. The real time radiation survey meters attached to the individuals dive suit indicated a maximum radiation dose rate of 35 millirem/hr. When the elevated radiation dose rates were encountered, the diver was directed to exit the pool.

A second radiation protection technician was observing the diver, at his work location at the south end of the pool, through a viewing glass placed on the surface of the pool. However, because of air bubbles surfacing, the technician did not have a clear view of the diver. Rather, the technician was observing bubbles. Further, because the diver was approximately 40 feet underwater, the bubbles took time to reach the surface. By the time the last bubbles had reached the surface, from where the diver was last located, and no further bubbles were evident, the technician realized that the diver was not immediately under him and the diver had apparently moved toward the north end of the pool based on observed bubbles at that location.

After the diver surfaced, the radiation protection personnel became aware that the diver had gone to the north end of the pool. The diver informed the technicians where he had been in the pool. The diver indicated that he had been about 15 feet outside the surveyed area. The diver had spent approximately 52 minutes in the water and was estimated to have spent about 1 to 1.5 minutes at the north end of the pool, of which 30 seconds was estimated to have been consumed during the surfacing activity to check a cable. The inspector noted that the diver had traversed about 15 feet outside the comprehensively surveyed areas.

As normally practiced, the diver's dosimetry package was pulled and was to be held for five hours before reading. The package was to be read after 5 hours in order to avoid short term fade effects when reading the TLDs.

Prior to processing the dosimetry for Diver A, BGE decided to initiate a fifth dive to complete the repair/inspection. A second diver (Diver B) was selected for this task. Since a dive team consisted of at least 3 individuals and only three individuals from the dive team were available, including Diver A, BGE decided to estimate the dose of Diver A, return his normal whole body TLD badge, and allow the diver to re-enter the spent fuel storage pool work area to support Diver B. BGE believed the diver had not received an exposure in excess of its administrative limits for either the whole body (4 rem administrative limit), or extremities (40 rem extremity limit). Also, the radiation dose rates at the work area for personnel supporting the diver was low (less than 0.2 millirem/hr). Consequently, BGE believed that any additional exposure would be negligible.

Prior to performing the fifth dive, and as a result of the problems with dive 4, BGE ensured all personnel performing and supporting the dive activity were well aware of the dive location and expected work scope. Further, BGE placed a camera in service to continuously monitor the diver (Diver B). Dive 5 was completed without any exposure control problems.

The inspector's review of the circumstances surrounding the event and BGE's response identified the following apparent violations.

Instructions to Workers

10 CFR 19.12 requires that all individuals working in or frequenting any portion of a restricted area be kept informed of the storage, transfer, or use of radioactive materials or of radiation in such portions of the restricted area.

The inspector noted that on the morning of April 3, 1997, BGE did not adequately inform a diver, diving in the Unit 2 spent fuel storage pool, of the storage of radioactive materials or of radiation in the spent fuel pool that the diver may encounter. Specifically, the instructions provided to the worker failed to adequately provide the extent of radiation surveys made to support the diver's work in the south end of the spent fuel pool and failed to adequately limit the scope of work performed by the diver commensurate with the extent of those radiation surveys. Further, BGE failed to instruct the diver as to the location of irradiated fuel assemblies. As a result, the diver moved from the approved dive area at the south end of the Unit 2 spent fuel pool (that had been comprehensively surveyed) traversed an unsurveyed portion of the spent fuel pool, travelled to the north end of the spent fuel storage pool, and entered high radiation fields caused by radiation streaming from irradiated spent fuel elements. Further, radiation fields readily accessible to the diver exceeded 500 rads/hr and the diver had not been provided any specific training on limitations or capabilities of the radiation survey instrumentation provided (e.g., response time).

Failure to provide adequate instructions to a diver who unknowingly entered an unsurveyed portion of the spent fuel pool on April 3, 1997, is an apparent violation of 10 CFR 19.12. **(EEI 50-317&318/97-02-15)**.

Access Control to Very High Radiation Areas

10 CFR 20.1602 requires that in addition to the requirements in 10 CFR 20.1601, BGE shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads or more in an hour at one meter from a radiation source or any surface through which the radiation penetrates. 10 CFR 20.1601 provides general requirements for control of access to high radiation areas and requires, in part, that in lieu of physical controls and alarms specified therein, a licensee may substitute continuous direct or electronic surveillance that is capable of preventing unauthorized entry. The inspector noted that BGE provided a viewing glass on the surface of the water to provide continuous surveillance of the diver who entered the Unit 2 spent fuel pool on April 3, 1997. BGE also provided multiple teledosimetry for the body and provided the diver with two independent radiation survey meters placed on a 30 inch wand.

The inspector noted that despite the presence of three senior radiation protection technicians monitoring the dive activity and the radiation monitoring systems

provided the diver in accordance with 10 CFR 20.1601, BGE's additional controls, as required by 10 CFR 20.1602 were ineffective in that they did not ensure personnel were not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads or more in an hour at one meter from a radiation source or any surface through which the radiation penetrates.

In addition, the following was noted.

- BGE's continuous surveillance of the diver with a viewing glass failed to detect that the diver 1) had left the approved dive zone, 2) dropped his hand held survey meter to perform visual inspections in an area outside the approved dive zone, 3) entered an unsurveyed area in close proximity to irradiated fuel, and 4) was directed to re-enter the same unapproved area to re-survey for elevated radiation readings. Further, the technician monitoring the activity through the viewing glass indicated that his view of the diver was obstructed by bubbles, and that he could not directly see the diver.

- The diver was provided no specific licensee training on the use, limitations, or capabilities of the survey instrumentation provided, particularly the hand-held survey meter used to detect elevated radiation fields away from the body. The diver performed general scanning surveys as he traversed the distance to the north end of the spent fuel pool with a survey meter (RO-7) that required 2.5 seconds to reach 90 % of actual radiation dose rates. The RO-7 was not provided with setable alarm setpoints. Rather the monitor would alarm (red light) if radiation dose rates were encountered greater than the detector's design. Further, the whole body monitoring teledosimetry system had a transmit time of two seconds.

- The area entered and traversed by the diver was not comprehensively surveyed prior to the diver's entry therein. Consequently, as the diver traversed the areas he believed he was traversing recently surveyed areas.

- The diver was provided inadequate training on the location of irradiated fuel assemblies and no physical barriers were provided to warn of their location. Further, the pool entry point was not posted as a high or very high radiation area.

- The diver is believed to have inadvertently entered (knuckles of right hand) areas with radiation dose rates in excess of 500 rads/hr. BGE's calculation indicated the diver's knuckles may have been in a radiation field of about 550 rads/hr (assuming a 6 second exposure)

- Radiation dose rates within the spent fuel pool exhibited significant gradients. For example licensee calculations and radiation surveys at 6 feet from the pool floor indicated greater than 20,000 rads/hr at 1.5 feet away from irradiated fuel, 12,000 rads/hr at 2.5 feet away from the irradiated fuel, 1,450 rads/hr 3.5 feet away from the irradiated fuel, and 559/rads/hr at 4.75 feet from the spent fuel.

The diver's head could have readily accessed radiation dose rates greater than 500 rads/hr.

- The inspector noted, that as a result of alarms on the wrist teledosimetry, the diver was directed to retreat from the area. However, he was subsequently directed to re-enter the area to perform additional surveys even though the technicians did not know the diver's location.

  The inspector noted that, based on the above, BGE's controls, required by 10 CFR 20.1602, to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads or more in an hour at one meter from a radiation source or any surface through which the radiation penetrates, were inadequate. This was an apparent violation of 10 CFR 20.1602. **(EEI 50-318/97-02-16)**

The inspector made the following additional observations.

Procedures

The inspector's review indicated the following.

- BGE did not have a station approved procedure for the diving operations. Rather a job coverage standard was used to provide radiological controls guidance.

- The job coverage standard specified (Section 6 D.) that an underwater camera and/or viewing box shall be used to maintain sight of the diver. The inspector noted that a camera was not used for the dive activity. However, although a viewing box was used, the technician monitoring the divers location could only see bubbles and did not maintain sight of the diver.

- The job coverage standard specified (Section 4.0 B) that all entrances, exits, calculated stay times and telemetry readout (or equivalent) were to be logged on the job coverage record. The inspector reviewed the job coverage record for dive 4 and determined that the entrance, exit, calculated stay time, were not logged. Further, other dive coverage records indicated a similar lack of such specified information.

- An engineering plan ((Unit 1 and 2 Refueling Pools and Spent Fuel Pool Diving Plan) was established. The dive plan contained an Attachment 1, Project Manager Check List, and an Attachment 2, Spent Fuel Pool Coordinator Dive Check List. Despite five dives occurring in total in the spent fuel pool and the reactor cavity, check sheets were only completed for the first dive.

- BGE did not implement guidance and suggestions for control of diving operations that were previously promulgated to the industry (e.g., make divers clearly aware of irradiated fuel assemblies, provide effective continuous observation of divers, establish physical barriers).

In response to the event, BGE took the following actions.

- BGE initiated an Issue Report for the event.
- BGE initiated a human performance evaluation of the event.
- BGE also generated a deficiency report.
- BGE performed a dose assessment of the diver.
- BGE enhanced training of the diver and support personnel, on job scope and survey location for the fifth dive.

c. Conclusion

Although BGE satisfactorily completed three prior dives, BGE did not maintain effective control of the fourth dive into the Unit 2 spent fuel pool.

R1.6 Unit 2 Outage Radiological Controls (Dose Assessment/Diver)

a. Inspection Scope (83750)

As discussed in Section R1.5 of this report, a diver sustained an unplanned exposure when he travelled to the north side of the Unit 2 spent fuel pool. The inspector reviewed the calibration and checking of radiation survey instrumentation and the adequacy of personnel monitoring and surveys for the diver.

b. Observations and Findings

The inspector's review indicated BGE performed operability checks of the real-time radiation monitoring equipment used for the diving activities. The inspector reviewed energy response curves for the gamma survey instrumentation and noted that the instruments provided generally adequate response over the energy spectrum. The inspector noted that a degraded energy spectrum may be expected based on multiple compton scattering of gamma photons. The inspector noted the detectors provided adequate response to the expected spectrum. In addition, after the April 3, 1997, unplanned exposure event, BGE performed a comprehensive evaluation of the operability of the real-time radiation instrumentation used during the diving activity. BGE concluded the instrumentation and electronics, with the exception of a slight over response of the right wrist detector, were functioning as expected.

In light of the potential for significant exposure to the diver attributable to radiation doserate gradients, the license performed an exposure assessment for the diver. BGE concluded that no occupational exposure in excess regulatory limits occurred. BGE concluded that because of the large dose rate gradients, the dose to the right extremity (right knuckles) needed to be increased as compared to the wrist TLD reading. BGE calculated a maximum dose to the extremities (right knuckles) of 885 millirem as compared to a wrist TLD badge readout result of 424 millirem (shallow dose equivalent). BGE also increased the dose to the whole body due to higher dose rates predicted at the elbow as compared to the chest, head, back, and thigh TLD readout results. BGE calculated a whole body dose of 270 millirem as

compared to a maximum whole body TLD readout result of 137 millirem (head). The inspector noted the maximum dose the lower extremity (ankle) was 21 millirem (shallow dose equivalent)

At the end of the inspection period, BGE was continuing to review these dose results. The inspector questioned the potential neutron dose to the diver attributable to being within close proximity of irradiated spent fuel. The inspector also questioned the need to correct the estimated results using radiation buildup factors. BGE indicated neutron badges worn by the diver did not indicate neutron exposure and indicated doses were continuing to be reviewed but no significant change was expected (i.e., radiation dose rates were well within 10 CFR 20 limits).

Notwithstanding the above dosimetry results, the inspector noted that 10 CFR 20.1501 requires that licensee's make or cause to be made surveys that may be necessary to comply with the regulations in 10 CFR 20 and are reasonable under the circumstances to evaluate the extent of radiation levels and the potential radiological hazards that could be present. 10 CFR 20.1201 provides limits for occupational exposures including exposure limits of 50 rems to the skin or any extremity.

The inspector noted, as discussed above, that BGE did not provide adequate control of Diver A to prevent him from leaving the approved dive area and entering areas not comprehensively surveyed. The diver had real-time teledosimetry attached to the dive suit above his knees and on his chest and back. The diver also was supplied with two radiation survey meters which were taped together and enclosed in a 30 inch wand type handle. His extremities were monitored by real-time teledosimetry at the wrists. No real-time teledosimetry was provided for the lower extremities. The alarms on the teledosimetry were set at an integrated dose of 100 millirem dose and a dose rate of 850 millirem/hr.

Notwithstanding the real-time monitoring provided, the inspector concluded that BGE's surveys of potential radiation dose rates to the diver's lower extremities as he traversed the non-comprehensively surveyed areas in the north end of the pool and as he was redirected to re-enter the area at the north end of the pool were inadequate to ensure compliance with occupational dose limits for the lower extremities (feet) for the following reasons.

- The shielding effect of water on gamma radiation produces extremely large radiation dose rate gradients in very short distances requiring the performance of comprehensive radiation surveys to detect isolated intense radiation sources. For example, licensee calculations indicated that a dose rate of approximately 28,000 times that at the knee would need to be present at the location of the feet to cause an alarm of the real-time radiation dose rate meter located at the knees of the diver (i.e., knee dose rate meters set to alarm at 850 mR/hr). The inspector noted that if such a dose rate existed (i.e., 23,000 rads/hr) it would result in a dose of approximately 6.6 rads per second to the foot and an overexposure to the extremity in about 7.5 seconds. The feet were not provided with real-time radiation dose rates but were monitored with TLDs. Although

subsequent readouts of TLDs monitoring the feet indicated dose to the feet of less than 100 millirem (as compared to the 50 rem limit), as BGE's calculations indicate, the potential for significant external exposures in difficult to survey conditions was present.

- The diver traversed the non-comprehensively surveyed area under water to the north end of the pool in about 50 seconds (based on licensee estimates) and performed scanning measurements as he went under the belief that the area had been comprehensively surveyed. Although the diver consciously looked for potential radiation sources as he traversed the area, including at his feet, the inspector noted that, considering the traverse time, the potential large radiation dose rate gradients that could be present, and the survey meter instrument response time of 2.5 seconds, a potential intense isolated radiation source could have been present and not detected during the transit. Further, the inspector's discussions with the diver indicated that he had not been provided training on the capabilities and limitations of the survey meter (e.g., response time). Further, the diver encountered elevated radiation readings at the north end of the pool which would mask radiation dose rates attributable to isolated small intense sources of radiation.

- The area the diver traversed was not comprehensively surveyed prior to his entry and at least 20 irradiated spent fuel elements had been transferred through the area.

The inspector concluded that failure to perform reasonable surveys of the potential radiation dose to the lower extremities of the diver to ensure compliance with 10 CFR 20 was an apparent violation **(EEI 50-317&318/97-02-17)**

c. Conclusion

No exposure of the diver in excess of regulatory limits was identified.

R1.7 Unit 2 Outage Radiological Controls (Radioactive Materials and Contamination)

a. Inspection Scope (83750)

The inspector selectively reviewed radioactive material and contamination control practices. The inspector reviewed the adequacy of supply, maintenance, and calibration and performance checks of survey and monitoring instruments; and the use of personal contamination monitors and friskers, including consideration of hot particle contamination. The inspector also reviewed the adequacy of surveys to detect personnel exposure due to skin contamination, particularly for hot particle contamination.

b. Observations and Findings

The inspector's tours of the station, including the Unit 2 containment, indicated that BGE implemented generally effective contamination control work techniques. The

inspector noted contamination levels within the Unit 2 containment to be generally low.

Although no hot particles were identified by BGE during surveys of steam generator platforms, the inspector questioned the adequacy of the surveys for material removed from the steam generators and handled by personnel. Specifically, steam generator inspection personnel wiped down, handled, and surveyed the material and there was no specific training program relative to surveying the material. Further, inspector observations indicated radiation protection personnel, monitoring the survey activity by camera, were not always able to observe the adequacy of surveys and monitoring of objects by steam generator eddy current test personnel on the steam generator platforms

c. Conclusions

BGE implemented a generally effective contamination control program. However evaluation of the adequacy of surveys, performed by steam generator eddy current personnel, of material removed from steam generator needed improvement.

R5 Staff Training and Qualification in Radiation Protection and Chemistry

R5.1 Contractor Training

a. Inspection Scope (83750)

The inspector reviewed the training and qualification records of selected contractor radiological controls personnel. The inspector evaluated the training and qualification of these individuals relative to applicable Technical Specification requirements, procedural requirements and 10 CFR 50.120. The inspector reviewed training records, personnel resumes, and discussed qualification criteria with cognizant licensee personnel. The inspector selected contractors providing oversight of significant radiological work activities and reviewed their training and qualification.

b. Observations and Findings

Radiological controls personnel were qualified in accordance with applicable requirements. BGE implemented a generally well-defined training and qualification program for contracted radiological controls personnel providing responsible radiological oversight during the outage. Job coverage standards were established and implemented to provide guidance for radiological coverage of various work tasks. Personnel received job specific training on the job coverage standards (e.g., diving). Weaknesses were identified in the training of personnel as follows.

- As discussed in Section R1.4 of this report, a radiation protection technician, who had not properly installed a lock on an emergency airlock, was not familiar with the expectations regarding movement of the lock. It appeared that personnel outside the radiation protection operations group (e.g., radwaste

handling personnel) could perform activities normally assigned to operational radiation protection personnel, without the benefit of training or understanding of all applicable instructions and guidance. BGE initiated a review of this matter.

- As discussed in Section R1.5 of this report, a diver had access to radiation fields in excess of 500 rads/hr and had not been provided any specific training on limitations or capabilities of the radiation survey instrumentation provided (e.g., response time).

- As discussed in Section R1.7, steam generator eddy current personnel did not receive specific training on surveys for material coming out of steam generators.

c. Conclusions

BGE implemented a generally good program to train and qualify contractor radiological controls personnel providing radiological oversight of outage radiological work activities. However, weaknesses in training of personnel outside the operations radiation protection group were noted. Further a diver who entered the spent fuel pool was not provided instructions on the use of his radiation survey instrument commensurate with the potential hazard present.

R6 RP&C Organization and Administration

R6.1 Outage Radiological Controls Organization

a. Inspection Scope (83750)

The inspector selectively reviewed the outage radiological controls organization.

b. Observations and Findings

The inspector noted that BGE had developed and implemented an outage organization. Radiological controls personnel were assigned to various lead individuals for various areas of the station. The following weaknesses were identified.

- Due to significant organizational communications problems and ineffective control of work, work groups assigned to control diving activities in the Unit 2 spent fuel pool on April 3, 1997, did not effectively understand or control significant radiological controls work activities. As a result, a diver, unknown to responsible individuals controlling the activities, travelled from a comprehensively surveyed area and entered a non-comprehensively surveyed area and sustained an unplanned exposure.

- The inspector reviewed the personnel assignments associated with the diving operations in the Unit 2 spent fuel pool. The following was noted.

- Four different lead licensee radiation protection technicians were providing direct radiological oversight of the activity. The individual who was the lead radiation protection technician for the fourth dive had apparently not been involved in diving activities over the part 2.5 years.

- Seven different contractor radiation protection technicians were involved in providing dive coverage for the four dives.

- Four different engineering support personnel were involved in the activity.

The inspectors' discussions with dive personnel indicated dissatisfaction with the use of numerous individuals, inconsistent directions, and a need to re-familiarize personnel with dive activities.

The inspector's interfaces with lead radiation protection personnel in the Unit 2 reactor containment indicated questionable understanding of responsibilities. BGE took actions to clarify responsibilities.

c. Conclusions

BGE implemented an adequate organizational structure for the outage. However, significant weaknesses were noted in the control of work activities, communications, and understanding of responsibilities.

R8 Miscellaneous Issues

R8.1 Plant Tour Observations

During the inspection, the inspector made various tours of the radiological controlled area. The inspector's review indicated generally good housekeeping.

V. Management Meetings

X1 Exit Meeting Summary

During this inspection, periodic meetings were held with station management to discuss inspection observations and findings. On May 7, 1997, an exit meeting was held to summarize the conclusions of the inspection. BGE management in attendance acknowledged the findings presented.

X2 Review of UFSAR Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to

the areas inspected to verify that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

BGE

P. Katz, Plant General Manager
K. Cellers, Superintendent, Nuclear Maintenance
K. Neitmann, Superintendent, Nuclear Operations
P. Chabot, Manager, Nuclear Engineering
T. Pritchett, Director, Nuclear Regulatory Matters
B. Watson, General Supervisor, Radiation Safety
C. Earls, General Supervisor, Chemistry
L. Gibbs, Director, Nuclear Security
T. Sydnor, General Supervisor, Plant Engineering
T. Forgette, Director - Emergency Preparedness
J. Riedel, MOV Project Manager
B. Rudell, General Supervisor, Project Management
L. Smialek, Senior Plant Health Physicist
W. Paulhardt, Radiation Safety Supervisor-Dosimetry
G. Phair, Supervisor Radiation Control
M. Rigsby, Supervisor-Radiation Technical Services
B. Watson, General Supervisor-Radiation Safety
R. Wyvill, ALARA Supervisor

NRC

W. Lazarus, Chief, Mechanical Engineering Branch, DRS

INSPECTION PROCEDURES USED

IP 62707: Maintenance Observation
IP 71707: Plant Operations
IP 93702: Prompt Onsite Response to Events at Operating Power Reactors
IP 61726: Surveillance Observations
IP 37550: Engineering
IP 37551: Onsite Engineering
IP 71750: Plant Support Activities
IP 83750: Occupational Exposure
IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902: Followup - Engineering
IP 82701: Operational Status of the Emergency Preparedness Program

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-317&318/97-02-01	EEI	Failure to implement engineering drawings during troubleshooting.
50-317&318/97-02-02	EEI	Inadequate corrective actions for fuel handling deficiencies.
50-317&318/97-02-03	EEI	Failure to complete Technical Specification 4.9.12.d.2 surveillance on fuel handling ventilation system.
50-317&318/97-02-04	EEI	Failure to comply with Technical Specification 3.9.6.a.2, refueling machine overload limit.
50-317&318/97-02-05	EEI	Failure to follow fuel handling procedures.
50-317&318/97-02-07	IFI	Resolve long-term issues to improve justification of MOV program assumptions for valve factor, load sensitive behavior, and stem friction coefficient for non-tested valves.
50-317&318/97-02-08	VIO	Failure to implement corrective actions as required by 10 CFR 50 Appendix B Criterion 16.
50-317&318/97-02-09	IFI	Resolve long-term design basis capability issues regarding steam generator feedwater isolation valves.
50-317&318/97-02-10	IFI	Resolve long-term design basis capability issues regarding SI pump refueling water tank mini-flow isolation valves.
50-317&318/97-02-11	IFI	Resolve open questions regarding pressure locking and thermal binding to enable issue of safety evaluation report for GL 95-07.
50-317&318/97-02-12	IFI	Revise Procedure MOV-009 to provide details regarding the extrapolation of test results.
50-317&318/97-02-13	UNR	Review air sampling and analysis program
50-317&318/97-02-14	EEI	Failure to implement corrective actions to identified deficiencies in high radiation area access.

50-317&318/97-02-15	EEI	Failure to provide instructions for a diver in a radiologically control area.
50-317&318/97-02-16	EEI	Failure to control access to a very high radiation area.
50-317&318/97-02-17	EEI	Failure to complete radiological surveys prior to personnel entry.

CLOSED

50-317&318/94-17-01	URI	Resolve design basis differential pressure issue regarding mispositioning scenarios.
50-317&318/94-17-02	URI	Resolve pressure locking and thermal binding issues regarding MOVs.
50-317&318/96-03-01	URI	Disposition of Unexplanded Steam Generator Tubes

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
RCA	Root Cause Analysis
UFSAR	Updated Safety Analysis Report
EEI	Escalated Enforcement Item
EDG	Emergency Diesel Generator
IR	Issue Report
ASME	American Society of Mechanical Engineers
PASS	Post-Accident Sample System
QA	Quality Assurance
MOV	Motor Operated Valve
TLD	Thermoluminescent Dosimeter
RCAR	Root Cause Analysis Report
HP	Health Physics
RCA	Radiological Controlled Area
RP	Radiation Protection
RP&C	Radiological Protection and Chemistry
RPM	Radiation Protection Manager
RWP	Radiation Work Permit
URI	Unresolved Item
EPRI	Electric Power Research Institute
PPM	Performance Prediction Model
PORV	Power Operated Relief Valve
NDE	Non Destructive Examination
ISI	Inservice Inspection
ECT	Eddy Current Test