

U.S. NUCLEAR REGULATORY COMMISSION  
REGION I

Report Nos.: 50-317/90-16  
50-318/90-16

License Nos.: DPR-53  
DPR-69

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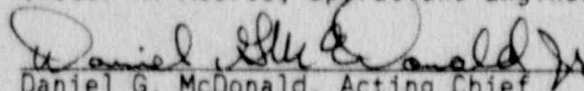
Facility: Calvert Cliffs Nuclear Power Plant, Units 1 and 2

Location: Lusby, Maryland

Inspection conducted: July 1, 1990, to August 11, 1990

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8/22/90  
Date

Inspection Summary:

This inspection report documents routine and reactive inspections during day and backshift hours of station activities including: plant operations; radiological protection; surveillance and maintenance; emergency preparedness; security; engineering and technical support; and safety assessment/quality verification.

Results:

No Violations were identified. One unresolved item was identified regarding the readiness of equipment that would be used for emergency response. (Section 2.2.b.) An Executive Summary follows.

## EXECUTIVE SUMMARY

Calvert Cliffs Resident Inspection Report Nos. 50-317/90-16 and 50-318/90-16

Plant Operations: (Modules 71707, 93702) The licensee was challenged with several events during this inspection period. Licensee response to these events was adequate. An unresolved question remains regarding the readiness of equipment that would be used for response to abnormal conditions (Section 2.2.b).

Radiological Protection: (Module 71707) The interdepartmental communications and coordination between the Radiation Safety section and other site organizations including Operation, Maintenance, and Outage Management have improved from previous inspections. Although the licensee has been relying heavily on the contractor personnel during the current extended outages for both units, contractor personnel in general are familiar with the licensee's station procedures and policies which indicates adequate training and management oversight.

Maintenance and Surveillance: (Modules 61726, 62703) Selected maintenance activities and procedures were observed and reviewed with no deficiencies noted. Various surveillance tests were observed and reviewed with no deficiencies noted. Inconsistencies were identified in the implementation of the locked valve surveillance program. Weakness in the licensee's maintenance program was evidenced by longstanding material deficiencies in the temporary steam generator sluicing system (Section 2.3).

Emergency Preparedness: (Modules 71707, 82301) A region-based inspection and a partial-participation exercise observation of this area were conducted during this inspection period with results contained in Inspection Report 50-317/90-19 and 50-318/90-18.

Security: (Module 71707). Routine review in this area identified no noteworthy findings. A region-based inspection of this area was conducted during this inspection period with results contained in Inspection Report 50-317/90-20 and 50-318/90-19.

Engineering and Technical Support: (Modules 71707, 90712) Routine review in this area identified no noteworthy findings. The inspector concluded that ongoing licensee activities to address defective 4/0 electrical lugs were conservative and safety conscious.

Safety Assessment/Quality Verification: (Modules 71707, 30703) The Offsite Safety Review Committee was observed and determined to be functioning in accordance with Technical Specifications and displayed a good safety perspective. The inspectors observed weakness in the licensee's program to ensure timely assessments of conditions adverse to quality. This was evidenced by the licensee's actions for the inoperative power supply in the Unit 1 shutdown panel (Section 2.2.b), material deficiencies in the temporary steam generator sluicing system (Section 2.3) and the defective 4/0 electrical lug issue (Section 7.2). The inspectors concluded, however that after management assessed the significance of the problems, licensee actions were effective.

## DETAILS

### 1. Summary of Facility Activities

Unit 1 remained in cold shutdown for the duration of the inspection period in a planned maintenance outage. At 1:30 p.m., on July 27, 1990, a bubble was drawn in the pressurizer to establish an operable reactor coolant system loop. This evolution allowed one train of the salt water system to be removed from service for maintenance.

Unit 2 remained defueled for the extended cycle 8 refueling outage with the fuel in the spent fuel pool. Major activities included the steam generator thermal sleeve inspection and repair.

On August 2, 1990, Mr. J. Lieberman, Director, NRC Office of Enforcement, and Mr. C. Cowgill, Acting Branch Chief for NRC Region I Reactor Projects, toured the site with the Senior Resident Inspector. Mr. W. Sellers, of the Department of Justice, accompanied Mr. Lieberman on the tour for general familiarization of a nuclear power plant and its major components.

On August 8, 1990, the licensee conducted their annual emergency exercise. This was a partial-participation exercise as detailed in Section 5 of this report.

### 2. Plant Operations

#### 2.1 Operational Safety Verification

The inspectors observed plant operation and verified that the facility was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted of the following plant areas:

- |                                   |                           |
|-----------------------------------|---------------------------|
| -- control room                   | -- security access point  |
| -- primary auxiliary building     | -- protected area fence   |
| -- radiological control point     | -- intake structure       |
| -- electrical switchgear rooms    | -- diesel generator rooms |
| -- auxiliary feedwater pump rooms | -- turbine building       |

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with technical specification (TS) requirements. Operability of engineered safety features, other safety related systems, and onsite and offsite power sources was verified. The inspectors observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Routine operations surveillance testing

was also observed. Compliance with TS and implementation of appropriate action statements for equipment out of service were inspected. Plant radiation monitoring system indications and plant stack traces were reviewed for unexpected changes. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets, system safety tags, and the jumper and lifted lead book. Plant housekeeping controls were monitored, including control and storage of flammable material and other potential safety hazards. The inspectors also examined the condition of various fire protection, meteorological, and seismic monitoring systems. Control room and shift manning were compared to regulatory requirements and portions of shift turnovers were observed. The inspectors found that control room access was properly controlled and that a professional atmosphere was maintained.

In addition to normal utility working hours, the review of plant operations was routinely conducted during portions of backshifts (evening shifts) and deep backshifts (weekend and midnight shifts). Extended coverage was provided for 9 hours during backshifts and 7.5 hours during deep backshifts. Operators were alert and displayed no signs of inattention to duty or fatigue.

a. Establishment of a Pressurizer Bubble and Reactor Coolant System Venting Evaluation

On July 27, 1990, the licensee established a pressurizer bubble in Unit 1. The inspectors observed portions of this evolution and independently verified the operation of the pressurizer level indications. Procedures were observed to be in use and operations were performed in a controlled manner. Portions of the reactor coolant system venting evolution were observed. This process involved alternately starting and stopping the reactor coolant pumps to sweep air from the steam generator tubes and venting trapped air from the reactor vessel head. The inspector verified that low temperature-overpressure protection (LTOP) controls for reactor coolant pump (RCP) operations were followed. The operators noted higher than expected bleedoff flow from 12A RCP and higher than expected vibrations from several of the pumps. The system engineer was in the control room at the time and is reviewing the observations of the operators.

b. Radiation Monitors

The inspectors reviewed the status of the radiation monitors during the inspection period and noted a large number of monitors out of service. For example, on July 31, the control room operators log listed twenty-one monitors that were out of service for various reasons. Although the licensee was taking adequate compensatory actions in each case, the inspectors expressed a concern regarding the prospect of a unit restart with the radiation monitors in their current condition. The Operations General Supervisor commented that he had a similar concern and was relying on an ongoing control room deficiency project to address and/or correct the problems as necessary for restart. The inspectors reviewed the most recent control room deficiency report, dated July 31, 1990, and verified that the monitors were identified for corrective action. The resident staff will continue to monitor this situation during their inspection effort.

2.2 Followup of Events Occurring During Inspection Period

During the inspection period, the inspectors provided onsite coverage and followup of unplanned events. Plant parameters, performance of safety systems, and licensee actions were reviewed. The inspectors confirmed that the required notifications were made to the NRC. During followup of the events, the inspector reviewed the corresponding CCI-118N (Calvert Cliffs Instruction, "Nuclear Operations Section Initiated Reporting Requirements") documentation, including details of the events, root cause analyses, and corrective actions taken to prevent recurrence. The following events were reviewed.

a. Emergency Diesel Generator Cable Separation

On July 10, 1990, when reviewing the interactive cable analysis (ICA), the licensee found that a severe fire in an electrical equipment room on the 69 foot elevation of the Unit 1 auxiliary building (Room 529) had the potential to prevent emergency diesel generators (EDG) 11 and 12 from supplying required loads in an accident. The detailed review of the ICA is part of the licensee's ongoing process to upgrade its fire protection program.

Room 529 contains cables and motor control centers for EDG 11 and the current transformer cables for EDG 12. These cables provide the signals for EDG 12 to Bus 14 differential current protection.

The licensee initiated a CCI-118N report and determined that a licensee event report will be issued to document this event as a condition outside the design basis of the plant. Other licensee actions included the initiation of a problem report to document the deficiency and a Field Engineering Change (FEC) request to reroute the current transformer cables for EDG 12 cable.

The licensee's initial assessment of this event determined that the cable arrangement did not satisfy 10 CFR 50, Appendix R, criteria for achieving and maintaining hot shutdown conditions in a fire scenario in conjunction with a loss of offsite power. This initial assessment also determined that the safety significance of this event for the current cold shutdown mode was small. This assessment is based on the fact that Appendix R assumptions for cold shutdown conditions are less limiting. Also, review of the cable design itself and the expected severity of a fire indicate that the cable would probably remain functional during and after a postulated fire. Additionally, there are smoke detectors in the room which would provide early warning to control room operators of fires in the area.

The inspector walked down the area and cable in question with the licensee. The room contained two storage areas in which approximately 300 pounds of plastic and 200 pounds of wood were stored. The inspector requested the licensee to evaluate the potential severity of a fire based on this material. The licensee determined that the room had a low combustibility loading with a severity equivalent to a two (2) minute fire. The inspector also verified that the smoke detectors located in room 529 were operable.

The inspector noted during the cable walkdown that the cable entered the west electrical penetration room, located below room 529, before entering the EDG 12 room and requested information on why a fire in this room was not also a concern. The licensee demonstrated via a review of cable locations and electrical system lineups that the only room of concern was room 529. Additionally, the licensee reviewed Unit 2 cable arrangements and found no similar problems.

The licensee has initiated efforts to reroute the current transformer cables for EDG 12. A restriction to limit entry into Mode 4 has been administratively imposed until repairs are complete.

The inspector concluded that the licensee's identification of this problem demonstrates a strong effort to ensure Appendix R criteria are met. The licensee's review of the safety significance was acceptable. Licensee actions to address, document, and report this event were acceptable.

b. Inoperative Power Supply in Unit 1 Remote Shutdown Panel

On July 18, 1990, the licensee initiated a CCI-118N report to evaluate reportability of a discovery that a power supply in the Unit 1 remote shutdown panel was inoperative during plant start-up and operation in April 1990. The power supply provides power for the remote shutdown controllers to the turbine driven auxiliary feedwater (AFW) pumps for speed and flow control to the steam generators as well as the controller for No. 11 atmospheric dump valve.

The CCI-118N report was generated after review of a problem report written on July 3, 1990. The problem report noted that the power supply was declared inoperative on May 24, 1990, yet there was a previous maintenance request (MR) dated October 20, 1989, that found that the power "on" indicating lights were not lit. The problem report originator was concerned that the power supply had been inoperative since that time.

The licensee determined that this event was not reportable as a Licensee Event Report since these controls are not required by the technical specifications and there are alternate local manual control stations for AFW and the atmospheric dump valves. These local manual control stations would require operators in addition to those needed for safe shutdown. The licensee also assessed that there were enough qualified personnel to man these additional stations and perform the safe shutdown functions if it had been required. Thus Appendix R criteria were satisfied.

The inspector noted that over six months elapsed from the time the power supply lights were found out until it was determined that the power supply itself was inoperative. Also, the power supply failure was found during a routine preventative maintenance procedure. Once the failure was found, an additional month elapsed before a problem report was written to address the concern of operating in April of this year without this equipment. The inspector determined that these circumstances represent a weakness in the licensee's problem identification process to ensure that issues which could affect operations are addressed and corrected in a timely manner.

In the process of determining reportability, the licensee identified concerns with the delays in the problem identification system similar to those noted by the inspector, as well as other concerns with facility work processes associated with this event. To further review the issues, the licensee initiated a Calvert Cliffs Event Review (CCER) group on July 25, 1990. This multidisciplinary group was chartered with the task of determining root causes for the unavailability of this safe shutdown equipment and to recommend corrective actions.

The inspector observed portions of this group's meetings. They demonstrated a good safety perspective and introspectively examined the issues. The preliminary root causes were inadequate recognition of the need for this equipment to be available and inadequate prioritization of the original maintenance request. Recommendations made by the group will be forwarded to the POSRC and site managers.

The inspector observed that this equipment was designated to satisfy Appendix R requirements for safe shutdown in a worst case fire scenario. The equipment, however, is not identified in the technical specifications, thus not tested per the surveillance program requirements to assure operability. The inspector expressed a concern regarding the possible inadequacy of the licensee's programs to assure that this and similar equipment are in a state of readiness when plant conditions warrant use of this equipment. The CCER group also identified a similar concern and the licensee is reviewing one issue. This item is unresolved pending the licensee's review of this concern and a review of corrective actions. (50-317/90-16-01 and 50-318/90-16-01).

The inspector concluded that licensee actions prior to the problem report initiation represent weakness in timely identification and resolution of issues. Actions subsequent to the problem report initiation were adequate to determine the root causes of this event.

c. Leak in the Saltwater System

On July 19, 1990, at 2024 hours, the licensee made a four hour notification to the NRC to report a small leak in the No. 11 saltwater (SW) system. The weepage was located in the toe area of the outer weld of a slip-on flange in the piping downstream of the No. 11 service water heat exchanger saltwater outlet valve. The weepage was discovered by workers performing a weld modification to the outer weld area as a result of previously identified SW problems.



The licensee entered technical specification (TS) action 3.4.10.1.C for ASME Code Class 3 components not conforming to the structural integrity requirements of TS 3.4.10.1. Action to perform repairs was immediately started and a dedicated watch was stationed to continuously monitor the weepage. The No. 11 SW system was judged to be operable by the licensee based on an assessment that the system is capable of performing its intended function, a very low leak rate, and a preliminary determination that the piping can withstand design seismic, deadweight, pressure, and thermal loads.

The licensee initiated detailed analysis of the thinned area to verify its initial determination that design loads could be withstood and to support plans for repairs.

The licensee performed additional ultrasonic test (UT) evaluations of the leak area with a smaller probe that allowed examination closer to the affected area. The UT data found a defective area under the toe of the weld about 2 inches long extending in the circumferential direction and about 0.5 inches wide longitudinally. The minimum wall thickness measured was 0.11 inches around the area where the weep was observed. The design nominal thickness is 0.375 inches. The licensee determined that the cause of the wall thinning was corrosion and the weepage was the result of melt-through of the thinned wall during welding. This determination was based on analysis of the UT data. Further examinations were performed on two additional flanges that were scheduled for this modification to identify other potential wall thinning conditions. All areas were above the nominal thickness.

The completed analysis of the affected area verified the licensee's initial determination that design loads could be withstood. Based on the analysis, the licensee exited T/S action statement 3.4.10.1.c. The affected piping section was subsequently replaced during other work on the No. 11 SW system. The inspector observed no adverse conditions.

d. Inadvertent Opening of a Power Operated Relief Valve (PORV)

On July 31, 1990, at 3:25 p.m., a Unit 1 power operated relief valve (PORV) ERV-404 inadvertently lifted. After verification of an actual lift and determination that it was inadvertent, the control room operator shut the PORV by selecting to "override shut" on the PORV control and closing its associated block

valve. Reactor coolant system pressure was verified by the operators to be about 205 psia at the time the PORV lifted. This pressure is well below the lift setpoint for LTOP protection. Pressure had decreased to about 194 psia when the PORV was closed. The control room operators then verified that the unit was in a stable condition. The licensee entered TS 3.4.9.3 action which requires restoration of the PORV within five days or depressurization and venting the RCS within the next 48 hours.

The licensee established a project team to identify and correct the cause of the failure. The project team identified the root cause to be a degraded connection on a resistor in the valve actuation logic for LTOP protection. In this circuit, the output of the pressurizer pressure instrument produces a current output with a higher current corresponding to a higher pressure. The current is passed through the resistor which had the degraded connection and the proportional voltage drop across the resistor is the pressure signal to the LTOP logic. A higher current produces a larger voltage drop thus, a larger pressure signal to the LTOP logic. In this case, the connection of the resistor to the loop was made up via a crimped lug and, over time, the electrical contact degraded. Work was in progress in the vicinity of this resistor at the time of the event and it was postulated that vibrations from this work could have increased the resistance. With a constant current, this increase in resistance caused a corresponding increase in the voltage drop. The increased voltage drop was sensed by the LTOP logic as a high pressure signal and opened the PORV. The effect of vibrations on the resistance was verified via troubleshooting. As a result, the licensee replaced the crimped connection with a soldered connection. The circuit was tested successfully and ERV-404 was declared operable on August 2, 1990, at 7:40 a.m.

The licensee identified that this was a generic problem with the other Unit 1 PORV, both Unit 2 PORVs, and possibly other control circuits. Additionally, the method for calibration and testing the circuits may not test for changes in resistance and result in non conservative settings. A CCI-118N report was generated on August 1, 1990, to determine reportability. The licensee also replaced the resistor for the other Unit 1 PORV and generated maintenance requests for repairs to the Unit 2 PORV control circuits. The licensee also generated a problem report to identify other possible circuits that have crimped connections. The licensee plans to evaluate these additional crimped connections and repair them as appropriate.

The inspector concluded that licensee response to this event was appropriate.

e. Inadvertent ESFAS Actuation

On August 2, 1990, at 9:24 a.m. an Engineered Safety Features Actuation System (ESFAS) actuation occurred on Unit 1. The licensee was in the process of re-energizing the "A" logic cabinet following maintenance work when the actuation occurred. The control room operators verified plant response and verified that the actuation was inadvertent. The licensee determined that no injection had occurred, LTOP conditions were not challenged, and a four-hour emergency notification of the event was made.

The licensee determined that a fault developed in the 120 VAC input to the "A" logic cabinet. The fault caused the 1Y01 vital AC bus fuse to fail which initiated the ESFAS. The following equipment actuated: 1-CVC-509 (Boric Acid Storage Tank Gravity Drain Valve) opened, 12 Emergency Diesel Generator started, and the penetration room vent fan started. The licensee confirmed correct system response to the actuation after reviewing plant computer printouts and operator information.

At 9:30 p.m., August 7, 1990, Unit 1 experienced spurious ESFAS actuation signal which caused the #14 Containment Air Cooler to shift from fast to slow speed. The control room operators again verified plant response and determined that the actuation was inadvertent. The licensee found that the 15V power supply for the "B" logic cabinet had dropped to 12V. Although logic trip modules were dimly lit for containment spray actuation (CSAS), recirculation actuation (RAS), containment radiation (CRS) and steam generator isolation (SGIS), actuation signals were not generated because voltage did not drop below 12V.

The licensee deenergized the "B" logic cabinet to replace the suspect 15V power supply. This action deenergized the auto load sequencers for both 11 and 12 Emergency Diesel Generators (EDGs) since logic cabinet "A" was already deenergized for repair. With both Unit 1 EDGs out of service, the action statement for technical specification 3.8.2.2 became applicable and the licensee established containment integrity. After replacing and properly adjusting the 15V power supply in the "B" logic cabinet, the licensee reenergized the cabinet and exited the associated action statement.

The inspector concluded that licensee response to both events was appropriate and will continue to follow licensee actions to trouble shoot, locate and correct the cause of the fault in the "A" logic cabinet.

### 2.3 Steam Generator Sluice

On July 24, 1990, the inspector observed performance of the operating instruction (OI-12B) for sluicing water from 22 to 21 steam generator (S/G) inside the unit 2 containment. The evolution to transfer water between the S/Gs was performed to facilitate maintenance on the 22 S/G thermal sleeves. A temporary sluice system was employed for the transfer since the permanent system was out of service for maintenance. The procedure was coordinated from the control room, but valve manipulations to establish and secure flow were made locally at the sluice rig. During the evolution, the inspector noted deficiencies in work practices, radiological control practices, material condition, and industrial safety.

The inspector found that the temporary sluice rig was not being monitored by operators stationed in the containment building. Concurrently, the inspector observed the rig leaking water onto the Unit 2 containment floor with approximately 6-10 gallons pooled near the rig. The inspector informed a Health Physics (HP) Technician of these problems. The technician indicated that he was aware of the leak from the sluice rig, did not know the location of the responsible operators, and did not assess the leak as an immediate concern. After further inquiry by the inspector, the technician located the operators who took actions to secure the rig and associated leak. The inspector concluded that operator monitoring of the sluicing evolution was inadequate and contrary to the general standards of performance described in licensee procedures (CCI-300K). Also, the HP technician did not follow the licensee's stated practice to limit the potential spread of contamination by aggressively acting to address the leak.

The inspector also noticed a 3 gallon bucket suspended on the sluice rig beneath an open threaded fitting. The operators informed the inspector that the bucket was installed to collect water leaking past two upstream isolation valves. When it was filled, the inspector observed one operator removing the bucket while another operator covered the leaking fitting with a gloved palm to prevent further leakage. The operators indicated that this was an unusual practice, but also considered it appropriate since a sample of the steam generator water prior to the sluicing evolution was found to have acceptable activity levels. The inspector determined after discussions with HP and operations supervisors that the licensee considered this an unacceptable and not a normal practice. The inspector concluded that this indicated weakness in operator and HP technician understanding of appropriate measures to address radiological control aspects of water leakage into containment. The inspector also determined that the leaking isolation valves were a longstanding material deficiency that demonstrated weakness in the licensee's program to identify, document, and correct conditions adverse to quality.

The inspector also noticed a mobile 600/120 volt power supply that was wet by and located in the pooled water from the leaking sluice rig. The inspector informed the operators and technician of this concern but did not witness prompt and conservative actions. The inspector considered this a weakness in operator awareness and responsiveness to a potential electrical safety hazard.

When informed of these weaknesses, the licensee initiated a thorough and timely investigation into the sluicing operation. The inspector found the licensee's assessment of the problems to be consistent with those noted above, and considered their proposed corrective actions to be acceptable. The inspector will follow implementation of the proposed actions to prevent recurrence.

### 3. Radiological Controls

During this inspection period, the inspector observed the implementation of selected portions of the licensee's radiological controls program. The licensee's Radiation Safety (RS) organization and implementation of management controls was inspected by interviews with various RS section personnel and review of the following documents:

- CCI 800, "Calvert Cliffs Radiation Safety Manual"
- Selected Radiological Control Reports and Problem Reports
- QA audit reports 90-02 and QA surveillance report S-90-14

With both units in extended outages, the licensee has been relying heavily on contractor personnel for radiation safety technician positions. Based on the inspector's observations and interviews with RS section personnel, the inspector determined that contractor personnel are familiar with the licensee's station procedures and policies which indicates adequate training and management oversight. An exception was noted during the inspector's observation of the S/G sluicing activity as described in Section 2.3 of this report.

The inspector reviewed QA audit 90-02 and QA surveillance S-90-14 which covered ALARA and radiological control operations. The inspector noted that the scope and findings of these audits indicate that adequate reviews of these areas were being performed. The RS section response to the findings were timely, appropriately addressed root causes, and corrective actions were taken.

The interdepartmental communications and coordinations between the RS section and other site organizations including Operation, Maintenance, and Outage Planning appeared to have improved from previous inspections. Since the early part of June, 1990, an RS section representative attends Operations pre-shift briefings. An RS section representative has been temporarily assigned as an Outage Planning and Scheduling coordinator to improve the efficiency of RS section support. The inspector noted that the number of either cancelled or postponed special work permit packages has been reduced considerably in recent months.

The RS section uses the Radiological Control Report (RCR) system as a means to initiate and track performance deficiencies in the radiological controls area. RCRs are issued to the responsible work group supervisors, who details corrective actions taken and returns the RCR to the RS section for review and closeout.

The RS section also documents problems and deficiencies via the Problem Report (PR) system per CCI-116, "Identification and Control of Nonconforming Conditions." The PR system was implemented on April 27, 1990, as an initial step in consolidating the licensee's various tracking systems for problems, maintenance, and other issues into a single system. Both the RCR system and the PR system have action level flags that would require escalation to the QA nonconformance report system. The inspector reviewed recent RCRs and PRs and noted no unacceptable conditions.

#### 4. Maintenance and Surveillance

##### 4.1 Maintenance Observation

The inspectors observed maintenance activities, interviewed personnel, and reviewed records to verify that work was conducted in accordance with approved procedures, technical specifications, and applicable industry codes and standards. The inspectors also verified that: redundant components were operable, administrative controls were followed, tagouts were adequate, personnel were qualified, correct replacement parts were used, radiological controls were proper, fire protection was adequate, quality control hold points were adequate and observed, adequate post-maintenance testing was performed, and independent verification requirements were implemented. The inspectors independently verified that selected equipment was properly returned to service.

Outstanding work requests were reviewed to ensure that the licensee assigned appropriate priority to safety-related maintenance. The inspectors observed/reviewed portions of the following maintenance activities.

##### a. Grouting of 4kv Switchgear

On July 30, 1990, the inspector witnessed the application of an epoxy grout underneath the switchgear tracks in several Unit 1 4kv cubicles. This work was being conducted in accordance with field change request 88-166 to correct sagging and misalignment of the switchgear. The craft personnel at the jobsite demonstrated an acceptable awareness of their task. Adequate precautions were observed regarding personnel safety while working inside the switchgear cubicles. No discrepancies were identified during this observation.

b. Resistance Temperature Detector (RTD) Temperature Recorder Repair

On July 10, 1990, the inspector witnessed the performance of maintenance on temperature recorders for various plant RTDs. This work was performed in accordance with maintenance order No. 200-164-284A. No discrepancies were identified during the observation.

c. Motor Operated Valve (MOV) Maintenance

The inspector reviewed selected portions of maintenance performed during this period on safety related MOVs. The inspector reviewed procedures, observed maintenance and post-maintenance testing, and assessed the status of previous MOV inspection open items. The inspector determined that MOV maintenance during the current outage included scheduled overhauls, corrective actions for surveillance deficiencies, and limit switch assembly overhaul and modifications.

On July 20, 1990, the inspector observed cleaning and regreasing of the limit switch assembly for Shutdown Cooling Return Isolation Valve 2-SI-652 (MO #209126927A). The inspector also observed static signature analysis (VOTES tests) on 2-SI-652 and 2-SI-617 (MO #200-106-208A). VOTES tests were performed to assess the as-found condition of the MOVs and establish baseline data for subsequent MOV performance tests. The inspector noted that licensee quality assurance personnel were present during the tests.

The inspector concluded that personnel performing MOV maintenance and testing were knowledgeable, maintenance was conducted in accordance with approved procedures, and work areas were clean and uncluttered. No noteworthy discrepancies were found during the inspector's review of MOV maintenance procedures or observations of MOV overhaul and testing.

The inspector also reviewed licensee maintenance activities to address five findings discussed in NRC Inspection Report 50-317/89-12 and 50-318/89-12. The status of these items are provided in sections 9.5 through 9.9 of this report.

#### 4.2 Surveillance Observation

The inspectors witnessed selected surveillance tests to determine whether properly approved procedures were in use, technical specification frequency and action statement requirements were satisfied, necessary equipment tagging was performed, test instrumentation was in calibration and properly used, testing was performed by qualified personnel, and test results satisfied acceptance criteria or were properly dispositioned. Portions of the following activities were reviewed.

- a. STP M-190-0, "Diesel Fire Pump Battery Weekly Check" and STP M-390-0, "Diesel Fire Pump Battery Quarterly Check"

The inspector observed performance of these tests. Tests were properly conducted and the technicians exhibited a safety conscious attitude. No adverse conditions were noted.

- b. STP M-672B-1, "Pressurizer Relief Valve (ERV) Channel Functional Test"

The inspector observed the pretest brief, performance of the test, and a post test debrief. Data from the completed test were also reviewed. The pretest brief addressed appropriate precautions and methods for conducting the test. The test was conducted in a professional and controlled manner. The technicians were knowledgeable of the procedure and the associated equipment. Some minor procedure enhancements were noted by the technicians and these were discussed at a debrief of the test. A procedure change was recommended by the technicians to incorporate these enhancements. The inspector observed no adverse conditions.

- c. STP O-8A-1, "11 Diesel Generator and 4KV Bus 11 LOCI Sequencer Test"

The inspector observed the pretest review, test performance, and reviewed the post test data. The test was conducted in a professional manner. The shift supervisor closely observed performance of the test to assure proper conduct. One problem noted during the test performance was a procedure error in the STP which did not reference the proper steps in the operating instruction to prelube the diesel generator. The shift crew initiated a procedure change and proceeded with the test. No adverse conditions were noted.



d. STP 0-93-1 "Locked Valve Verification Outside Containment"

The inspector observed selected portions of this verification. Discrepancies noted during performance were promptly reported to the control room. The inspector noted that only certain valves in the diesel generator rooms were checked while others were not. The most notable valves that were not checked were in the air start lineup. Valves for jacket water cooling and lube oil cooling were also omitted. The inspector asked the shift supervisor if these valves were part of another surveillance. The shift supervisor was unable to identify any other surveillance that verified these valves in their locked positions. The inspector discussed this observation with the surveillance coordinator. The inspector determined from this conversation that although there is not a technical specification requirement for checking these valves, they are checked as part of a locked valve verification program started in the mid 1970's. Based on this discussion, the surveillance coordinator stated that a review to determine which locked valves should be verified would be conducted. The inspector had no further questions.

5. Emergency Preparedness

A team of five NRC Region I and Headquarters personnel observed the licensee's partial-participation annual emergency preparedness exercise conducted on August 8, 1990, and performed routine inspection activities under Module 82301. The results of the inspection will be documented in NRC inspection report 50-317/90-19 and 50-318/90-18.

The resident inspectors observed portions of the emergency preparedness drill conducted by the licensee on July 30, 1990. This drill was in preparation for the licensee's partial-participation annual exercise on August 8, 1990. The inspectors observed activity in both the technical support center and the control room simulator. No noteworthy items were identified.

6. Security

A region-based inspection of this area was conducted during this inspection period with results contained in Inspection Report 50-317/90-20 and 50-318/90-19.

The resident inspectors reviewed the compensatory security actions established following the removal of a salt water pump from the intake structure. Adequate measures had been implemented and no unacceptable conditions were identified.

During routine inspection tours, the inspectors observed implementation of portions of the security plan. Areas observed included access point search equipment operation, condition of physical barriers, site access control, security force staffing, and response to system alarms and degraded conditions. These areas of program implementation were determined to be adequate. No unacceptable conditions were identified.

## 7. Engineering and Technical Support

The inspector reviewed selected design changes and modifications made to the facility which the licensee determined were not unreviewed safety questions and did not require prior NRC approval in accordance with 10 CFR 50.59. Particular attention was given to safety evaluations, Plant Operations Review Committee approval, procedural controls, post-modification testing, procedure changes resulting from this modification, operator training, and UFSAR and drawing revisions. The following activities were reviewed:

### 7.1 Generic Implications and Resolution of CEA Failure at Maine Yankee for Convert Cliffs

On June 7, 1990, while conducting cold functional testing of the control element assemblies (CEAs) following the cycle 12 refueling outage at Maine Yankee, one CEA could not be fully inserted in the core. Subsequent inspection of the CEA revealed that the end cap was missing from the center CEA finger, the lower stainless steel spacer and boron carbide pellets had fallen out of the center finger and an axial crack existed at the lower end. The upper stainless steel spacer was cocked in the bottom of the CEA finger and was causing the center CEA finger to bind against the guide tube. The CEA could not be fully inserted in the core during cold functional testing due to interference from boron carbide pellets that had fallen into the center CEA guide tube.

On June 25, 1990, the NRC staff met with the Combustion Engineering Regulatory Response Group (CERRG) to discuss the generic implications and proposed resolution of the CEA failure that occurred at Maine Yankee. The CERRG concluded that the CEA failure was most likely due to irradiation assisted stress corrosion cracking (IASCC). The CERRG also concluded that the failure mechanism was only applicable to the older Combustion Engineering (CE) CEA design which did not have an absorber plug (other than boron carbide pellets at the bottom of the center CEA finger). In order to resolve this issue, the CERRG proposed specific Action Programs for each affected licensee. The CERRG submitted its report to the NRC addressing this issue by letter dated June 26, 1990. The licensee stated that Unit 1 has no old design CEAs installed and there are no plans to use them in the future. Thus, the problem is not applicable to Unit 1.

In response to the NRC letter dated July 6, 1990, the licensee stated that the current plans for Unit 2 are to eventually replace all 68 CEAs of the older combustion engineering design. If the vendor delivery schedule for the new CEAs does not support the Unit 2 restart, which is currently scheduled for late 1990, the licensee plans to implement the CERRG proposed specific action programs. BG&E also plans to respond to NRC's request for additional information regarding the action program at least 60 days prior to the scheduled Unit 2 restart. The inspector will review the status of this issue prior to Unit 2 restart.

## 7.2 Defective 4/0 Electrical Lugs

On May 24, 1990, the licensee initiated an internal problem report to document cracking found in Thomas and Betts (T&B) size 4/0 electrical terminal lugs. The cracks were found around the barrel of the lug after crimping. The licensee also initiated an internal non-conformance report (NCR) to isolate the T&B size 4/0 lugs. On July 26, 1990, the licensee's NCR screening group identified the cracked lugs as a potential restart issue and confirmed cracked 4/0 and 2/0 lugs on installed plant equipment. The licensee made a 10 CFR Part 21 notification to NRC Region I regarding the defective lugs.

The licensee project team formed to address this issue determined that the suspect lugs were commercial grade purchases from T&B through a purchasing agent, Graybar Co. T&B machined the lugs from copper casts received from an unspecified foundry. Neither T&B nor the foundry established lot control or material history on the casts and lugs.

Initially, the team conservatively bound the investigation by incorporating all sizes of T&B cast copper lugs installed since the licensee began procurement in April 1985. The team also recommended that the licensee quarantine the lugs in storage for laboratory testing and review field applications to determine the extent of defective parts installed in the plant. After receiving initial laboratory test results and material purchase order information, the team reduced the scope of the issue to 4/0 and 2/0 lugs. The team determined that several orders of these lugs had excessively large grain structure, which characteristically result from rapid cooling during production. The large grain structure reduced metal toughness, making the cast lugs more brittle and susceptible to cracking when subject to mechanical stresses caused by crimping.

The team continued to evaluate the effect of defective lugs on equipment operability although preliminary assessments were that the observed cracks did not significantly degrade the mechanical or electrical capability of the lugs. To prevent the receipt and installation of lugs with similar material deficiencies, the team initiated a review to validate terminal lug dedication criteria. The team also proposed a revision to field inspection criteria for crimps to include an enhanced visual inspection for cracks.

The inspector noted that over two months elapsed between problem report initiation (May 24, 1990) and when cracked terminal lugs were identified as a potential restart problem (July 26, 1990). The inspector determined that this delay manifests a weakness in the licensee's program to ensure timely corrective actions for conditions adverse to quality. The inspector concluded however, that ongoing licensee activities to address this issue were conservative and safety conscious. The inspector will review the status of this issue prior to Unit 1 restart.

## 8. Safety Assessment and Quality Verification

### 8.1 Plant Operations and Safety Review Committee

The inspector attended several Plant Operations and Safety Review Committee (POSRC) meetings. TS 6.5 requirements for required member attendance were verified. The meeting agendas included procedural changes, proposed changes to the TS, Facility Change Requests, and minutes from previous meetings. Items for which adequate review time was not available were postponed to allow committee members time for further review and comment. Overall, the level of review and member participation was adequate in fulfilling the POSRC responsibilities. No unacceptable conditions were identified.

### 8.2 Review of Written Reports

Periodic and Special Reports, Licensee Event Reports (LERs), and Safeguards Event Reports (SERs) were reviewed for clarity, validity, accuracy of the root cause evaluation and safety significance description, and adequacy of corrective action. The inspector also verified that the reporting requirements of 10 CFR 50.73, 10 CFR 73.71, Station Administrative and Operating, and Security Procedures, and Technical Specification 6.9 had been met. The following reports were reviewed:

- LER 90-18      Axial Shape Index Not Monitored as Required by  
Technical Specifications
- LER 90-19      Inadequate Breaker Coordination
- LER 90-20      Inoperable Fire Door Affects Halon System Operation

No unacceptable conditions were identified.

### 8.3 Offsite Safety Review Committee

The inspectors observed selected activities of the offsite safety review committee (OSSRC) during the inspection period. This effort included attendance at two OSSRC meetings held during the period. In addition, the inspector reviewed the OSSRC Manual and a representative sample of previous meeting minutes.

The inspector determined that the OSSRC was functioning in accordance with the technical specifications. The offsite members displayed a good knowledge of events and problems at the facility and clearly recognized the broad implications of generic issues. The committee consisted of a good mixture of in-house as well as outside expertise. The OSSRC recommendations to the Vice-President Nuclear were reviewed and implemented as necessary. No concerns were identified.

### 9.0 Licensee Action on NUREG-0660, NRC Action Plan Developed as a Result of the TMI-2 Accident

On October 31, 1980, the NRC staff issued NUREG-0737 which provided guidance for implementing TMI action plan items. On December 17, 1982, Generic Letter No. 82-33 transmitted Supplement 1 to NUREG-0737 which broke down the action items into numbered descriptions. Licensee letters containing commitments to the NRC were used as a basis for acceptability, along with NRC clarification letters. The following items were reviewed:

#### 9.1 Item III.D.3.4.3, Control Room Habitability Requirement

The inspector reviewed the appropriate sections in the plant technical specifications (TS) and the Updated Final Safety Analysis Report (UFSAR) and compared them to the licensee's safety analysis and the NRC staff's safety evaluation. Applicable surveillance test procedures were reviewed to determine adherence to the TS requirements and operability of the control room HVAC system. The inspector also walked down the system with the cognizant licensee engineer.

Based on this review, the inspector concluded that the control room HVAC system is designed, operated, maintained, and tested in accordance with the UFSAR, TS, and the licensee's commitments.

Previous inspections (combined Inspection Reports 50-317/85-27 and 50-318/85-25, and 50-317/85-31 and 50-318/85-26) in this area had identified three unresolved items: 317/85-27-01 and 318/85-25-01; 317/85-31-01 and 318/85-26-01; and 317/85-31-02 and 318/85-26-02. The inspector reviewed these items and determined that the licensee has satisfactorily resolved the items. The details of the licensee's actions in this area are described in section 10.11 of this report. The inspector had no further questions. This item is closed.

9.2 Item II.F.2.4, Instrumentation for Detection of Inadequate Core Cooling

The subcooled margin monitor(SMM), the heated junction thermocouple (HJTC)/ the reactor vessel level monitoring system (RVLMS), and the core exit thermocouples(CET) comprise the inadequate core cooling instrumentation required by the Item II.F.2. The function of the inadequate core cooling instrumentation is to enhance the ability of the plant operator to diagnose the approach to, and recovery from the inadequate core cooling condition. Additionally, they aid in tracking reactor coolant inventory.

Previous inspections in this area are documented in inspection reports 50-317/80-08 & 50-318/80-08, 50-317/80-16 & 50-318/80-16, and 50-317/85-24 & 50-318/85-21. This item had remained open pending full implementation of RVLMS and the licensee submittal of the technical specification(TS) amendment requests for CET and RVLMS.

Full implementation of the RVLMS was completed in June, 1988 for Unit 1 and June 1987 for Unit 2, respectively. The inspector verified that the RVLMS for both units are operable and calibrated. The inspector also verified that the appropriate operational procedure, OI-1L, Reactor Vessel Level Monitoring System, is being used and the training for the operators on the RVLMS has been completed.

The inspector noted that the licensee submitted the TS amendment requests for CET and RVLMS on July 10, 1987, and June 16, 1988, respectively. The licensee submitted a revision to the TS amendment on August 3, 1990, to require a more restrictive action statement. The NRR licensing project manager informed the inspector that the NRR review of the licensee's TS amendment requests is nearly complete. The inspector noted that in the interim, the licensee is administratively implementing the proposed TS requirements. Based on the above review, this item is closed.

## 10. Followup of Previous Inspection Findings

Licensee actions taken in response to open items and findings from previous inspections were reviewed. The inspectors determined if corrective actions were appropriate and thorough and previous concerns were resolved. Items were closed where the inspector determined that corrective actions would prevent recurrence. Those items for which additional licensee action was warranted remained open. The following items were reviewed.

### 10.1 (Closed) Violation 50-317/88-07-02 and 51-318/88-08-02

This violation involved the failure to include all failures and challenges to the pressurizer power operated relief valves or safety relief valves in the licensee annual report to the NRC. This is a requirement of Technical Specification 6.9.1.5.c. The inspector reviewed the licensee response to this violation, dated July 14, 1988, as well as the administrative procedure (CCI-155) that was revised to ensure this requirement is met. In addition, the inspector reviewed the subject reports to the NRC from the previous two years that reported the results of the valve performance. The licensee corrective actions regarding this violation were determined to be adequate. This item is closed.

### 10.2 Closed (50-317/89-06-01 and 50-318/89-06-01)

During a previous inspection, a violation of the requirements of both 10 CFR 50.59(a)(1) and technical specification 6.5.1.6 had been identified regarding failure to perform a written safety evaluation. Specifically, a change of intent to Operating Instruction 29, "Salt Water System," had been made to allow throttling of the salt water system discharge valves without a written safety evaluation required by 10 CFR 50.59 or without a review and approval by the Plant Operations Safety Review committee (POSRC).

The licensee's corrective actions included a revision to the Calvert Cliffs Instruction (CCI)-101, "Calvert Cliffs Implementing Procedure Development and Control," which incorporated a detailed set of screening criteria for "change of intent" determination. Changes to all procedures must be approved by two members of the plant management staff prior to review by POSRC. One of these persons must be a licensee supervisor from the affected discipline. The procedure change must then be reviewed by POSRC and approved by the plant manager.

Licensee implementation of the revised CCI-101 was reviewed extensively during a Readiness Assessment Team Inspection conducted in November 1989. The team had determined that licensee's corrective actions in this area were adequate. Although the checklist containing a set of screening criteria was somewhat difficult to interpret and cumbersome to use, licensee personnel were making conservative determinations as to which procedure changes constituted changes of intent. During this inspection period, the inspector reviewed several procedure changes and found no discrepancies. This item is closed.

10.3 (Closed) Violation (50-317/89-04-02 and 50-318/89-04-03)

This violation concerned a failure of POSRC to meet its responsibility of reviewing facility operations to detect potential safety hazards. The inspector reviewed the licensee's response to the violation and associated documentation to support that response. The inspector also interviewed selected members of the POSRC to assess their safety perspective, their sensitivity to safety concerns, and their methods to identify safety concerns. The inspector determined from these interviews that the POSRC members have a good safety perspective and are sensitive to safety concerns. They were also knowledgeable of methods to identify concerns. Additionally, the members expressed confidence that a conservative attitude exists in the POSRC. The members stated that the committee had freedom to review safety issues without pressure from management to be less conservative. They also believed that management supported their recommendations. The inspector concluded that the licensee's corrective actions are effective. This item is closed.

10.4 (Closed) Unresolved Item (50-318/89-11-04)

This issue involved a concern as to whether or not there was violation of Technical Specification 6.8.3.a regarding the manner in which STP-O-55-A-2, "Containment Integrity Verification (MODE 6)" was revised. Technical Specification 6.8.3.a allows temporary changes to be made provided there is not a change of intent to the original procedure. The specific issue was whether the change to STP-O-55-A-2 constituted a change of intent. The inspector reviewed the associated documentation and discussed this issue with the licensee. Included was a review by the POSRC which concluded that the change made was not a change of intent. Additionally, the licensee has revised its Administrative Procedure CCI-101, "Calvert Cliffs Implementing Procedure Development and Control" to provide guidance in determining if a temporary change constitutes a change of intent. The inspector concluded that the licensee's actions were acceptable. This item is closed.



#### 10.5 (Closed) Violation (50-318/89-12-001)

This open item involved the licensee's failure to provide procedural requirements for MOV stem lubrication (Violation A1) and the lack of quantifiable MOV limit switch and bypass switch settings (Violation A2).

- a. Violation A1 - The inspector reviewed the licensee's preventive maintenance requirements for Bulletin 85-03 MOVs. These procedures require the licensee to verify ample lubrication of MOV actuator stems at least every refueling outage. The licensee indicated that this preventive maintenance requirement will be established for all plant MOV's by 7/31/92.
- b. Violation A2 - The inspector confirmed that the licensee has not established quantifiable limit switch settings to deenergize MOV motor circuits. The licensee indicated however, that quantifiable settings will be established upon completion of the ongoing MOV Design Bases Review (estimated completion date (ECD) July 31, 1992). This task involves reconfirmation of MOV design bases requirements and static or dynamic signature analysis.

The inspector concluded that the licensee's proposed actions should appropriately address these items. The inspector will evaluate licensee actions during the assessment of licensee implementation of NRC Generic Letter 89-10. This item is closed.

#### 10.6 (Closed) Unresolved Item (50-318/89-12-002)

This item involved potential lubrication deficiencies in MOV main gear box assemblies. From discussions with licensee personnel and review of preventive maintenance records (PM # 2-52-M-R-8 thru 15), the inspector determined that MOV's with suspected grease problems were reinspected, cleaned, and regreased. The inspector found that in addition to quantitative (ASTM) grease sample tests, the licensee established qualitative inspection and acceptance criteria for main gear case and main gear box grease. The licensee's qualitative inspection criteria includes grease quantity, consistency, presence of foreign material, odor, and color. To ensure valid and consistent assessments of adequate grease, the licensee requires the MOV Project Manager or MOV Lubrication Specialist to confirm the acceptability of grease samples. The licensee also agreed to include specific grease inspection acceptance criteria in preventive maintenance procedures. This item is closed.

10.7 (Closed) Unresolved Item (50-318/89-12-003)

This issue involved the capability of motor operators set to close on limit to positively close valves despite wear or erosion of the valve seat. The inspector determined that BG&E considers limit switch closure like torque switch closure to be a viable and conservative method of closing MOVs. The licensee also indicated that this position was supported by historical data indicating acceptable performance from MOVs set to close on limit. However, the licensee's proposed MOV program would require periodic VOTES tests to assess valve performance, and confirm the repeatability and effectiveness of limit switch closure. The program would also consider implications of limit switch adjustments required between scheduled diagnostic tests to ensure valve closure. The licensee indicated that the proposed program will be instituted for all MOVs when MOV baseline data is established (ECD July 31, 1992).

The inspector concluded that the licensee's proposed actions should appropriately address this item. The inspector will evaluate these actions during the assessment of licensee implementation of NRC Generic Letter 89-10. This item is closed.

10.8 (Closed) Unresolved Item (50/318/89-12-004)

This issue involved the licensee's use of 2 rotor limit switch assemblies to control MOV operation. The inspector determined that the licensee understands that adjustments to the torque bypass and limit switches for 2 rotor assemblies may have undesirable effects on open and close indication lights. The licensee has implemented a modification (FEC 90-61) to require 4 pole limit switches for all plant MOVs by July 31, 1992. This action would allow MOV indicating light switches and control switches to be set and adjusted on independent rotors. The licensee indicated that training of maintenance personnel provides interim assurance that changes to bypass switch settings and limit switch settings will not adversely affect MOV position indication.

The inspector concluded that the licensee's proposed action should appropriately address this item. The inspector will evaluate this action during the assessment of licensee implementation of NRC Generic Letter 89-10. This item is closed.

10.9 (Closed) Violation (50-317/89-12-005 and 50/318/89-12-005)

This issue concerned the licensee's failure to follow procedures which resulted in incorrect MOV torque switch settings. The inspector determined that the licensee's corrective actions included inspections and torque switch adjustments for all NRC Bulletin 85-03 MOVs, procedure revisions which require MOV Project Mangers to assign all torque switch settings, and training for MOV maintenance personnel. The inspector concluded that the licensee's corrective actions were adequate. This item is closed.

10.10 (Closed) Violation (50-318/89-04-04)

This violation involved the licensee's failure to adequately document test results. The licensee's corrective actions included successful tests for greater than 10% of safety related snubbers required for MODES 5 and 6, updating administrative procedures for control of lost or damaged records, and centralizing responsibility for the development and scheduling of surveillance test procedures (STPs) and review of test results. The inspector concluded that licensee actions to address the identified deficiencies were adequate. The inspector also performed a routine review of ongoing snubber surveillance activities and found no additional problems. This item is closed.

10.11 UNR (50-317/85-27-01 and 50-318/85-25-01)

This item had remained open pending licensee's completion of corrective maintenance and walkdown of the control room HVAC system. The inspector verified during this inspection period that the following corrective maintenance had been completed by the licensee: 1) cracked panels in the air handling unit had been weld repaired; 2) airleaks at the cooling coil penetrations had been sealed; and 3) latches on duct windows had been replaced. The licensee also corrected additional deficiencies including a crack in a panel for #12 air handling unit and missing plugs in test penetration holes in the ductwork which were found during the licensee's followup walkdown. The inspector walked down the system with the licensee's cognizant system engineer on August 6, 1990, and determined that the system maintenance and material condition were adequate. This item is closed.

10.12 Closed (UNR 50-317/85-31-01 and 50-318/85-26-01)

During a previous inspection, a question was raised regarding the licensee's assumptions involved in their estimation of the total inleakage rate for the control room HVAC system in the recirculation mode. In response to this question, the licensee conducted an engineering test (ETP 86-01) to measure leakage rates through the control room HVAC isolation dampers. The inspector reviewed the completed

test procedure and noted that the measured inleakage across one shut louvered damper in the common discharge duct with the butterfly damper was 213.1 cubic feet per minute (cfm). The licensee then revised the original post-LOCA control room dose calculation based on the results from ETP 86-01. The licensee's calculation results indicated that the 30-day integrated post-LOCA control room dose with a more conservative inleakage rate increases the skin dose by 0.2 rem and the whole body dose by 0.02 rem, which is still well within the NRC limits. This item is closed.

10.13 Closed (UNR 50-317/85-31-02 and 50-318/85-26-02)

This item had remained open pending completion of the licensee's actions to correct the deficiencies identified with the louvered isolation dampers in the control room HVAC system. During this inspection period, the inspector noted that the licensee has replaced broken blade brackets and adjusted the damper linkages. The operability of these dampers was subsequently verified during an engineering test (ETP 86-01) conducted in March 1986. The test results indicated that the louvered dampers shut on a control room high radiation signal with the acceptable level of inleakage through these dampers in the recirculation mode. This item is closed.

11. Management Meeting

During this inspection, periodic meetings were held with station management to discuss inspection observations and findings. At the close of the inspection period, an exit meeting was held to summarize the conclusions of the inspection. No written material was given to the licensee and no proprietary information related to this inspection was identified.

11.1 Preliminary Inspection Findings

Unresolved Item 50-317/90-16-01 and 50-318/90-16-01, Review Readiness of Appendix R Equipment Required to Ensure Safe Shutdown.

11.2 Attendance at Management Meetings Conducted by Region Based Inspectors

<u>Date</u>	<u>Subject</u>	<u>Inspection Report No.</u>	<u>Reporting Inspector</u>
8/2/90	Security Inspection	50-317/90-20 50-318/90-19	D. Limroth
8/9/90	Emergency Preparedness Partial-Participation Exercise	50-317/90-19 50-318/90-18	E. Fox