July 30, 2002

EA-02-138

Mr. Peter E. Katz
Vice President - Calvert Cliffs Nuclear Power Plant
Constellation Generation Group
Calvert Cliffs Nuclear Power Plant, Inc.
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-317/02-04, 50-318/02-04

Dear Mr. Katz:

On June 29, 2002, the NRC completed an inspection at your Calvert Cliffs Nuclear Power Plant Units 1 & 2. The enclosed report documents the inspection findings which were discussed on July 10, 2002, with Mr. David Holm and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one preliminary finding of low to moderate safety significance (white) was identified, which does not represent an immediate safety concern. As described in Section 2PS2 of this report, on May 23, 2002, Calvert Cliffs Nuclear Power Plant failed to prepare a shipment of radioactive material to a waste processing facility in a manner such that, under conditions normally incident to transportation, the radiation level at any point on the external surface of the package would not exceed 200 millirem per hour, as specified by the Department of Transportation's (DOT) regulation, 49 CFR 173.441(a). As a result, upon arrival at the processing facility on May 24, 2002, the radiation dose rates measured on a portion of the external surface of the package were as high as 300 millirem per hour, which is in excess of the 200 millirem per hour limit specified by the regulatory requirement.

This finding was assessed using the Public Radiation Safety Significance Determination Process and was preliminarily determined to be white, i.e., a finding having low to moderate safety significance which may require additional NRC inspection. This preliminary determination was based on our assessment that the external radiation limit for the package of radioactive material, which you offered for transportation, was determined to have exceeded the external surface radiation limit established by 49 CFR 173.441(a) upon receipt, but was not greater than 5 times the regulatory limit.

Your staff took immediate corrective measures to evaluate this condition and initiated actions to preclude recurrence. These actions included initiating a formal root cause evaluation, suspending any shipments involving radiation dose rates greater than 100 millirem per hour until the root cause was identified, dispatching personnel to the vendor facility to inspect the container, and quarantining and evaluating the radiation survey instruments used to survey this particular shipment.

The finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's website at www.nrc.gov.

We believe that we have sufficient information to make a final significance determination regarding this finding. However, before we make a final decision, you have the opportunity to request a Regulatory Conference, or provide a written position on your perspectives of the facts and assumptions applied by the NRC to determine this finding and its significance. If you choose to request a Regulatory Conference, you should be prepared to meet within 30 days of the receipt of this letter. In such case, we encourage you to provide supporting documentation at least one week prior to the conference in order to facilitate effectiveness and efficiency. A Regulatory Conference for a matter of this type would be open for public observation. If you decide to provide a written response, please send your submittal to the NRC within 30 days of the receipt of this letter.

Please contact Ms. Michele Evans, Chief, Reactor Projects Branch, at (610)337-5224 within 10 business days of the date of receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, a Notice of Violation is not being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In addition, the inspectors identified two issues of very low safety significance (green). These issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating this issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at the Calvert Cliffs facility.

The NRC has increased security requirements at Calvert Cliffs Nuclear Plant, Inc., in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web Site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket Nos.: 50-317, 50-318
License Nos.: DPR-53, DPR-69

Enclosures: Inspection Report 50-317/02-04 and 50-318/02-04

Attachment 1 - Supplementary Information
Attachment 2 - Operator Licensing Report on Interaction (ROI)
Attachment 3 - Guidance on Implementation of 10 CFR 55.53(f)(2)

cc w/encl: M. Geckle, Director, Nuclear Regulatory Matters (CCNPPI)
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K. Burger, Esquire, Maryland People's Counsel
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Region I Docket Room (with concurrences)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-317, 50-318
License Nos.:	DPR-53, DPR-69
Report Nos.:	50-317/02-04 50-318/02-04
Licensee:	Calvert Cliffs Nuclear Power Plant, Inc.
Facility:	Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Location:	1650 Calvert Cliffs Parkway Lusby, MD 20657-4702
Dates:	May 19, 2002 - June 29, 2002
Inspectors:	David Beaulieu, Senior Resident Inspector Leonard Cline, Resident Inspector Ronald Nimitz, Senior Health Physicist E. Harold Gray, Senior Reactor Inspector Paul Frechette, Physical Security Inspector John Caruso, Senior Operations Engineer Christopher Welch, Resident Inspector, R. E. Ginna
Approved by:	Michele G. Evans, Chief, Projects Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000317-02-04, 05000318-02-04; Calvert Cliffs Nuclear Plant, Inc.; on 5/19-6/29/2002; Calvert Cliffs Nuclear Power Plant, Units 1 & 2. Licensed Operator Requalification, Operability Evaluations, Radioactive Material Processing and Transportation.

The inspection was conducted by resident inspectors, a senior health physicist, and regional specialist inspectors. The inspection identified two green findings, which were Non-Cited Violations, and one preliminary white finding, which was an apparent violation. The significance of most findings is indicated by their color (green, white, yellow, or red) using IMC 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be "green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation for failure to comply with the requirements of 10 CFR 55.53(f)(2) for reactivating operator licenses to support refueling outages as senior operators limited to fuel handling (LSRO).

  This finding was determined to be more than minor but of very low safety significance. It is more than minor because the use of inappropriately activated LSROs could be a precursor to operator errors which, in turn, could lead to a significant event. Specifically, improper re-activation would result in improper training which could cause errors in fuel handling activities resulting in fuel damage and potential radiological releases. The SDP is entered because the performance deficiency is related to operator license conditions. The performance deficiency was determined to be of very low safety significance (green) because more than 20% of the LSRO license reactivations to support refueling operations did not meet the requirements of 10 CFR 55.53(f)(2). No refueling events have occurred due to this training deficiency. (Section 1R11)

Cornerstone: Barrier Integrity

Green. The inspectors identified a Non-Cited Violation for inadequate design control associated with the Units 1 and 2 main steam line break (MSLB) accident analyses. The analyses credited the closure of the main feedwater isolation valves (MFIVs) to limit containment peak pressure even though in certain single failure scenarios, the valves may not fully close due to high differential pressure.

This violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," is based on the example provided in NRC Manual Chapter 0612, Appendix E, Example 3.i. The above MFIV finding is more than minor because an error identified in an accident analysis assumption requires the accident analysis to be re-performed to assure accident analysis requirements are met. The MFIV finding was determined to be of very low safety significance (green) based on the fact that when the licensee revises their

MSLB accident analyses to credit closure of the main feedwater regulating valves, it is likely to result in a net reduction in containment peak pressure. (Section 1R15.2)

Cornerstone: Public Radiation Safety

- Preliminary White. From an in-office review, the inspector identified an apparent finding of low to moderate safety significance. On May 23, 2002, the licensee failed to prepare a shipment of radioactive material to a waste processing facility in a manner such that, under conditions normally incident to transportation, the radiation level at any point on the external surface of the package would not exceed 200 millirem per hour, as specified by the Department of Transportation regulation 49 CFR 173.441(a). Upon arrival at the processing facility on May 24, 2002, the radiation dose rates, measured on portions of the external surface of the package, were as high as 300 millirem per hour, which is in excess of the limits specified by the regulatory requirement.

  The failure to properly prepare the shipment in a manner to assure conformance with the requirements of 49 CFR 173.441(a) was determined to have low to moderate safety significance, using the Public Radiation Safety Significance Determination Process. The finding involved the transportation of radioactive material in which an external radiation limit was exceeded, but was not greater than 5 times the regulatory limit. (Section 2PS2)

Reports Details

On June 19, 2002, Unit 1 started up from the refueling outage which began February 15, 2002. Unit 1 reached 100 percent power on June 24, 2002, where it remained until the end of the inspection period. Unit 2 operated at or near 100 percent power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (Partial Walkdown)

a. Inspection Scope

The inspectors conducted equipment alignment partial walkdowns to evaluate the operability of selected redundant trains while the affected train was inoperable. The walkdown included a review of system operating instructions to determine correct system lineup and verification of critical components to identify any discrepancies that could affect operability of the redundant train or backup system. The inspectors performed partial system walkdowns on the following systems:

- 1B emergency diesel generator (EDG) was inspected on June 25, 2002, while the 1A EDG was out of service for planned maintenance.
- 21 high pressure safety injection (HPSI) train was inspected on June 19, 2002, while the 22 HPSI pump was out of service for pump and motor oil samples.

The inspectors reviewed the following Calvert Cliffs Nuclear Power Plant documentation:

- Operating Instruction OI-21B-1, “1B Diesel Generator”
- Operating Instruction OI-03A-2, “Safety Injection and Containment Spray”

b. Findings

No findings of significance were identified.

1R05 Fire Protection - Fire Area Tours

a. Inspection Scope

The inspectors conducted tours of areas important to reactor safety to evaluate conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation. The inspectors used administrative procedure SA-1-100, “Fire Prevention,” during the conduct of this inspection.

The areas inspected included:

- Control room
- Intake structure
- Unit 2 emergency core cooling system pump room

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The following inspection activities were performed using NUREG 1021, Rev. 8, Supplement 1, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program," Appendix A, "Checklist for Evaluating Facility Testing Material." License reactivations for the past two year Requalification program cycle were reviewed for conformance with the requirements of 10 CFR 55.53 (f)(2).

b. Findings

Introduction

Green. A Non-Cited Violation of 10 CFR 55.53(f)(2) was identified regarding the licensee's methods and standards used to reactivate operator licenses to support refueling outages.

Description

An unresolved item (URI 50-317; 50-318/01-12-01) was identified during the biennial Licensed Operator Requalification Program inspection conducted the week of November 12, 2001, and documented in NRC Inspection Report 50-317; 50-318/01-012. The site practice had been to have staff licenses stand one shift of under-instruction watch in the control room, conduct a tour of refueling equipment, and attend four hours of pre-refueling classroom training as a basis for reactivation as a limited refueling SRO. 10 CFR 55.53(f)(2) requires, in part, that the Senior Reactor Operators Limited to Fuel Handling (LSRO) stand one shift of under-instruction watch in the position to which the individual will be assigned (i.e., on the refueling floor as a Fuel Handling Supervisor). The NRC Office of Nuclear Reactor Regulation and Office of General Council provided guidance and clarification for resolution of this issue in Report on Interaction No. 01-16, "Interpretation of 10 CFR 55.53 - License Reactivation," which is included as Attachment 2 to this inspection report.

Accordingly, the performance deficiency was that the licensee's methods and standards used to reactivate LSROs was inadequate in that the individuals stood their entire reactivation shift in the control room rather than in the position to which the individual would be assigned (i.e., on the refueling floor as a Fuel Handling Supervisor).

Analysis

The inspector evaluated the issue relative to NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning

Screening," and determined that the performance deficiency is more than minor because the use of inappropriately activated LSROs could be a precursor to operator errors which, in turn, could lead to a significant event. Specifically, improper re-activation would result in improper training which could cause errors in fuel handling activities resulting in fuel damage and potential radiological releases. NRC IMC 0612, Appendix B, second Section C, Question 9, specifies that operator requalification findings related to operator license conditions are evaluated using NRC IMC 0609, "Significance Determination Process." NRC IMC 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," Flowchart Blocks #24 and #27 address operator requalification performance deficiencies related to operator license conditions. IMC 0609, Appendix I, Flowchart Block #27, specifies that when more than 20% of the records reviewed by the inspector have deficiencies, the finding is of very low safety significance (green). Based on the inspector finding that more than 20% of the LSRO license reactivations to support refueling operations did not meet the requirements of 10 CFR 55.53(f)(2), this performance deficiency was determined to be of very low safety significance (green). No refueling events have occurred due to this training deficiency.

Enforcement

The licensee initiated Issue Report IR3-000-855 to document this problem within their corrective action program. The licensee ceased their prior practices at the time of the inspection and have initiated a corrective action item to revise their Operator Requalification Program Manual to change their methods for re-certifying inactive SRO license holders to perform Fuel Handling Supervisor duties. The corrective actions taken or planned by the licensee appeared to be reasonable.

10 CFR 55.53(f)(2) requires LSROs that wish to reactivate their licenses to complete at least one shift under-instruction under the direction of a senior operator and "in the position to which the individual will be assigned" (in this case as a Fuel Handling Supervisor on the refueling floor). Contrary to this requirement, the licensee reactivated their inactive LSROs by allowing them to complete one shift of under-instruction as a control room supervisor rather than on the refueling floor as a Fuel Handling Supervisor. However, because the violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (Issue Report IR3-000-855), it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 50-317; 50-318/02-004-01)** Unresolved Item 50-317; 50-318/01-12-01 is **closed**.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed performance-based problems involving a selected in-scope structure, system, or component (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the most recent system health reports and system functional failures of the last two years. The following SSC was reviewed:

- Unit 1 Containment air coolers (CAC). The licensee classified this system as (a)(1) in January 2002 because it exceeded its functional failure performance criteria of less than three functional failures over two years. The failures were due to a ground on the fast speed windings for 13 CAC, a degraded slow speed starter contact block for 12 CAC, and a short between the 13 CAC starter and its starter mounting bracket. The licensee's corrective action plan to address these conditions was documented in Issue Report IR3-080-025.

The inspectors also reviewed the following Calvert Cliffs Nuclear Power Plant documentation:

- Station Procedure MN-1-112, "Managing System Performance"
- Maintenance Rule Scoping Document, Revision 18
- Maintenance Rule Indicator Report, May 2002

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For the selected maintenance orders (MO) listed below, the inspectors verified: (1) risk assessments were performed in accordance with Calvert Cliffs procedure NO-1-117, "Integrated Risk Management;" (2) risk of scheduled work was managed through the use of compensatory actions; and (3) applicable contingency plans were properly identified in the integrated work schedule.

- MO1200103411 — On June 25 and 26, 2002, the 1A emergency diesel generator was removed from service for maintenance.
- MO2200104093 — On June 19, 2002, the Unit 2 auxiliary feedwater system cross connect valve, 2-CV-4550, was removed from service for maintenance.
- MO2199901382 — On June 10, 2002, ventilation flow to the 21 switchgear room was temporarily removed from service to support replacement of the 21 switchgear ventilation air conditioning compressor.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Containment Spray Pump and Charging Pump Operability

a. Inspection Scope

The inspectors reviewed operability determinations to assess the correctness of the evaluations, the use and control of compensatory measures if needed, and compliance with technical specifications. The inspector's review included a verification that the operability determinations were made as specified by the licensee's procedure NO-1-106, "Functional Evaluations/Operability Determination." The technical adequacy of the determinations was reviewed and compared to technical specifications, the final safety analysis report, and associated design basis documents. The following evaluations were reviewed:

- Operability of the Unit 1 containment spray pumps following the pump overhaul completed during the Unit 1 2002 refueling outage. To increase the Unit 1 containment spray system's margin of safety with respect to design basis flow rate, the licensee had replaced each pump's original 10-1/16 inch impeller with a 10-1/4 inch impeller.

- Based on an inspector question, the licensee assessed charging pump operability considering past evidence that a charging pump relief valve commonly lifts when a third charging pump starts when the reactor coolant system is at normal operating pressure. The licensee's operability assessment stated that even if two charging pump relief valves opened and failed to reseat, there would be sufficient charging flow following a loss of coolant accident and therefore, the charging pumps were determined to be operable. Their assessment characterized the lifting relief valves as a reliability issue, not an operability issue. The licensee is evaluating options to minimize the likelihood that relief valves lift during three pump operation.

Findings

No findings of significance were identified.

.2 Containment Pressure Response Analysis for Main Steam Line Break

a. Inspection Scope

The inspector reviewed the licensee's operability assessment associated with Issue Report IR3-052-140, which described an inspector-identified deficiency regarding Design Calculation CA05892, “Containment Response to Old Steam Generator and Replacement Steam Generator Design Basis Accident for the Updated Final Safety Analysis Report,” and Design Calculation CA05684, “Steam Line Break for Containment for the Replacement Steam Generators,” Revision 4.

b. Findings

Introduction

Green. The inspector identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” regarding inadequate main steam line break (MSLB) accident analyses for Units 1 and 2.

Description

To support the replacement of Unit 1 steam generators, the licensee prepared Design Calculations CA05892 and CA05684 that re-analyzed the containment pressure response described in Updated Final Safety Analysis Report (UFSAR), Chapter 14.20.3, “Main Steam Line Break.” Containment peak pressure is dependent upon the initial amount of water in the ruptured side and the amount of feedwater added before feedwater is isolated, particularly during the first 180 seconds. The MSLB accident analysis for the old and new steam generators at Units 1 and 2 incorrectly assumes that the main feedwater isolation valves (MFIVs) will completely close on a steam generator isolation signal or a containment spray actuation signal. The inspector identified that following the single failure of any upstream pump to trip, the motor operated FWIVs would not completely close due to the high differential pressure across the valves.

According to Design Calculation CA03474, “Thrust Calculations for Generic Letter 89-10 Motor Operated Valves,” the FWIVs are analyzed for a maximum closing differential pressure of 275 psid (the discharge pressure of the condensate pumps, which do not receive a trip signal.) The UFSAR, Chapter 14.20, safety analysis describes several single failures that the inspector found would result in exceeding 275 psid across the MFIVs including: (a) main feedwater pump fails to trip (shutoff head of 1165 psi plus 275 psi condensate pump pressure for a total of 1440 psid across the MFIVs); (b) condensate booster pump fails to trip (shutoff head is 361 psi plus condensate pump discharge pressure of 275 psi for a total of 636 psid across the MFIVs); and (c) heater drain pump fails to trip (shutoff head is 575 psi across the MFIVs).

The licensee documented the finding in Issue Report IR3-052-140 and prepared an operability assessment. Because the error existed in their previous MSLB analyses, their operability assessment covered Unit 1, as well as Unit 2, whose steam generators have not yet been replaced. The operability assessment noted that neither the old or new MSLB analysis credited the closure of the main feedwater regulating valves which close in 20 seconds. The Unit 1 analysis assumes that MFIVs close in 65 seconds and the Unit 2 analysis assumes 80 seconds for MFIV closure. Because the containment peak pressure occurs only 5 minutes into the event, reducing the amount of feedwater added during the first few minutes will reduce containment peak pressure. Although the feedwater regulating bypass valve travels to 56 percent open, the total amount of feedwater added during the first few minutes is less than the current analysis and therefore, containment peak pressure is less, when crediting the closure of the main feedwater regulating valves. Accordingly, the containment will remain below its 50 psi design pressure following a MSLB.

As corrective actions, the licensee plans to revise the MSLB accident analyses for Units 1 and 2 to credit closure of the main feedwater regulating valves. In addition, they plan to evaluate the design calculation for other motor operated valves to determine if there are other examples where they did not account for a single failure in determining the maximum differential pressure the valve is credited to operate.

To assess the acceptability of the licensee's operability assessment and their planned change to the MSLB accident analyses, the inspector reviewed NUREG 0138, Issue 1, "Treatment of Non-Safety Grade Equipment in Evaluations of Postulated Steam Line Breaks." NUREG 0138 states, "If the single active failure postulated for this accident is the failure of the appropriate safety grade main feedwater isolation valve to function, then credit is taken for closing the non-safety grade main feedwater control valve or tripping the main feedwater pump in that line....It is the staff position that utilization of these components as a backup to a single failure in safety grade components adequately protects the health and safety of the public."

The licensee had several opportunities to identify that the MSLB accident analysis inappropriately credited the closure of the MFIVs. The licensee began crediting MFIV closure in 1983 in response to NRC Bulletin 80-04, "Analysis of a Main Steam Line Break with Continued Feedwater Addition," when concerns with insufficient closing thrust for motor operated valves were not well known. However, the insufficient closing thrust for the MFIVs should have been identified in response to NRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." In addition, the licensee should have identified the MFIV issue in subsequent revisions to the MSLB accident analysis such as the revision that reflects the Unit 1 replacement steam generators.

Analysis

Based on the example provided in NRC Manual Chapter 0612, Appendix E, Example 3.i, the finding is more than minor because an error identified in an accident analysis assumption requires the accident analysis be re-performed to assure accident analysis requirements are met. The MFIV finding was determined to be of very low safety significance (green) because when the licensee revises their MSLB accident analyses to credit closure of the main feedwater regulating valves, it is expected to result in a net reduction in containment peak pressure.

Enforcement

10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that "measures shall be established to assure that the applicable regulatory requirements and the design basis...are correctly translated into specifications, drawings, procedures, and instructions." Contrary to this, the design basis, as described in the UFSAR, Section 14.20.3 and a pending revision to reflect the Unit 1 replacement steam generators (Design Calculation CA05892), were not correctly translated into the specification for the MFIVs. As a result, the UFSAR Chapter 14.20.3 (including the pending Unit 1 update) MSLB analysis inappropriately credited the closure of the MFIVs to isolate feedwater to the ruptured side, which erroneously resulted in a lower calculated containment peak pressure. However, because the violation was of very low safety significance and because the issue was entered into the licensee’s corrective action program, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 50-317; 50-318/02-004-02)**

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test procedures and associated testing activities for selected risk significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; and (6) equipment was returned to the status required to perform its safety function. The following maintenance orders were reviewed:

- MO 1200201688, 11B safety injection header check valve, 1-SI-128, is binding, determine cause and repair as needed, that was retested utilizing procedures STP O-67C-1, “Miscellaneous Check Valve Testing,” and STP O-65J-1, “Safety Injection Check Valve Quarterly Operability Test.”
- MO 1200102923, Install a new style KOP-FLEX coupling on 12 auxiliary feedwater pump and turbine, that was retested utilizing procedure STP O-5A-1, “Auxiliary Feedwater System Quarterly Surveillance Test.”

- MO 1200201732, Fabricate and install a new orifice plate (1FO4507) for the 12 turbine-driven auxiliary feedwater pump oil coolers cooling supply flow, that was retested utilizing procedure STP O-5A-1, “Auxiliary Feedwater System Quarterly Surveillance Test.”

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

For the Unit 1 refueling outage, inspectors verified that licensee control of Unit 1 safety-related equipment was in accordance with administrative procedure NO-1-103, “Conduct of Lower Mode Operations,” and verified operators were tracking and maintaining minimum essential equipment status in accordance with administrative procedure, NO-1-207, “Nuclear Shift Operations Turnover.” During this period the inspectors also reviewed the following activities related to the Unit 1 refueling outage for conformance with the applicable procedures, and witnessed selected activities associated with each evolution:

- Containment Restoration
- Preparations for entering Modes 4, 3, 2, and 1
- Plant Heatup and Startup Activities

The inspectors reviewed licensee’s analyses and corrective actions associated with the following outage related issue reports:

- IR3-062-023, A loud noise occurred on May 10, 2002, during the initiation of shutdown cooling

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk-significant systems, structures, and components (SSCs) to assess whether the SSCs satisfied technical specifications, updated final safety analysis report, technical requirements manual, and licensee procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- STP O-5A-1, “Auxiliary Feedwater System Quarterly Surveillance Test”

- STP O-13-1, “Shutdown Engineered Safety Feature Actuation System Logic Test”
- STP M-510-CT1, “Reactor Protection System Steam Generator Level Transmitter Calibration”

b. Findings

No findings of significance were identified.

E2.2 Steam Generator Replacement Project

a. Inspection Scope

The inspector reviewed a portion of the radiographs (RTs) of the completed post-weld heat-treated reactor coolant system pipe welds and the pre-post weld heat treatment RTs of the Steam Generator (SG) 11 and SG12 girth welds and related RT procedure to verify their adequacy. The ultrasonic testing data sheets for reactor coolant system welds FW-1, FW-2, and FW-3 on SG11 and the applicable procedures were also reviewed for adequacy. The inspector observed the control of work in progress on the SG11 and SG12 girth welds to ensure that acceptable welds, in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, were achieved. In addition, the inspector reviewed the engineering evaluation regarding Issue Report IR3-082-457 on alignment stresses during girth weld fit-up of SG11 to verify the issue was appropriately resolved.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspector conducted an in-office review of the circumstances involving a shipment, containing radioactive materials, made on May 23, 2002, (Shipment No.02-087) from Calvert Cliffs Nuclear Power Plant to a waste processing vendor facility in Oak Ridge, Tennessee. The review included examination of the licensee’s performance relative to the preparation of the shipment. The following documents were reviewed:

- Issue Report IR3-077-457
- Low Level Waste Manifest - Shipment No. 02-087
- Shipper outgoing radiation surveys - initial and verification (Shipment No. 02-087)
- Vendor incoming full vehicle survey No. 02-1622, dated May 24, 2002

- Vendor Report to the State of Tennessee dated July 7, 2002. Subject: Calvert Cliffs Sealand Greater Than 200 Millirem per Hour on Contact
- Calvert Cliffs Summary Memorandum - May 29, 2002

The review was against applicable requirements contained in 10 CFR 71, “Packaging and Transportation of Radioactive Material,” and 49 CFR parts 170 through 189, “Transportation,” as applicable.

b. Findings

Introduction

The inspector identified a finding having low to moderate safety significance involving the licensee’s failure to prepare a shipment of radioactive material to a waste processing facility on May 23, 2002, in a manner that, under conditions normally incident to transportation, the radiation level at any point on the external surface of the package would not exceed 200 millirem per hour, as specified by the Department of Transportation’s (DOT) regulation, 49 CFR 173.441(a), “Radiation Level Limitations.” Upon arrival at the processing facility on May 24, 2002, the radiation dose rates, measured on the external surface of the package, were in excess of the limits specified by the regulatory requirement. The finding constitutes an apparent violation of 10 CFR 71.5, “Transportation of Licensed Materials,” which requires compliance with the applicable requirements of the DOT regulations in 49 CFR Parts 170 through 189.

Description

On May 23, 2002, the licensee shipped a box trailer (package) containing radioactive waste materials from its Calvert Cliffs facility to a vendor facility in Oak Ridge Tennessee for processing. The shipment (02-087) consisted of compacted and non-compacted radioactive waste, and was shipped as exclusive use, low specific activity. The total activity was 100 millicuries of solid/metal mixed oxides. The licensee’s radiation survey of the package performed prior to shipping indicated that the maximum radiation level on any external surface of the package was 70-80 millirem per hour.

When the shipment arrived at the vendor’s facility, a receipt radiation survey was performed by the vendor (Radiation Survey No. 02-1622, dated May 24, 2002.) The radiation survey indicated that contact radiation dose rates on an external surface of the package (i.e., at one point on the upper right rear side of the box trailer, 12 feet from the back; and 12 feet high) exceeded 200 millirem per hour. The vendor surveyed the area with two independent radiation survey meters and found radiation dose rates on contact with this external surface were 250 and 300 millirem per hour, respectively. The vendor also surveyed the area with two other radiation survey instruments having a maximum range of only 200 millirem per hour; and both indicated off-scale readings, i.e., radiation dose rates greater than 200 millirem per hour on contact. All of the instruments used by the vendor were within their calibration due dates. The vendor informed the licensee of this condition on May 28, 2002.

Analysis

The licensee's failure to ensure radiation levels did not exceed applicable DOT dose rate limits under conditions normally incident to transportation is a performance deficiency since compliance with the requirement was reasonable and within the licensee's ability to achieve. However, the occurrence did not represent an immediate safety concern since: (1) the potential existed only during transport of the package (about a day); (2) radiation levels were not significantly in excess of regulatory limits; and, (3) the specific area of elevated radiation level was relatively inaccessible to members of the public.

Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function; and was not the result of any willful violation of NRC requirements or licensee procedures. This finding is more than minor in that the issue was associated with the Transportation Packaging attribute of the Public Radiation Safety cornerstone; and the issue affected the objective of this cornerstone in that failure to comply with the radiation limits applicable to the transportation of radioactive materials in the public domain may compromise public health and safety relative to exposure to radioactive materials resulting from routine civilian nuclear reactor operation.

The licensee's failure to prepare a shipment of radioactive material in a manner that, under conditions normally incident to transportation, the radiation level at any point on the external surface of the package would not exceed 200 millirem per hour, was preliminarily determined to have low to moderate safety significance (White) using the Public Radiation Safety Significance Determination Process. The finding involved radioactive material control relative to the transportation of radioactive materials. In this case, a radiation limit (specified by a specific regulatory requirement, 49 CFR 173.441) was exceeded relative to an external radiation level specification, but was not greater than 5 times the regulatory limit.

<u>Enforcement</u>

10 CFR 71.5 requires each licensee who transports licensed materials on public highways to comply with the requirements of the DOT regulations in 49 CFR Parts 170 through 189. 49 CFR 173.441(a), "Radiation Level Limitations," requires that each package of radioactive material offered for transportation be designed and prepared for shipment so that, under conditions normally incident to transportation, the radiation level does not exceed 200 millirem per hour at any point on the external surface of the package.

On May 23, 2002, Calvert Cliff Nuclear Power Plant shipped radioactive waste material to vendor processing facility in Oak Ridge, Tennessee; but failed to prepare the shipment so that, under conditions normally incident to transportation, the radiation level would not exceed 200 millirem per hour at any point on the external surface of the package, as required by 49 CFR 173.441(a). Specifically, on arrival at the processing facility on May 24, 2002, the vendor measured radiation levels between 250 and 300 millirem per hour on a portion of the external surface of the package.

The licensee documented this issue in its corrective action program as Issue Report IR3-002-1009. The licensee also initiated immediate actions to preclude recurrence, including initiation of a formal root cause evaluation and suspension of shipments involving radiation dose rates greater than 100 millirem per hour until the root cause of this occurrence was fully understood. Additionally, the licensee quarantined and evaluated the radiation survey instruments that were used for the radiation surveys for this particular shipment, and dispatched personnel to the vendor facility to inspect the container and gather information. This apparent violation is being considered for escalated enforcement consistent with the NRC Enforcement Policy, NUREG-1600. **(AV 50-317; 50-318/02-004-03)**

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization

a. Inspection Scope

The following activities were conducted to determine the effectiveness of Calvert Cliffs behavior observation portion of the personnel screening and fitness-for-duty (FFD) programs, as measured against the requirements of 10 CFR 26.22 and the Calvert Cliffs Fitness for Duty Program documents.

Five supervisors representing the Mechanical, Instrument and Controls, Contract Administration, Engineering Assessment, and Information Technology departments were interviewed, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. Two Access Authorization/Fitness-for-Duty self-assessments, two semi-annual FFD testing data reports, an audit, event reports, and loggable events for the four previous quarters were reviewed. Five individuals who perform escort duties were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were also reviewed.

b. Findings

No findings of significance were identified.

3PP2 Access Control

a. Inspection Scope

The following activities were conducted during the period June 3-7, 2002, to verify that Calvert Cliffs has effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10 CFR 73.55(d), and the Physical Security Plan and Procedures.

Site access control activities were observed, including personnel and package processing through the search equipment during peak ingress periods and vehicle searches. Testing of all access control equipment; including metal detectors, explosive material detectors, and X-ray examination equipment, was observed. The access control event log, an audit, and three maintenance work requests were also reviewed.

A review was conducted of two Issue Reports (IRs) generated and entered into the licensee's corrective action program to address concerns identified during the previous inspection conducted in April, 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

40A1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data for the below listed cornerstones to verify individual PI accuracy and completeness. This inspection examined data and plant records from 1999 through the second quarter of 2002, including review of PI Data Summary Reports, and chemistry records.

- Units 1 and 2 Fitness-for-Duty
- Units 1 and 2 Personnel Screening
- Units 1 and 2 Protected Area Security Equipment
- Units 1 and 2 Reactor Coolant System Activity

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

(Closed) Licensee Event Report 50-318/2002-001: Pump Flexible Drive Gear Wear Causes Emergency Diesel Generator Inoperability

On January 24, 2002, during the biennial inspection of the 2A emergency diesel generator (EDG), unusual wear was discovered on the flexible drive gear assembly for the engine's lube oil pump. Replacement of the worn gear assembly required that the EDG remain out of service for a time period greater than allowed by the plant's technical specifications. Enforcement Discretion was granted by the NRC to allow Unit 2 to continue operation with the 2A EDG out of service until February 2, 2002, while the flexible drive gear assembly was replaced. The apparent cause of the wear was that the backlash for the drive gears for the lube oil pump was zero, and the alignment of two bearing bores on the gear assembly were out of specification. The licensee's causal analysis determined that these two conditions had existed since original assembly. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented this condition in its corrective action program as Issue Report IR3-

080-051. Further details regarding this Enforcement Discretion are described in NRC Inspection Report 50-317/01-014, 50-318/01-014. LER 50-318/2002-001 is **closed**.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 10, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

Attachment 1

Supplementary Information

a. Key Points of Contact

P. Katz, Vice President
K. Neitmann, Plant General Manager
L. Weckbaugh, Manager, Nuclear Support Services
D. Holm, Manager, Nuclear Operations
M. Korsnick, Manager, Work Management
J. Spina, Manager, Nuclear Maintenance
M. Geckle, Director, Nuclear Regulatory Matters
G. Gwiazdowski, Director, Nuclear Security/Emergency Planning
R. Szoch, General Supervisor, Plant Engineering
J. Alvey, General Supervisor, Security Operations
J. Evans, Acting General Supervisor Nuclear Training
P. Harrison-Dean, Fitness for Duty Program Manager
D. Dean, Supervisor, Security Access
J. Hornick, Supervisor Initial Training Unit
S. Sanders, General Supervisor, Radiation Safety
W. Paulhardt, Assistant General Supervisor, Radiation Safety
E. Roach, Radiation Safety Supervisor, Material Processing
D. Jordan, Principal Radiation Safety Technician
T. Kirkham, Senior Plant Health Physicist
M. Yox, Licensing Analyst, Regulatory Matters

b. List of Items Opened, Closed or Discussed

Opened

50-317; 50-318/02-004-03	AV	Failure to prepare a shipment of radioactive material so as not to exceed the transportation radiation level limits of 49 CFR 173.441(a). (Section 2PS2)

Closed

50-317; 50-318/01-012-01	URI	Licensee methods and standards used to reactivate licenses to support refueling outages appeared to be inconsistent with 10 CFR 55.53(f)(2). (Section 1R11)
50-318/2002-001-00	LER	Pump flexible drive gear wear causes emergency diesel generator inoperability. (Section 4OA3)

(Attachment 1 - Continued)

Opened and Closed

50-317; 50-318/02-004-01	NCV	Failure to comply with the requirements of 10 CFR 55.53(f)(2) for reactivating licensees to support refueling outages as senior operators limited to fuel handling. (Section 1R11)
50-317; 50-318/02-004-02	NCV	Failure to comply with the requirements of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," regarding the Unit 1 and 2 main steam line break accident analyses. (Section 1R15)

c. List of Documents Reviewed

Steam Generator Replacement Project

QEP 12.6, "Radiographic Examination," Rev. 1
CCNPP RSG Drawing -Weld Map, Large Bore Non-Destructive Examination, Rev. 1
WGI-PS4-UT-1, "UT by P-Scan of Austenitic and Ferritic Pipe Welds," Rev. 0
SGPR-UT-2, "UT of Class 1 and 2 Vessel Welds," Rev. 0
SGPR-PS4-UT-2, "UT by P-Scan of Ferritic Vessel Welds Over 2 Inches Thick," Rev. 0
SGRP-UT-3, "UT of Ferritic Piping Welds (RCS)," Rev. 0
PDI-UT3, "Thru-Wall Sizing by Ultrasonics"
A sample of girth weld radiographs for steam generators 11 and 12 grind-outs, at the 1/3 and 2/3 weld completion levels.

Attachment 2

Operator Licensing Report on Interaction (ROI)

Subject:	Interpretation of 10 CFR 55.53 - License Reactivation		
Type of Action:	Waiver Policy Interpretation: Request for HQ Action		
From:	R. Conte, Chief, Operational Safety Branch RI/DRS/OSB	Date:	12/18/01
To: Info.:	D. Trimble, Chief, IOHS ADAMS Assession No. ML	Proposed Due Date:	1/31/02

Background/Issue:

During a recent Licensed Operator Requalification Program (LORP) inspection at Calvert Cliffs, the inspectors noted that the site practice has been to have staff licensees stand one shift under the direction of the senior operator in the control room, conduct a tour of refueling equipment, and attend four hours of pre-fuel-move classroom training as a basis for reactivation as a limited refuel SRO. This practice appears to be inconsistent with the requirements of 10 CFR 55.53(f)(2) that requires in part that the under-direction shift be stood in the position to which the individual will be assigned. The under-direction time in the control room appears to not have met the intent of the rule.

Region I reviewed ROI94-38 and the related questions (#253, 278 and 289) in NUREG-1262 that provided guidance that would indicate that Calvert Cliff's practice is unacceptable for the under-direction time and is therefore a violation. However, in ROI 94-38, Region I also noted that IOLB had intended to revise 55.53 in the long term to further address this area and that this ROI had provided an interpretation of NRC regulations but had not received OGC concurrence.

Region I intends to make this an unresolved item for Calvert Cliffs pending IOLB direction and resolution.

Recommended Action/Resolution:

Region I recommends that clarification be issued to licensees concerning the requirements 10 CFR 55.53(f)(2) as it applies to this situation with concurrence from OGC. If the practice is deemed unacceptable as noted above, Region I will take appropriate enforcement action.

Final Action/Resolution:

As discussed in the attachment, Calvert Cliffs' practice is unacceptable. We understand that Region I will take appropriate enforcement action. The program office is considering the appropriate method of promulgating this guidance to other licensees and the need to clarify the regulation.

(Attachment 2 - Continued)

File Subject(s:)	10CFR55; NUREG-1021 Specify Other:	
Distribution:	OLBCs, ROI logbook Post on Web: No	

Signature/Concurrences

Originator:	R. Conte, Chief, Operational Safety Branch RI/DRS/OSB **/RA/**	Date
OGC:	S. Treby **/RA/**	Date: 2/7/02
IOHS CH:	D.C. Trimble, Chief, IOHS/IEHB **/RA/**	Date: 1/24/02
IOLB CH:	T. Quay, Chief, IEHA **/RA - D. Trimble for**/	Date: 1/24/02

Distribution Completed by IOLB Secretary (Initials): Date:

Attachment 3

Guidance on Implementation of 10 CFR 55.53(f)(2)

Section 55.53(f)(2) clearly requires senior operators limited to fuel handling, who wish to reactivate their licenses, to complete at least one shift under the direction of a senior operator and "*in the position* to which the individual will be assigned." The question is whether an operator, who will be assigned only to supervise fuel handling in the fuel handling area, satisfies the regulatory requirement when the individual performs that shift in the control room. The answer to that question turns upon the meaning of the term "in the position," which, in the staff's judgment, refers to the <u>scope of duties</u> to which the operator will be assigned.

Although neither the statements of consideration nor the answers to public questions (in NUREG-1262) associated with the 1987 rule change (which added this requirement) provide definitive guidance regarding the specific intent of the quoted passage, the staff expects that the under-direction watch would be performed in the fuel handling area during refueling operations. This expectation is a logical and reasonable extension of the following regulatory requirements (with <u>emphasis</u> and explanations added, as appropriate):

10 CFR 55.4 defines *systems approach to training* (SAT) to mean a training program that includes, among other things, a systematic analysis of the <u>jobs to be performed</u>, learning objectives derived from the analysis, and training implementation based on the learning objectives. If a facility licensee wishes to activate and use a licensed senior operator only to perform fuel handling duties, it only makes sense to complete the under-direction training activities on the refueling floor, where the operator will actually perform the job, rather than in the control room.

10 CFR 50.54(m)(2)(iv) requires facility licensees to have present, during alteration of the core of a nuclear power unit including fuel loading or transfer, a person holding a senior operator license or a senior operator license limited to fuel handling to <u>directly</u> supervise the activity and, during this time, the licensee shall <u>not assign other duties</u> to this person. If a facility licensee plans to use a senior operator to perform only fuel handling duties, it makes little sense for that senior operator to perform the under-direction watch in the control room, which is a remote location with no direct involvement in the fuel handling activities, and to train for activities that fuel handlers are specifically prohibited from performing while supervising fuel handling.

In order to maintain an active license status, 10 CFR 55.53(e) requires licensees to *actively perform the functions of an operator or senior operator* for a minimum number of shifts per calendar quarter. 10 CFR 55.4 defines *actively performing the functions of an operator or senior operator* to mean that an individual has a position on the shift crew that requires the individual to be licensed as defined by the facility's technical specifications, and that the individual carries out and is responsible for the <u>duties covered by that position</u>. Therefore, by analogy, the training to reactivate a license must involve the duties of the operator in that position.

(Attachment 3 - Continued)

Although it may not appear logical for an active senior operator who normally stands watch in the control room to oversee the first fuel handling shift of an inactive senior operator who normally only stands watch on the refueling floor, in reality, that probably makes more sense than having the inactive senior operator stand one under-direction watch in the control room doing things that are generally unrelated to the activities that he or she will subsequently perform without supervision on the refueling floor.