July 17, 2003

Mr. George Vanderheyden
Vice President - Calvert Cliffs Nuclear Power Plant
Constellation Generation Group
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000317/2003003 AND 05000318/2003003

Dear Mr. Vanderheyden:

On June 28, 2003, the US Nuclear Regulatory Commission (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 9, 2003, with Mr. Neitmann 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, the inspectors identified one issue of very low safety significance (Green). No violations of regulatory requirements occurred.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspection licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year '02, and the remaining inspection activities for Calvert Cliffs are scheduled for completion in July 2003. The NRC will continue to monitor overall safeguards and security controls at Calvert Cliffs.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any), 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.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000317/2003003 and 05000318/2003003
w/Attachment: Supplemental Information

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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.:	05000317/2003003 and 05000318/2003003
Licensee:	Calvert Cliffs Nuclear Power Plant, Inc. (CCNPPI)
Facility:	Calvert Cliffs Nuclear Power Plant
Location:	1650 Calvert Cliffs Parkway Lusby, Maryland 20657-4702
Dates:	March 30, 2003 - June 28, 2003
Inspectors:	Thomas Hipschman, Senior Resident Inspector Joseph M. O'Hara II, Resident Inspector Peter Wilson, NRR Senior Reactor Analyst Marvin Sykes, NRR Reactor Operations Engineer Joe Furia, Senior Health Physics Inspector (Sections 2OS1, 2OS2, 2OS3, and 2OA4) E. H. Gray, Senior Reactor Inspector (Sections 1R08, 4OA5) Tom Burns, Reactor Inspector (Sections 1R08, 4OA5) Neil Perry, Senior Project Engineer
Approved by:	James M. Trapp, Chief, Projects Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

050003172003-003, 050003182003-003; 3/30/2003-6/28/2003; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Event Follow-up.

The report covered a 3 month period by resident and NRC Headquarters inspectors and announced inspections by regional specialists. One Green finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (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. NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a finding because the work practices during a turbine governor valve control circuit troubleshooting activity were inadequate and resulted in a reactor trip control.

This finding is greater than minor because it affected an attribute and the objective of the Initiating Events Cornerstone in that the work practices inadequacies resulted in a perturbation in plant stability by causing a reactor trip. The finding is of very low safety significant in accordance with Phase 1 of the reactor safety SDP because, although it caused a reactor trip, it did not increase the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a combination of a reactor trip and loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. (Section 4OA3.2)

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near 100 percent power for the entire inspection period.

Unit 2 completed its refueling outage and achieved criticality on April 20, 2003 and reached 100 percent power on April 24, 2003. On May 28, 2003, the unit was inadvertently tripped from full power as a result of a turbine troubleshooting activity. Unit 2 returned to 100 percent power on May 29, 2003, and operated at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors verified that systems, structures, and components associated with the 4KV, 13KV and various pump room coolers would remain functional when challenged by hot weather conditions expected to occur during the onset of summer. The inspectors reviewed the Updated Final Safety Analysis Report, Individual Plant Examination of External Events, and Technical Specifications. Additionally, the inspectors walked down selected areas around 4KV, 13KV transformers and various emergency core cooling system (ECCS) pump room coolers to verify that the fans and coolers appeared to be functioning as designed per operating procedures listed in the attachment.

The inspectors also verified that systems, structures, and components associated with the various pump rooms in the auxiliary and turbine buildings would remain functional when challenged by seasonal weather conditions, such as hurricanes. The inspectors reviewed the Updated Final Safety Analysis Report, Individual Plant Examination of External Events, Technical Specifications and ERPIP 3-0, Attachment 20, “Severe Weather.” Additionally, the inspectors walked down selected areas around various pump rooms to verify that the water tight doors appeared to be functioning as designed.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns.

The inspector performed five partial system walkdowns during this inspection period. The inspectors conducted equipment alignment partial walkdowns to evaluate the operability of a selected redundant train 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:

- Unit 2, high pressure safety injection - portions inside the Containment Building inspected during the outage
- Unit 2, auxiliary feed water system - portions inside the Containment Building inspected during the outage
- Unit 1, component cooling water system - portions inside the Auxiliary Building
- Unit 1 auxiliary feed water system - portions inside the Auxiliary Building
- Unit 1, 1A emergency diesel generator

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

- OI-3, “Unit 2 High Pressure Safety Injection”
- OI-32A-2, “Unit 2- Auxiliary Feedwater System”
- OI-32A-1, “Unit 1- Auxiliary Feedwater System”
- OI-16-1, “Component Cooling”
- OI-21A-1, “Unit 1A Emergency Diesel”

Complete System Walkdown.

The inspectors performed a complete system walkdown of the 125 Vdc (Units 1 and 2) and the vital 120 Vac (Unit 2) systems, both risk-important mitigating systems. The walkdown was conducted to identify any discrepancies between the existing equipment lineup and the required lineup. Inspection attributes included verifying that electrical power was available as required; major system components were correctly labeled, lubricated, cooled, and ventilated; hangers and supports were correctly installed and functional; essential support systems were operational; and ancillary equipment and debris did not interfere with system performance. Maintenance work requests on the system for deficiencies that could affect the ability of the system to perform its function were reviewed. Documentation associated with unresolved design issues such as temporary modifications, operator work-arounds, and items tracked by plant engineering were also reviewed to assess their collective impact on system operation.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of thirteen 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.

- OC Emergency Diesel Generator (EDG) Building
- 1A Emergency Diesel Generator (EDG) Building
- 1B Emergency Diesel Generator (EDG) Room
- Unit 1 Emergency Switchgear Rooms
- Unit 1 Emergency Core Cooling System Pump Rooms
- Unit 2 Emergency Core Cooling System Pump Rooms
- Unit 1 Intake Structure
- Unit 2 Intake Structure
- Unit 1 Service Water Pump Room
- Unit 1 Auxiliary Feedwater Pump Room
- 2A Emergency Diesel Generator (EDG) Building
- 2B Emergency Diesel Generator (EDG) Room
- Unit 1 4KV and 13KV Transformers

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The licensee's examination activities, performed in response to the NRC Order "Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors" (February 11, 2003) and NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles" were inspected using Temporary Instruction, TI 2515/150, Revision 1. In addition, the inspector reviewed examination plans and test method commitments provided by the licensee in their response to NRC Bulletin 2002-02. The details of the inspection scope and results are in Section 4OA5 of this inspection report; as specified by the TI. The inspector reviewed the licensee's response dated February 18, 2003 to the Order.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed licensed operator simulator training in order to assess operator performance. Areas of focus included high risk operator actions, operators' activities associated the Emergency Plan, and previous lessons learned items. The inspectors also evaluated the clarity and formality of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift supervisor. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room, especially regarding recent control board modifications. The following simulator training scenario was reviewed:

- The inspectors observed a scenario on June 6, 2003, involving a loss of coolant accident. The inspectors evaluated the performance of risk significant operator actions including emergency operating procedure, EOP-5-1, "Loss of Coolant," and EOP-8-1, "Functional Recovery Procedures."

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (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 SSCs were reviewed:

- Unit 1 & Unit 2 Main Steam Systems: The systems were appropriately classified (a)(1) because of several repetitive non-risk functional failures of the main steam isolation valve (MSIV) to open on demand during surveillance testing identified during the previous 8 monitored quarters . Initial corrective actions were implemented to increase the PM periodicity; however, similar failures have occurred during 2Q/2003 which will result in additional monitoring under (a)(1).

Licensee investigations into the cause of the more recent failures is ongoing and the results of their investigation will be documented in Issue Report IR4-013-798 and AIT IR200300166. The licensee plans to continue monitoring the system in accordance with (a)(1) through October 2003.

- Unit 1 Containment Air Cooling System: The system was identified as an (a)(2) system. The system returned to (a)(2) September 2002 after approximately 8 months of monitoring under (a)(1). The licensee implemented appropriate corrective actions to improve containment air cooler reliability by replacing the affected components and increased monitoring for three surveillance periods prior to returning the system to (a)(2). The inspectors reviewed the licensee's corrective action plan documented in Issue Report IR3-080-025 and AIT IR200100938.

- Unit 1 Plant Compressed Air System: The 11 plant air compressor system was classified as (a)(1) due to repeat functional failures and system unavailability hours exceeding system level performance criteria during the previous 8 monitored quarters. The licensee documented these failures in Issue Report IR4-016-511 and Maintenance Order (MO) 1200301793. The licensee implemented corrective actions to improve plant air compressor reliability by replacing the affected components and monitoring the system to ensure the corrective actions have been effective. The licensee plans to continue monitoring the system in accordance with (a)(1) through December 2003.

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

- Station Procedure MN-1-112, "Managing System Performance"
- Maintenance Rule Scoping Document, Revision 20
- Maintenance Rule Indicator Report, May 2003

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.

- MO 0200301283, Repair Switchyard House Roof
- MO 2200203341, Remove Spare TCB and Install TCB-6
- MO1200302027, Unit 1 Waterbox Cleaning
- MO 2200302145, Panel 2T11 EHC Control

- MO 2200203306, Perform PM’s on 24 Inverter
- MO 2200302110, Perform Cleaning and Maintenance on 12B SRW HX

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events

a. Inspection Scope

The inspectors reviewed operator performance during plant startup, power ascension, rod drop, and reactor trip. (See Section 4OA3 for additional information). The inspectors examined operator logs, equipment response, sequence of events recorder logs, and alarm response procedures to determine if operators performed the appropriate actions in accordance with their training and procedures.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed three operability determinations and four operability evaluations 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 operability determinations and evaluations were reviewed:

- Operability Determination 02-013 Seismic Qualifications of Auxiliary Feedwater (AFW) Turbine Driven Pumps
- Operability Determination 02-015 4KV ABB/Westinghouse Breakers
- Operability Determination 03-003 4KV ABB/Westinghouse Breakers
- IR4-013-798/ AIT IR200300166, 12 MSIV Shuttle Valve
- IR3-080-025/ AIT IR200100938, U1 Containment Air Coolers
- IR4-018-598, 2-RC-403-MOV PORV Block Valve
- IR4-017-061, 11A RCP Vapor Seal Leakage

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated the cumulative effects of significant operator workarounds for potential effects on the functionality of mitigating systems. The workarounds were reviewed to determine: (1) if the functional capability of the system or human reliability in responding to an initiating event was affected; (2) the effect on the operator's ability to implement abnormal or emergency procedures; (3) if operator workaround problems were captured in the licensee's corrective action program.

b. Findings

No findings of significance were identified.

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) that equipment was returned to the status required to perform its safety function. The following maintenance orders were reviewed:

- MO 2200302038, CEA 27 Rod Drop.
- MO 21592250, 1SRW-1640-CV Stroke Time.
- MO 200200043, PORV Block Valve
- MO 2199904530, Pressurizer Safety Valve Setpoint Verification

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

During the Unit 2 refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities. The inspectors completed a containment closeout Inspection with an equipment operator and a senior reactor operator in the Unit 2 containment building.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and/or 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 or reviewed:

- STP-0-7B-2, B Train Engineered Safety Features Logic Testing
- STP-0-047B-2, MSIV Partial Stroke Testing
- STP-M-573-2, System Leakage
- STP-M-200-2, RTCB Testing
- STP-M-213, Calibration of Power Range Nuclear Instruments
- STP-O-33-2, Containment Atmosphere RMS Monthly Test
- STP-O-65H-2, Pressurizer PORV Block Valve Quarterly Operability Test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification TA-2-03-0019, which temporarily installed jumpers on 2MOV403 close indication on the PORV block valve. The inspectors assessed: (1) the adequacy of the 10 CFR 50.59 evaluation; (2) that the installation was consistent with the modification documentation; (3) that drawings and procedures were updated as applicable; and (4) the adequacy of the post-installation testing.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training on June 3, 2003, and during an emergency preparedness drill on June 24, 2003. The inspectors verified that emergency classification declarations and notification activities were properly completed.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

During the period from April 7-11, 2003, the inspector conducted the following activities and reviewed the following documents to determine the effectiveness of controls for radiologically significant work areas. The inspector obtained this information via interviews with licensee personnel; walk-down of systems, structures, and components; and examination of records, procedures, or other pertinent documents. The inspector's review focused on areas and activities with higher radiological risk.

The inspector made tours in radiological controlled areas to determine: adequacy of posting and barricading of High, Locked High and Very High Radiation Areas (as necessary); adequacy of radiation protection job coverage; adequacy of radiological surveys and general postings; and adequacy of air sampler location for representative air sampling. The inspector questioned workers and radiological controls personnel during the in-plant tours to ascertain knowledge and understanding of ambient radiological conditions and radiological controls and to verify conduct of adequate briefings.

The inspector reviewed radiological surveys (e.g., airborne radioactivity, loose and fixed surface contamination, and beta and gamma radiation dose rates) associated with the work, stay time calculations (as appropriate), conformance with applicable special radiation work permits (SWPs), use of engineering controls, internal and external exposure controls, discrete radioactive particle controls, use of operating experience, and completion of internal exposure assessments. Work activities reviewed included: conduct of under vessel head penetration inspections; Unit 2 steam generator replacement work activities; refueling; radiography of steam generator and piping welds; and demobilization of outage related equipment from the Unit 2 containment.

The inspector reviewed occupational exposures for workers: to determine if workers received unplanned internal or external exposures; to identify maximum occupational doses received; and to verify applicable radiological controls were adequate for conditions present. The inspectors also reviewed the work packages for the three highest exposure SWPs utilized in performing the reactor head inspection (SWP 2003-2312, 2003-2321, and 2003-2341).

The reviews in this area were against criteria contained in 10 CFR 19, 10 CFR 20, site Technical Specifications, and applicable radiation procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector selectively reviewed the adequacy and the effectiveness of the program to reduce occupational radiation exposure to as low as is reasonably achievable (ALARA). The following matters were reviewed:

- The inspector compared the original outage occupational exposure goal (209 person-rem) to the current accumulated outage exposure (239.26 person-rem as of April 7, 2003). The inspector discussed reasons for differences between the actual and estimated exposures with ALARA personnel including effective dose rates in the Unit 2 containment and emergent work encountered. The inspector reviewed performance in occupational exposure reduction for the outage. Tasks reviewed included: steam generator replacement; reactor vessel head inspection; radiography; and refueling.

- The inspector toured the Unit 2 containment to observe on-going work activities; to verify that workers were implementing acceptable ALARA practices (e.g., using low dose ALARA areas); and to determine if these activities were consistent with occupational reduction practices as specified in applicable ALARA reviews (considering work plans and remaining work). The inspector discussed the impact of early outage completion on accumulated doses.

- The inspector attended various outage and planning meetings. The inspector attended the integrated pre-job briefing on April 7, 2003, for Unit 2 containment access. The inspector also attended the Unit 2 outage status meetings on April 8-11, 2003.

The review was against criteria contained in 10 CFR 19, 10 CFR 20, site Technical Specifications, and applicable site procedures.

b. Findings

No findings of significance were identified.

20S3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity including: portable field survey instruments, friskers, portal monitors, and small article monitors. The inspector obtained this information via: interviews with licensee personnel; walk-down of systems, structures, and components; and, examination of records, procedures, or other pertinent documents. The inspector conducted a review of instruments observed, specifically verification of proper function and certification of appropriate source checks for these instruments, which were utilized to ensure that occupational exposures were maintained in accordance with 10 CFR 20.1201.

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 listed cornerstones to verify individual PI accuracy and completeness. This inspection examined data and plant records from 2000 through the first quarter of 2003, including review of PI Data Summary Reports, Licensee Event Reports, operator narrative logs, and the monthly and quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases. The information contained in these records was compared against the criteria contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, to verify that all conditions that met the NEI criteria were recognized, identified, and reported as a Performance Indicator.

- Unplanned Scrams per 7000 Critical Hours, Units 1 and 2.
- Scrams with a Loss of Normal Heat Removal, Units 1 and 2.
- Unplanned Power Changes per 7000 Critical Hours, Units 1 and 2.
- HPSI Unavailibility, Units 1 and 2.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

1. Selected Issue Follow-up

a. Inspection Scope

The inspectors selected eight issue reports for detailed review (IR4-015-716, IR4-015-656, IR4-016-601, IR4-019-956, IR4-015-716, IR4-014-741, IR4-015-832, IR4-014-815). The issue reports were associated with component mispositionings by Calvert Cliff's personnel. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed and appropriate corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of the Constellation Nuclear's corrective action program as delineated in QL-2, "Self Assessment/Corrective Action Program."

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

1. Pressurizer Enclosure Fire

a. Inspection Scope

On April 7, 2003, a small smoldering fire started in the Unit 2 pressurizer enclosure due to slag from welding that fell through fire protection blankets and ignited a piece of herculite underneath. The Calvert Cliff's fire brigade responded and extinguished the fire with water. Constellation Nuclear declared an unusual event because the fire was in a vital area and lasted for greater than 15 minutes. The inspectors toured the scene of the fire and discussed the event with Calvert Cliffs personnel. There was a human performance error to prevent fires from starting due to not properly securing fire blankets and by personnel movement in the area which caused the fire blankets to move away from the pressurizer enclosure. However, the inspectors determined that this was a minor issue because it did not have an actual impact on safety related systems, structures, or components, and it could not be reasonably viewed as a precursor to a significant event because of the location of the fire inside the pressurizer enclosure. The pressurizer enclosure is a reinforced concrete structure designed as a shield wall to block any passage of missiles to the containment, and it would have been unlikely for the smoldering fire to expand beyond the structure. Unit 2 was shutdown at the time of the fire with the vessel flooded up. Additionally, this issue is minor because it did not affect fire mitigation defense-in-depth elements such as detection and manual suppression capability; it did not affect automatic suppression capability; and it did not affect fire barriers. There was no damage noted to plant equipment, and no injuries to personnel. No violation of regulatory requirements occurred.

b. Findings

No findings of significance were identified.

2. Unit 2 Reactor Trip

a. Inspection Scope

The inspectors observed control room personnel response to a reactor trip on May 28, 2003. The inspectors arrived in the control room shortly after the reactor trip and observed operator action by the control room staff, including operator briefings, actions required by emergency and off-normal procedures and monitoring of plant conditions. As part of the follow-up to this event, the inspectors observed plant chart recorders, and sequence of event recorder logs. Additionally, the inspectors compared requirements of plant procedures to observation of operators' performance, and discussed the event with plant personnel. The following documents were reviewed and used as criteria for evaluating the operators' response to this event:

- EOP-0, "Post-Trip Immediate Actions"
- EOP-1, "Reactor Trip"

b. Findings

Introduction. A Green self-revealing finding was identified for a human performance error which resulted in a reactor trip during troubleshooting.

Description. On May 28, at 11:34 a.m., a self-revealing finding was identified when Calvert Cliffs Unit 2 tripped from 100 percent power during maintenance troubleshooting activities on a turbine generator governor valve control circuit. The cause of the trip was attributed to human performance error.

Instrument Maintenance Technicians were conducting troubleshooting on the Unit 2 Main Turbine Automatic Electro-Hydraulic (AEH) Controllers. A short circuit while troubleshooting induced a loss of voltage to the Automatic Electro-Hydraulic (AEH), and caused the governor valves to shut unexpectedly. This resulted in a rapid loss of load causing a high pressurizer pressure automatic reactor trip.

The licensee's investigation determined that the root cause of the unit trip was human performance error due to work practices and supervisory oversight. Maintenance technicians were moving energized test leads (connected to the circuit cards) back and forth between test equipment. The technicians felt that moving between test equipment was less risky than connecting and reconnecting to the test points. With this decision, they introduced a previously unknown failure mechanism with a Bayonet Neill Concelman (BNC) connector, which initiated the short circuit that resulted in the reactor trip. Even though pertinent information to the trip risk was known, the troubleshooting team members assumed each other were aware of the situation, never discussed this critical information, and took unnecessary risks in the final stages of the troubleshooting. When the troubleshooting team expanded the scope of the work activity to test additional points in the "manual" portion of the circuit, they significantly increased plant trip risk. Contrary to Calvert Cliffs procedure MN-1-110, "Conduct of Maintenance", the team did not obtain the appropriate review and approval by plant management for changes to an approved troubleshooting plan.

Constellation Nuclear corrective actions (IR200300217) include awareness training on the event, revisions to maintenance training programs on appropriate work practices when using test equipment, and the reinforcement of management's expectations in the supervisory oversight role.

Analysis. The deficiency associated with this event was human performance error due inadequate work practices and supervisory oversight. This self-revealing finding is greater than minor because it affected an attribute and the objective of the Initiating Events Cornerstone in that the work practices inadequacies resulted in a perturbation in plant stability by causing a reactor trip. The finding was assessed using the Phase 1 Significance Determination Process for Reactor Safety Inspection Findings and was determined to be of very low safety significance (Green). The finding was of very low safety significance because, while the finding resulted in an actual reactor trip, the finding did not contribute to the likelihood of a primary of secondary system LOCA initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. This finding is in CCNPPI's corrective action program as IR 200300217.

Enforcement. No violation of regulatory requirements occurred. (FIN 05000318/2003003-01)

3. (Closed) LER 50-317/2001-002-00, Appendix R Steam Generator Dry-Out Calculation Omitted Blowdown Flow

On September 5, 2001, it was determined that steam generator blowdown flow was not credited in the calculation developed to determine the time to steam generator dry-out during an Appendix R event. The event was identified in NRC Fire Protection Inspection Report No. 50-317/01-007 and 50-318/01-007, and a non-cited violation was issued against 10 CFR 50, Appendix B, Criteria III, "Design Control." The operability determination, 01-016, Inaccuracies in the loss of main feedwater accident analysis, regarding this event, was reviewed in NRC Inspection Report 50-317/01-11, 50-318/01-11, with no findings of significance identified. The inspector verified with engineering personnel that the corrective actions had been appropriately completed. The final corrective action was recently installed during the Unit 2 refueling outage; it added a blowdown isolation during an AFAS signal. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the issue in Issue Reports (IR)3-075-582 and 583. This LER is closed.

4. (Closed) LER 50-317, 318/2002-002-00, Potential High Pressure Safety Injection Pump Run-out Failure

On February 21, 2002, plant personnel determined that a potential high pressure safety injection pump run-out failure mode existed if a large break loss-of-coolant-accident occurred during reverse flow testing of pump discharge check valves. The test resulted in a plant configuration where the non-tested pump was aligned to both headers, and therefore its flowrate would have increased and probably caused the pump to fail due to run-out during a loss-of-coolant-accident. Corrective actions included revising the test procedures to include a step to shut the header isolation valve for the pump being tested, while its discharge check valve is being reverse flow tested. The inspector verified that the test procedures for both units have been appropriately revised, including a precaution step explaining the changes. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the issue in IR3-076-959. This LER is closed.

5. <u>(Closed) LER 50-317/2002-003-00</u>, Reactor Trip Due top Loss of Reactor Coolant Pump Motor Oil

On July 24, 2002, operators manually tripped Unit 1 due to a high reactor coolant pump thrust bearing temperature. The high bearing temperature was caused by the loss of oil to the bearing due to failure of a butt weld on a motor oil cooler line. Corrective actions included repair of the failed weld, and inspection and repair of all other similar pump welds. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the issue in IR3-061-964. This LER is closed.

6. <u>(Closed) LER 50-317, 318/2002-004-00</u>, Post-Accident Monitoring Instrumentation Not Seismically Connected

On August 27, 2002, plant personnel identified that a condition had existed that could have prevented the post-accident monitoring system from fulfilling its safety function. A review of a causal analysis identified that loose pins on the cable connectors for the containment area radiation high range indicators could cause the channels to fail during a seismic event. The cables had become loose in their connectors due to approximately 20 years of unmonitored mechanical wear from repeated disengagement/engagement operations. Additionally, the proper extraction tool, which is used to remove the connector pins, was not always used, increasing the wear. Corrective actions included repair of all instrumentation with this style connector and four sets of extraction tools were made available for use. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the issue in IR3-081-622, and IR4-009-082. This LER is closed.

4OA4 <u>Steam Generator Replacement</u>

1a. <u>Inspection Scope</u>

The inspector reviewed radiological controls for on-going steam generator replacement work activities. The following matters were reviewed:

- Current occupational exposure performance relative to goals and dose tracking;
- Ongoing radiological controls for work activities including: temporary shielding, contamination controls, adherence to radiation protection procedures including radiation work permits, and radioactive material management;
- Interim storage facility for the steam generators, including shielding design and exterior dose rates and their impact on the environment.

The review was against criteria contained in 10 CFR 19, 10 CFR 20, site Technical Specifications, and applicable site and project procedures.

b. <u>Findings</u>

No findings of significance were identified.

2.a. <u>Inspection Scope</u>

The inspectors reviewed S/G post installation verification and testing inspections. The following attributes were reviewed:

- The licensee's post-installation inspections and verifications program and its implementation.
- The conduct of RCS leakage and secondary leakage testing and the test results.
- Calibration and testing of instrumentation affected by S/G replacement.

The review was against criteria contained in site Technical Specifications, and applicable site and project procedures.

b. <u>Findings</u>

No findings of significance were identified.

40A5 <u>TI 2515/150, Revision 1 - Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Bulletin 2002-02)</u>

a. <u>Inspection Scope</u>

The inspector reviewed the licensee's inspection activities to detect evidence of leakage and/or cracking of RPV head penetration (control element drive mechanism, in core instrumentation and the vessel head vent) nozzles in response to NRC Bulletin 2002-02 as required by temporary instruction TI 2515/150, Revision 1. The licensee performed a visual examination to evaluate the integrity of the vessel head and penetration intersections to confirm the absence of flaws and boric acid deposits.

The inspection included interviews with examination personnel, data analysts and metallurgical engineering personnel to assess their knowledge of these activities. The inspector reviewed the analysts' training and qualification records to verify that the personnel qualification process adequately prepared the assigned staff to perform the examination and analyze the ultrasonic data. Also, the inspector reviewed the examination procedures to determine whether they provided adequate guidance and examination criteria to implement the examination plan.

The inspector selected six control element drive mechanism (CEDM) nozzles and one in-core instrumentation penetration to observe and evaluate the effectiveness of the visual (VT) and ultrasonic test (UT) to verify that the test methods could reliably detect an existing "leak path" or actual leakage from a failure of the vessel head penetration. The inspector verified by observation that the reactor vessel head was free of dirt, debris, boron deposits, insulation, significant oxidation and any material that could adversely affect viewing of all penetrations (360 degrees around the circumference of the nozzle) and the vessel head in its entirety. The inspector verified that the procedures used required that anomalies, deficiencies and discrepancies identified during the examination process be evaluated and the results documented.

Ultrasonic test (UT) results used to determine if leakage had occurred into the interference fit zone were reviewed. This included observation of UT data presentations for a sample of the Calvert Cliffs Unit 2 CEDMs, and examples, for comparison, of UT results from plants that had leakage into the interference fit zone.

b. Findings

No findings of significance were identified.

The specific reporting requirements of TI 2515/150, Revision 1 are documented in Supplemental Information section of this inspection report.

4OA6 Meetings, including Exit

On April 3, a group of technical reviewers from NRR including Mr. A. Hiser visited the Calvert Cliffs site, toured containment, reviewed RPV head inspection activities and met with CCNPPI technical personnel and senior management. The inspection results of the reactor pressure vessel head and penetration inspection were presented to Mr. Bill Holston, Manager Engineering Services, and other members of his staff at the conclusion of the inspection on April 7, 2003.

The inspector presented the inspection findings regarding the steam generator replacement radiological controls inspection to licensee representatives on April 11, 2003.

On July 9, 2003, the resident inspectors presented the inspection results to Mr. Kevin Nietmann and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during any of the quarterly inspection activities.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Vanderheyden, Vice President
K. Neitmann, Plant General Manager
L. Weckbaugh, Manager, Nuclear Support Services
M. Geckle, Manager, Nuclear Operations
D. Holm, Manager, Nuclear Maintenance
B. Holston, Manager, Engineering Services
P. Furio, Acting Director, Nuclear Regulatory Matters
G. Gwiazdowski, Director, Nuclear Security/Emergency Planning
R. Szoch, General Supervisor, Plant Engineering
T. Kirkham, Radiation Protection Supervisor
R. Lopez, ALARA Specialist
S. Sanders, General Supervisor-Radiation Safety
P. Serra, ALARA Specialist
R. Wyvill, ALARA Supervisor
J. York, Radiation Protection Supervisor

NRC personnel

A. Hizer, NRR
Z. Bart Fu, NRR
J. Collins, NRR
E. Reichelt, NRR
N. Sanfilippo, NRR
R. Davis, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Opened and Closed

05000318/2003003-01	FIN	Troubleshooting human performance error results in reactor trip (4OA3.2)

Closed

50-317/2001-002-00	LER	Appendix R Steam Generator Dry-Out Calculation Omitted Flowdown Flow (Section 40A3)
50-317;50-318/2002-002-00	LER	Potential High Pressure Safety Injection Pump Run-Out Failure (Section 4OA3)
50-317/2002-003-00	LER	Reactor Trip Due To Loss of Reactor Coolant Pump Motor Oil (Section 40A3)
50-317; 50-318/2002-004-00	LER	Post-Accident Monitoring Instrumentation Not Seismically Connected (Section 4OA3)

Discussed

NONE

REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATION NOZZLES (NRC BULLETIN 2002-02) TI 2515/150, Revision 1 -

Penetration Nozzles Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel with certification to the American Society of Mechanical Engineers (ASME), Section XI, Level II and Level III for visual examiners. In addition, Level II and Level III examiners had received training in this type of inspection. The training included a review of industry experiences, lessons learned, inspection results and procedure requirements.

Ultrasonic test personnel performing calibration or data analysis functions were qualified to a minimum of Level II in ultrasonic examination. In addition, data analysis personnel had documented training in the analysis system and had received documented training on reactor head penetration (RHP) examination techniques and a documented period of RHP analysis experience.

a.2. The examination was performed using adequate procedures. The procedures had been demonstrated on a mock up of a vessel head penetration. The procedures specified the extent of the inspection required, provided detailed documentation requirements and provided clear inspection standards and acceptance criteria on which personnel were trained. The examination procedure was approved by the licensee's Level III ultrasonic test examiner. The 90% applicable to the extent of examination around the circumference, as being equal to 100% was not needed as the UT of each CEDM was done 100% around the circumference.

The 2" extent above the weld was generally not achievable with about 1.5" inch being more typical. HQ-NRR is involved in providing "relief " for this condition.

The PT procedure is listed in the Documents Reviewed. The video tape of the original PT and extent of grinding and post grinding developer application was viewed by HQ staff and the regional inspector. The final PT was, as noted above, dispositioned as no relevant indications. There was clearly no deep discontinuity present in the head vent weld area that was PT examined.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies. The inspectors noted that the UT results of the examination to evaluate if leakage had occurred thru the interference fit zone would be strongly affected by the presence of residual water in that area. The Calvert Cliffs interference fit zones and the gap area between them and the CEDM welds appeared to not contain water. The UT results of the interference fit region were consistent with the as-built geometric conditions above the CEDM to head welds and did not indicate that leakage had occurred.

a.4. The examination performed was capable of identifying the primary water stress corrosion phenomena described in the bulletin.

a. The reactor vessel head was free of dirt, debris, insulation, significant oxidation and any material that could adversely affect viewing of the penetrations (360 degrees around the circumference of the nozzle) and the vessel head in its entirety. Due to dose considerations, all nozzle penetrations, including the vent line, were remotely inspected for a full 360 degree view using a high resolution camera delivered by a robotic crawler.

b. Small boron deposits as described in Bulletin 2001-01 could be identified and characterized by the visual technique used. No boron deposits were identified at the penetrations or on adjacent areas of the vessel head.

c. No material deficiencies were identified. PT did not identify any relevant indications. The PT examination identified several small indications that were ground and PT inspected again. Most of these cleared on the re-PT and the remaining small indications were dispositioned as nonrelevant by the Level III PT examiner. The PT process including related grinding was a high radiation exposure process, with 1.6R on the first cycle.

d. The ALARA radiation exposure controls were effective in minimizing personnel exposure during the insulation removal and visual examination of the penetrations and the vessel head. The visual examination was accomplished from the top of the vessel head flange. Also, a significant effort was made to reduce exposure to personnel involved in activities in close proximity to the vessel head. The CEDM cooling shroud and the blanket insulation were both easily removed from the location to be inspected. There were no significant items which would impede an effective examination of the outside surface of the head and penetrations in the future. There was no significant challenge to radiation exposure controls.

e. The guide sleeves that are inside most of the CEDMs and CEDM weld distortion result in a narrow gap in some areas between the sleeve OD and the CEDM inner diameter (ID). This required special tooling to move the sleeves to provide for access space for full circumferential UT examination of many of the CEDMs. The basis for the temperature used (593.7° F) in the susceptibility ranking calculation was an analysis documented in Combustion Engineering report CE NPSD-1074, CEOG task 953 (Evaluation of Reduction in Fluid Temperature in the Reactor Vessel Upper Plenum Due to Increased Bypass Flow, dated February 1997). The temperature in the upper head region was determined using a model with analysis performed that is typical for Calvert Cliffs, Unit 2 (2700 MWt). There have been no changes in plant operation to date that have resulted in a change in the original maximum design temperatures in the upper head region.

f. The basis for the temperature used (593.7 degrees F) n the susceptibility ranking calculation was an analysis documented in Combustion Engineering report CE NPSD-1074, CEOG task 953 (Evaluation of Reduction in Fluid Temperature in the Reactor Vessel Upper Plenum Due to Increased Bypass Flow, dated February 1997). The temperature in the upper head region was determined using a model with analysis performed that is typical for Calvert Cliffs, Unit 2 (2700 Mwt). There have been no changes in plant operation to date that have resulted in a change in the original maximum design temperatures in the upper head region.

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection

Procedures

OI-27C, “4.16KV,System
ERPIP 3-0, Attachment 20, Severe Weather
OI 27B, 13.8 1KV System
OI 22C, ECCS Pump Room Ventilation
OI 32 A-1, Auxiliary Feedwater
OI 15-1, Service Water System

Section 1RO4: Equipment Alignment

Procedures

OI-3, Unit 2 High Pressure Safety Injection
OI-32A-2, Unit 2 - Auxiliary Feedwater System
OI-32A-1, Unit 1 - Auxiliary Feedwater System
OI-16-1, Component Cooling
OI-21A-1, Unit 1A Emergency Diesel

Section 1RO5: Fire Protection

Procedures

SA-1-100, Fire Prevention

Section 1R11: Licensed Operator Requalification Program

Procedures

EOP-5-1, Loss of Coolant
EOP-8-1, Functional Recovery Procedures

Section 1R12: Maintenance Effectiveness

Issue Report**s**

IR4-013-798 / AIT IR200300166
IR3-080-025 / AIT IR200100938
IR4-016-511

Maintenance Orders

MO 1200301793

Procedures

Station Procedure MN-1-112, Managing System Performance
Maintenance Rule Scoping Document, Revision 20
Maintenance Rule Indicator Report, May 2003

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Maintenance Orders

MO 0200301283, Repair Switchyard House Roof
MO 2200203341, Remove Spare TCB and Install TCB-6
MO 1200302027, Unit 1 Waterbox Cleaning
MO 2200302145, 2 Panel 2T11 EHC Control
MO 2200203306, Perform PM's on 24 Inverter
MO 2200302110, Perform Cleaning and Maintenance on 12B SRW HX

Section 1R15: Operability Evaluations

Operability Orders

Operability Determination 02-013 Seismic Qualifications of AFW Turbine Driven Pumps
Operability Determination 02-015 4KV ABB/Westinghouse Breakers
Operability Determination 03-003 4KV ABB/Westinghouse Breakers

Issue Reports

IR4-013-798/ AIT IR200300166, 12 MSIV Shuttle Valve
IR3-080-025/ AIT IR200100938, U1 Containment Air Coolers
IR4-018-598, 2-RC-403-MOV PORV Block Valve
IR4-017-061, 11A RCP Vapor Seal Leakage

Section 1R19: Post-Maintenance Testing

Maintenance Orders

MO 2200302038, CEA 27 Rod Drop.
MO 21592250, 1SRW-1640-CV Stroke Time.
MO 200200043, PORV Block Valve
MO 2199904530, Pressurizer Safety Valve Setpoint Verification

Section 1R22: Surveillance Testing

STP-0-7B-2 B Train Engineered Safety Features Logic Testing
STP-0-047B-2 MSIV Partial Stroke Testing
STP-M-573-2 System leakage
STP-M-200-2 RTCB Testing
STP-M-213 Calibration of Power Range Nuclear Instruments
STP-O-33-2 Containment Atmosphere RMS Monthly Test
STP-O-65H-2 Pressurizer PORV Block Valve Quarterly Operability Test

Section 1R23: Temporary Plant Modifications

Temporary Modifications

TA-2-03-0019

Section 2OS1: Access Control to Radiologically Significant Areas

Special Radiation Work Permits

2003-2312
2003-2321
2003-2341

Section 40A2: Problem Identification and Resolution

Issue Reports

IR4-015-716
IR4-015-656
IR4-016-601
IR4-019-956
IR4-015-716
IR4-014-741
IR4-015-832
IR4-014-815

Section 4OA3: Event Follow-up

Procedures

EOP-0, Post-Trip Immediate Actions
EOP-1, Reactor Trip”
MN-1-110, Conduct of Maintenance

Issue Reports

IR 200300217

Section 4A05: TI 2515/150, Revision 1 - Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Bulletin 2002-02)

Orders/NRC Bulletins

Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors (February 11, 2003)
NRC Bulletin 2002-02, Reactor Pressure Vessel Head And Vessel Head Penetration Nozzle Inspection Programs
Calvert Cliffs Nuclear Power Plant 30-Day Response to NRC Bulletin 2002-02
NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity

Reports

Summary Report for Calvert Cliffs Unit 2 RVH CRDM Penetration Visual Inspection

Procedures

54-ISI-100-09 Remote Ultrasonic Examination of Reactor Head Penetrations
54-ISI-367-03 Procedure for The Visual Exam for Leakage of Reactor Head Penetrations
54-ISI-21-31 Written Practice for Personnel Qualification Ultrasonic Method
54-ISI-137-01 Remote Ultrasonic Examination of Reactor Vessel Head Vent Line Penetrations
54-ISI-244-07 Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface
54-ISI-250-00 Liquid Penetrant Examination of Reactor Vessel Head Penetration J-Groove Welds
54-PT-6-07 Visible Solvent Removable Liquid Penetrant Examination Procedure
NDE-5200-CC Color Contrast Liquid Penetrant Examination

Root Cause Analysis

RCAR 94-09Root Cause Corrosion of Carbon Steel Studs and Nuts on ICI Flanges
93-32-0041 Failure Analysis of Leak on #13 Reactor Vessel Level Monitoring System Upper Omega Seal

Drawings

DWG 6022107D CCU2 Bare Head Inspection
DWG 233-415 Closure Head Assembly
DWG 12017-0026 Closure Head Nozzle Details
DWG 12017-78 Nozzle Requirements Closure Head

LIST OF ACRONYMS

AEH	Automatic Electro-Hydraulic
ADAMS	Agency wide Documents Access and Management System

ALARA	As Low As Reasonably Achievable
AFW	Auxiliary Feedwater
BNC	Bayonet Neill Concelman
CCNPPI	Calvert Cliffs Nuclear Power Plant, Inc.
CFR	Code of Federal Regulations
CEA	Control Element Assembly
CEDM	Control element drive mechanism
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Plan
IMC	Inspection Manual Chapter
LOCA	Loss of Coolant Accident
MO	Maintenance Order
MSIV	Main Steam Isolation Valve
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
RCA	Radiological Controlled Area
PI	Performance Indicator
PORV	Power Operated Relief Valve
RPV	Reactor Pressure Vessel
RTCB	Reactor Trip Circuit Breaker
SDP	Significance Determination Process
S/G	Steam Generator
SRW	Service Water
SW	Salt Water
SWP	Special Work Permit
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VPL	Valve Position Limiter