July 27, 2004

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000317/2004005 AND 05000318/2004005

Dear Mr. Vanderheyden:

On June 30, 2004, the U.S. 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 2, 2004 with 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.

This report documents three NRC-identified findings and one self-revealing finding of very low safety significance (Green) all of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program (CAP), the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, a licensee-identified violation, which was determined to be of very low safety significance is listed in this report. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN. Document Control Desk, Washington, D.C. 20555-0001 with copies to the Regional Administrator, Region 1; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Calvert Cliffs Facility.

In accordance with 10 CFR 2.390 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 05000317/2004005 and 05000318/2004005
w/Attachment: Supplemental Information

cc w/encl:
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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.:	05000317/2004005 and 05000318/2004005
Licensee:	Calvert Cliffs Nuclear Power Plant, Inc. (CCNPPI)
Facility:	Calvert Cliffs Nuclear Power Plant
Location:	1650 Calvert Cliffs Parkway Lusby, MD 20657-4702
Dates:	April 1, 2004 - June 30, 2004
Inspectors:	Mark A. Giles, Senior Resident Inspector Joseph M. O'Hara II, Resident Inspector John R. McFadden, Health Physicist Barry Norris, Senior Reactor Inspector Neil Perry, Senior Project Engineer Timothy O'Hara, Reactor Inspector Stephen M. Pindale, Senior Reactor Inspector Thomas F. Burns, Reactor Inspector Brice A. Bickett, Reactor Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000317/2004005, 05000318/2004005; 4/1/2004 - 6/30/2004; Calvert Cliffs Nuclear Plant, Units 1 and 2; Maintenance Risk Assessments and Emergent Work Control, Operability Evaluations, Identification and Resolution of Problems.

The report covered a three-month period of inspection by resident inspectors and announced inspections performed by a senior project engineer, two senior reactor inspectors, three reactor inspectors, and a health physicist. The inspection identified four Green findings, which were determined to be non-cited violations. 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 non-cited violation of 10 CFR 50.65(a)(4) which requires that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Specifically, the licensee failed to identify and therefore assess and manage the risks associated with performing maintenance on the 'B' channel of the reactor protection system (RPS) while operating in a reduced inventory condition. This maintenance activity resulted in the loss of one of the two shutdown cooling (SDC) operating trains for about 18 minutes with a corresponding heatup of the reactor coolant system (RCS) of 2 degrees Fahrenheit (°F).

  This finding is greater than minor because it affected an attribute and objective of the Initiating Event Cornerstone in that human performance inadequacies resulted in an event that upset plant stability during shutdown operations. This issue was evaluated in accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, Shutdown Operations SDP, and was determined to be of very low safety significance. The inspectors identified that a contributing cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R13)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, associated with a self-revealing finding, which requires that measures be established to assure that design basis are correctly translated into procedures. Specifically, the licensee failed to incorporate a design flow calculation into an operating procedure which allowed the licensee to operate the 12 component cooling water heat exchanger (CC HX) in excess of its maximum shell side flow versus time curves. This failure resulted in tube failures in the only available, and in-service CC HX which supported SDC operations of the RCS.

  This finding is greater than minor because it affected an attribute and the objective of the Mitigating System Cornerstone in that inadequate procedure quality resulted in degraded availability, reliability and capability of a system that responds to initiating events to prevent undesirable consequences.   In accordance with Inspection Manual Chapter (IMC) 0609, Appendix G, Shutdown Operations SDP, this finding was determined to be of very low safety significance (Green) since the safety function of the component cooling water system was not lost. (Section 1R15)

- Green. The inspector identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI. The licensee failed to promptly correct a testing deficiency identified during a CX relay failure in 1998. When action was taken in October of 2001, it was not sufficient to prevent further CX relay failures in December 2003 and February 2004.

  This finding is more than minor because it affects the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affects the availability and reliability of the 480 volt (v) electrical distribution system. The finding is of very low safety significance because the finding did not represent an actual loss of safety function and did not screen as potentially risk significant due to a seismic, fire, flooding or severe weather initiating event. The inspectors identified that a contributing cause of this finding was related to the cross-cutting area of Problem Identification and Resolution. (Section 4OA2)

- Green. The inspector identified a non-cited violation for failure to implement procedures to control maintenance activities required by Technical Specification 5.4.1.a. and Regulatory Guide 1.33. The licensee failed to implement procedures to ensure that planned, scheduled maintenance was actually being performed. Maintenance personnel, by procedure, are permitted to decide whether or not to clean and lubricate 480v breakers. If the maintenance personnel decide not to perform the scheduled clean and lubricate, no method is specified or available to report this situation to maintenance and engineering management.

This finding is more than minor because it affects the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affects the availability and reliability of the 480v electrical distribution system. The finding is of very low safety significance because the finding did not represent an actual loss of safety function and did not screen as potentially risk significant due to a seismic, fire, flooding or severe weather initiating event. (Section 4OA2)

B. Licensee-Identified Violations

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. This violation is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent reactor power and remained unchanged until April 8, when power was reduced to 93 percent for a scheduled auxiliary feedwater (AFW) flow surveillance test. On April 9, reactor power was again reduced from 93 percent reactor power to 0 percent reactor power for a scheduled refueling outage (RFO). Following completion of the RFO on May 9, the unit was returned to 100 percent reactor power. On June 17, reactor power was reduced to 94 percent due to the indication of elevated hydrogen gas temperatures associated with the main turbine generator. Reactor power was restored to 100 percent following repairs on June 18, and remained there for the rest of the inspection period.

Unit 2 began the inspection period at 100 percent reactor power and remained there until May 22, when reactor power was reduced to 85 percent due to the loss of the 22 circulating water pump. Repairs were completed and the unit was returned to 100 percent reactor power the same day. On June 4, reactor power was reduced to 85 percent for condenser waterbox cleaning and the performance of other associated maintenance. On June 6, power was increased to 100 percent, where it remained until June 12, when reactor power was reduced to 85 percent to perform routine turbine control valve testing. Following the testing, power was increased to 100 percent where it remained for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.04 - 2 samples)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating strategies in preparation for potential severe tornado events. This review included an assessment of station procedures Emergency Response Plan Implementation Procedure (ERPIP) 3.0, Immediate Actions, Attachment 20, Severe Weather, ERPIP 3.0, Immediate Actions Attachment 21, Personnel Recall for Severe Weather, and Operations Administrative Policy OAP 00-01, Severe Weather Operations. Two risk significant systems were selected for this inspection, the 1A emergency diesel generator, and the Unit 1 auxiliary feedwater system (AFW). The inspectors conducted discussions with control room operators and systems engineers to understand protective measures applicable to these systems, and also performed partial field walkdowns of these systems to verify correct system alignment prior to potential tornado events.

The inspectors also reviewed the licensee's response to a tornado watch event that occurred on May 31, 2004. Specifically, the inspectors reviewed ERPIP 3.0, Immediate Actions, Attachment 20, Severe Weather, ERPIP 3.0, Immediate Actions Attachment 21, Personnel Recall for Severe Weather, and Operations Administrative Policy (OAP) 00-01, Severe Weather Operations, as well as conducted discussions with licensee personnel following the severe thunderstorm.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 4 samples)

a. Inspection Scope

The inspectors verified that select equipment trains of safety-related and risk significant systems were properly aligned. The inspectors reviewed plant documents to determine the correct system and power alignments, as well as the required positions of critical valves and breakers. The inspectors verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or potentially impact the availability of associated mitigating systems. The applicable documents for this inspection are located in the Attachment. The inspectors performed the following partial system walkdowns:

- 11 Charging Pump
- Unit 1 11 and 12 CC Water HXs
- Unit 1 12 Service Water Header
- Unit 1 12B Service Water HX

b. Findings

No findings of significance were identified.

1R05 Fire Protection

1. Fire Brigade Annual Observation (71111.05A - 1 sample)

a. Inspection Scope

The inspectors observed a fire brigade drill which was conducted on June 1, 2004, involving a simulated fire in a flammable storage cabinet located in the station blackout OC Diesel Generator Building. The inspectors observed the brigade members donning protective equipment, transitioning to the scene of the simulated fire, and fighting the simulated fire. The inspectors observed the fire brigade leader performing an assessment of the fire, evaluating the need for off-site assistance, communicating with team members and the control room supervisor, and directing the actions of the brigade to extinguish the fire. The inspectors attended the post drill debriefing conducted between the assessment team and the fire brigade members. Constellation procedure SA-1-101, Fire Fighting, and the Fire Fighting Strategies Manual were referenced for this inspection activity. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified.

2. Fire Area Walkdowns (71111.05Q - 7 samples)

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The inspectors assessed the material condition of fire protection suppression and detection equipment to determine whether any conditions or deficiencies existed which could impair the operability of that equipment. The inspectors reviewed administrative procedure SA-1-100, Fire Prevention, during the conduct of this inspection. The applicable documents for this inspection are located in the Attachment. The inspectors toured the following areas important to reactor safety:

- 0C Emergency Diesel Generator Room
- Unit 1 Turbine Driven AFW Room
- Unit 2 Turbine Driven AFW Room
- Unit 1 Service Water (SRW) HX/Motor Driven AFW Pump Room
- Unit 2 SRW HX/Motor Driven AFW Pump Room
- Unit 1 CC Water HX Room
- Unit 2 Emergency Core Cooling System Pump Room

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08 - 6 Samples)

a. Inspection Scope

The purpose of this inspection was to assess the effectiveness of the licensee's ISI program for monitoring degradation of the RCS boundary, risk significant piping system boundaries, and the containment boundary. The inspectors assessed the inservice inspection activities using the criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI.

The inspectors reviewed documents and observed portions of selected non-destructive examination (NDE) activities, and included samples of NDE activities associated with the repair and replacement of components. The sample selection was based on the inspection procedure objectives and risk significance. Specifically, the inspectors focused on components and systems where degradation would result in a significant increase in risk of core damage. These reviews were conducted to verify the activities were performed in accordance with ASME Code requirements. The inspectors reviewed a sample of examination reports and issue reports initiated during ISI examinations to evaluate the licensee's effectiveness in the identification and resolution of problems.

The inspectors reviewed portions of the steam generator management plan and the final operational assessment to assess the steam generator inspection and management program. The inspectors reviewed plant specific steam generator information, tube inspection criteria, integrity assessments, degradation modes and tube plugging criteria. The licensee conducted eddy current testing (ECT) of all tubes in steam generators #11 and #12 to identify and quantify tube degradation mechanisms and to confirm tube integrity following the first cycle of operation. The inspectors observed a sample of tubes being examined from each generator to verify the licensee examined the entire length. The inspectors interviewed data management and data acquisition personnel and resolution analysts. Also, the inspectors reviewed examination data for selected tubes from both steam generators, and evaluated the characterization and disposition of the identified flaws to assess the implementation of the steam generator inspection program. No tubes were identified as candidates for in-situ pressure testing during the inspection.

The inspectors observed portions of four in-process NDE activities and reviewed documentation and examination reports for an additional five NDE activities. The activities included volumetric and surface examinations. The inspectors reviewed two samples involving pressure boundary welding activities, and reviewed two samples involving component replacements performed during previous operating cycles (2000 and 2002). The inspectors reviewed one sample of an ultrasonic thickness measurement of the containment liner performed during the previous Unit 2 operating cycle which resulted in the acceptance of a condition for continued operation without repair (issue report (IR) 4-000-255 and engineering evaluation ES199900382).

The inspectors observed portions of the following specific examination activities:

- Magnetic particle (MT) examination of integral attachments to 16-FW-1218-R-3;
- Ultrasonic examination (UT) calibration for nozzle-to-pressurizer bottom head weld C69-PZR;
- UT examination of reactor vessel head penetrations # 2, 4, 9, 16, 29, 35 and 45;
- Visual examination (VT) of the reactor pressure vessel (RPV) head bare metal including vessel head penetrations #2, 4, 9, 16, 29, 35 and 45 and their intersection with the vessel head;
- ECT examination of steam generator #11 tubes 95-75, 104-82, 117-79 and steam generator #12 tubes 89-91, 105-89 and 111-89.

In addition, the inspectors reviewed the radiographs and interpretation of test results of field welds 1, 2, 3, 2R1 and 3R1 for installation of check valve 1-CKVCVC-186 in the chemical volume control system.

The inspectors reviewed welding activities associated with the repair and replacement of a pressure retaining component to verify these activities were performed in accordance with the requirements of ASME Section IX and XI. The inspectors reviewed completed work order MO 1200002340 (Installation of Valve 1-CVC-186) in the chemical volume control system. The inspectors reviewed the identification, removal, and repair by welding of rejectable indications detected in field welds 2 and 3 during the installation of valve 1-CKVCVC-186. The inspectors reviewed welding procedures P8-T and P8-T/LH and the supporting procedure qualification records 1, 2, 3A, 4, 11, 12, 15, 16, 48, 72 and 107 for compliance with the requirements of ASME Section IX. Also, the inspectors verified that welders completing welds 1, 2, 3, 2R1 and 3R1 were qualified for this welding in accordance with the requirements of ASME Section IX.

The inspectors reviewed the VT results of the drywell liner to verify compliance with the requirements of ASME Section XI, IWE (requirements for Class MC and Metallic Liners of Class CC components).

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

a. Inspection Scope

The inspectors observed a licensed operator simulator training scenario conducted on June 18, 2004, in order to assess operator performance as well as the adequacy of operator requalification training. The scenario involved failures of the 11A 480 volt ac bus, an upper seal failure in the 12A reactor coolant pump (RCP), which eventually resulted in the subsequent tripping of the pump and reactor, failure of the 11 HPSI pump, and a rupture of the 12 LPSI pump suction piping. During this inspection, the inspectors focused on high-risk operator actions performed during implementation of the emergency operating procedures, emergency plan implementation, and classification of the event. 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 operations and manipulations, 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 applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples)

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scoping, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components (SSC) scoped in the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. Documents applicable to this inspection are listed in the Attachment. The inspectors conducted this inspection for the following equipment issues:

- Unit 1 11 AFW Forced Oil System Modification Boring Error
- Unit 1 12 CC Water HX Tube Failures

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 5 samples)

a. Inspection Scope

The inspectors reviewed the licensee's assessments concerning the risk impact of removing from service those components associated with the work items listed below. This review primarily focused on activities determined to be risk significant within the maintenance rule. The inspectors compared the risk assessments and risk management actions performed by station procedure NO-1-117, Integrated Risk Management, to the requirements of 10 CFR 50.65(a)(4), the recommendations of NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Section 11, "Evaluation of Systems to Be Removed From Service," and approved station procedures. The inspectors compared the assessed risk configuration to actual plant conditions to evaluate whether the assessment was accurate and comprehensive. In addition, the inspectors assessed the adequacy of the licensee's identification and resolution of problems associated with maintenance risk assessments and emergent work activities. The applicable documents for this inspection are located in the Attachment. The inspectors reviewed the following selected work activities:

- Unit 1 'B' RPS Channel Power Supply Replacement
- 22 Salt Water Header Discharge Check Valve Replacement
- 1A Emergency Diesel Generator Maintenance
- 12 Turbine Driven AFW Pump Maintenance

- 12B SRW Heat Exchanger/Header Maintenance

b. Findings

Introduction. The inspectors identified an NCV of very low safety significance (Green) for the licensee's failure to perform an adequate risk assessment as required by 10 CFR 50.65(a)(4) for maintenance activities. Specifically, the licensee failed to identify that maintenance activities being performed on the Unit 1 reactor protection system (RPS), which induced a ground during the performance of those activities, could have had an adverse impact on SDC during reduced inventory operations. This maintenance could have been postpone until the plant was place in a preferable operational state for performing this activity.

Description. On April 13, 2004, at 11:30 p.m., the licensee commenced a draindown of the Unit 1 RCS to 37.4' (mid-loop, reduced inventory operation) to support the installation of nozzle dams and perform RCP seal maintenance. Concurrent with this evolution, pre-approved maintenance was being performed on the Unit 1 'B' RPS channel which consisted of the replacement of redundant power supplies. At 11:50 p.m. following replacement of the power supplies, maintenance technicians re-energized the 'B' channel which revealed a ground on 12 120V vital AC bus 1Y02-13. In response, maintenance technicians de-energized the channel and began troubleshooting activities to identify the location of the ground which was found to be in one of the newly installed power supplies. The defective power supply was replaced. At 12:35 a.m. on April 14, 2004, the licensee terminated the draindown of the RCS at an elevation of 37.4', the centerline of the hot leg, with a time to RCS boiling upon the complete loss of shutdown cooling being 16 minutes. At 4:33 a.m., the licensee attempted to re-energize the 'B' RPS channel again. This action caused the fuse for 120V Vital AC bus 1Y02-13 to blow which removed power from the bus. The de-energization of bus 1Y02-13 caused a loss of power to 1HIC-5208, 12 CC HX salt water outlet valve controller, causing it to fail low which resulted in the closure of 1CV-5208, 12 CC HX saltwater outlet valve. These consequences resulted in the loss of one of the two in-service SDC trains.

Maintenance technicians replaced the blown fuse in about 18 minutes which restored power to the affected bus, allowing control room operators to regain control of 1HIC-5208. Valve 1CV-5208 was then reopened, which restored not only the saltwater flow through the 12 CC HX, but also the affected SDC train. During this event, the RCS and CC System temperatures increased by 2 degrees Fahrenheit (°F) and 17 °F, respectively.

The inspectors reviewed the licensee's risk assessment requirements specific to the power supply replacement activity as stated in NO-1-103, Conduct Of Lower Mode Operations. Section K, Reduced Inventory Operations, steps 3.a.(2) and 3.a.(3) of that procedure stated that outage management will identify all activities that are scheduled during reduced inventory that could affect the equipment and areas identified by operations as needing protection, and that outage management will eliminate any activities that are identified that are not critical or near critical path from the schedule during reduced inventory. The inspectors conducted discussions with maintenance and outage management personnel and learned that this power supply replacement activity was not a critical path activity, and that the CC trains were considered equipment that warranted protection as defined in NO-1-207, Nuclear Operations Shift Turnover, Attachment 17, Minimum Essential Equipment For Unit 1 In Reduced Inventory. The inspectors determined that the licensee failed to identify that the power supply replacement activity could adversely affect a CC train, and therefore appropriately eliminate or postpone the scheduled power supply replacement activity.

Analysis. The performance deficiency associated with this finding was that the licensee failed to effectively manage the risk associated with the 'B' channel RPS maintenance activity, which ultimately resulted in the loss of a SDC train while in a reduced inventory condition. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or Calvert Cliffs procedures. This finding is greater than minor because it affected an attribute and objective of the Initiating Event Cornerstone in that human performance inadequacies resulted in an event that upset plant stability during shutdown operations.

This issue was evaluated in accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, Shutdown Operations SDP, and was determined to be of very low safety significance (Green). The issue screen through phase I of the SDP and a phase II safety significance evaluation was performed and reviewed by the Region I Senior Reactor Analyst. During this event, both trains of shutdown cooling remained in service. Only the salt service water supply to one of the two in service and cross-tied component cooling heat exchangers was lost. One fully redundant train of shutdown cooling remained available throughout this event. Instrumentation designed to alert control room operators of a degradation in shutdown cooling system performance remained operable. The ability to reopen the inadvertently closed salt service water valve manually from the control room remained available. Redundant RCS inventory makeup sources also remained available. The RCS temperature increase was 2°F during the 18 minute event. Based on the plant conditions at the time of this event, the phase II SDP analysis determined that this finding was of very low safety significance (Green). This finding, which involved ineffective work scheduling by outage management, was associated with the cross-cutting area of human performance.

Enforcement. 10 CFR Part 50.65(a)(4) states in part, "before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities." Contrary to the above, the licensee failed to perform an adequate risk assessment of the channel 'B' RPS maintenance activities. The licensee's risk assessment failed to identify the potential impact of the maintenance activity, thus precluding the licensee from appropriately managing the associated risk. Because this event was of very low safety significance, and has been entered into the licensee's corrective action program as IR4-030-018, this violation if being treated as an NCV, consistent with Section VI.A of the

NRC Enforcement Policy. NCV 05000317/2004-05-01, Failure to Assess and Manage Risk Associated With Unit 1 RPS Power Supply Replacement Activities During Reduced Inventory

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14 - 4 samples)

1. Unit 1 Loss Of The 12 Shutdown Cooling Train During Reduced Inventory Operations

a. Inspection Scope

The inspectors assessed operator performance associated with a Unit 1 event that involved the loss of one train of SDC while in reduced inventory conditions. This event occurred during the recent Unit 1 RFO on April 13, 2004, after the RCS had been drained to approximately 37.4' (mid-loop, reduced inventory operation).

After mid-loop RCS conditions had been reached, at approximately 4:33 a.m. on April 13, the licensee energized the 'B' RPS channel in an effort to restore the channel following power supply replacement activities. This was the second attempt to energize the channel. The first attempt resulted in the identification of a ground that was subsequently repaired. During the second attempt, power was lost to 120 volt vital ac bus 1Y02-13 and was attributed to a blown supply fuse. This failure resulted in a loss of power to 1HIC-5208, causing 1CV-5208, 12 CC HX Saltwater Outlet Valve, to fail close. This ultimately resulted in the loss of one train of SDC. This abnormal condition was terminated when maintenance technicians replaced the blown fuse, allowing 1CV-5208 to be manipulated to the pre-event position.

The inspectors reviewed operator logs, applicable procedures, and conducted discussions with operations and maintenance personnel. From this review, the inspectors noted that during this event, which lasted about 18 minutes, the RCS system heated up approximately two degrees. The inspectors also learned that an override switch to valve 1CV-5208 was present in the control room and remained an available method for control room operators to fully open 1CV-5208, if desired, and restore the affected SDC train. Control room operators indicated that this measure was not exercised since the RCS heatup was minimal. The inspectors concluded that operator performance during this event was appropriate. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified

2. 22 Circulating Water Pump Tripping and U-2 Emergent Power Reduction

a. Inspection Scope

The inspectors assessed operator performance associated with an emergent power reduction conducted on May 22, 2004, in response to the tripping of the 22 circulating water pump. The power reduction was performed to ensure that condenser vacuum did not lower to the point that a manual turbine trip, and subsequent reactor trip on Unit 2 was required.

At approximately 11:16 a.m. on May 22, maintenance technicians were pulling cables from an overhead conduit to restore from a temporary modification used during the recent Unit 1 refueling outage. This modification provided a power supply to miscellaneous containment loads by utilizing a spare 480 volt ac breaker. During this activity, one of the cables being pulled inadvertently struck the 22 circulating water pump exciter breaker, causing it to open, which resulted in the tripping of the 22 circulating water pump. Control room operators entered abnormal operating procedure (AOP) AOP-7G, Loss of Condenser Vacuum. In order to remain above the manual reactor trip criteria as stated in AOP-7G, control room operators reduced reactor power to approximately 85 percent. The 22 circulating water pump exciter, and the 22 circulating water pump breakers were closed, and reactor power was returned to 100 percent at approximately 7:00 p.m. the same day.

In order to assess operator performance during this abnormal event, the inspectors obtained and reviewed control room condenser vacuum recorder plots and control room logs, conducted discussions with various operations personnel, and reviewed AOP-7G to understand entry conditions and required actions contained within the procedure. Based on this review, the inspectors concluded that the licensee's response was appropriate and in accordance with approved operating procedures. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified

3. 12 Component Cooling Water Heat Exchanger Tube Failure Event

a. Inspection Scope

On April 16, 2004, Unit 1 reactor operators received a "CC Head Tank Level Low" alarm, and entered AOP-7C, Loss of CC Water, for a decreasing CC water system head tank level. The inspectors responded to the control room to assess the impact that the decreasing head tank level had on CC Water system operability, and to evaluate the performance of licensed operators during this abnormal event. In accordance with AOP-7C, operators were dispatched to perform visual inspections of susceptible areas in attempts to locate and isolate the leak. No visible leaks were identified. As a result, the licensee isolated the 11 CC HX, which had already been taken out of service for planned maintenance. This action, however, did not isolate the leak which operations personnel estimated to be approximately 100 gpm. Further troubleshooting revealed that the leak was located in the 12 CC HX, the remaining in-service CC HX.

The inspectors observed subsequent discussions conducted in the control room between the licensee management staff and operations personnel associated with a proposed repair plan for the 12 CC HX. The licensee elected to return the 11 CC HX to service, then isolate the 12 CC HX which terminated the event. Maintenance activities performed shortly after the 12 CC HX was isolated identified six sheared tubes.

The inspectors also reviewed operators' logs, plant traces, and the licensee's usage of AOPs while mitigating this event. Based on this review, the inspectors concluded the operators use of AOPs, and the implementation of technical specifications were appropriate. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified

4. Generator Temperature Monitoring System (GTMS) Indication Failure and U-1 Emergent Power Reduction

a. Inspection Scope

On June 17, 2004, Unit 1 reactor operators received a "Generator Temperature Monitoring System (GTMS)" alarm due to the indication of elevated hydrogen gas temperature associated with the Unit 1 main turbine generator. The licensee entered AOP-7E, Main Turbine Malfunction, and notified the inspectors. The licensee reduced reactor power in accordance with OP-3, Normal Power Operation, and monitored the gas temperature. When the gas temperature began to decrease, the licensee stopped reducing reactor power at 94 percent. Operators took actual temperature readings in the plant at the inlet and outlet of the gas cooler. Those readings were normal. The licensee exited AOP-7E based upon a systems engineering evaluation of the troubleshooting results which concluded that the GTMS recorder had failed at the alarm set point. On June 18, reactor power was restored to 100 percent.

The inspectors reviewed operators' logs, plant traces, and the licensee's use of AOPs while mitigating the event. The inspectors had discussions with operators and systems engineers and concluded the operators use of AOPs, operating procedures, and technical specifications were adequate. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to verify that the operability of systems important to safety was properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. In addition, the inspectors reviewed the selected operability determinations to verify they were performed in accordance with NO-1-106, Functional Evaluation - Operability Determination, and QL-2-100, Issue Reporting and Assessment. The applicable documents for this inspection are located in the Attachment. The inspectors reviewed the operability evaluations for the issues listed below:

- 21/22 CC Water Heat Exchangers (Historical Exposure To Excessive CC Flowrates Similar To 12 CC HX Tube Failure Event)
- 12 CC Water HX With Baffle Plate Vibrations
- 11 CC Water HX following SW Header Maintenance
- 13 Charging Pump Degraded Backup Selector Switch
- 22 Vital Battery Cracked Post Seal Rings

b. Findings

Introduction. The inspectors identified a self-revealing finding of very low safety significance (Green) which resulted in an NCV regarding the licensees failure to properly translate design calculation limits into operating procedures as required by 10 CFR 50, Criterion III, Design Control. Specifically, the licensee failed to incorporate a design flow calculation into an operating procedure which allowed the licensee to operate the 12 component cooling water heat exchanger (CC HX) in excess of its maximum shell side flow versus time curves. This excessive flow resulted in the failure of six tubes in the only available, and in-service CC HX which was supporting SDC of the RCS.

Description. On April 16, 2004, the Unit 1 control room operators received alarm “CC Head Tank Level Low,” and entered AOP 7C, Loss of CC Water. Operators initially believed a leak existed in the 11 CC HX, which was Out-Of-Service (OOS) for preplanned maintenance on the 11 salt water header. The 11 CC HX was isolated in an attempt to stop the leak, however operators later determined that the leak existed in the 12 CC HX. Operators isolated the 12 CC HX and restored the 11 SW header which allowed the 11 CC HX to be placed back in service. These actions terminated the event, and restored a non-degraded SDC loop to service. The licensee entered technical specification action statement 3.9.4.A for not having a shutdown cooling loop operable and in-service based upon an engineering evaluation that assessed the leakage and the degraded condition of the 12 CC HX. Once the 11 CC HX was returned to service, the licensee exited the action statement for having both trains of

SDC inoperable. Investigation and troubleshooting of the alarm condition revealed that the lowering water level was caused by tube failures in the 12 CC HX which was induced by excessive flow rates. After non-destructive testing (NDT) and visual inspections were performed, the licensee removed and plugged a total of 25 tubes.

The inspectors reviewed the pre-event CC system configuration and determined that the excessive shell-side CC flow in the 12 CC HX was initiated when the licensee performed OI-29, Saltwater System, which secured CC flow through the 11 CC HX, in preparation for 11 salt water header maintenance activities. This alignment resulted in two CC water pumps remaining in service with flow through only the 12 CC HX.

The licensee determined that the flow rate through the 12 CC HX during this event was about 7300 gpm. The inspectors reviewed operational logs, computer data points, and the licensees calculations and reached the same conclusion. Through follow up discussions with engineering personnel, the inspectors learned that in 1993 the licensee contracted a vendor to develop a bounding analysis of flowrates specifically for the CC water heat exchangers in the CC system. The inspectors reviewed flow-related graphs contained in the analysis, and concluded that flows greater than 7500 gpm would result in vibrational tube failure in a relatively short period of time. The inspectors, however, also noted that the graph was in error. The vendor's analysis concluded that at a flow rate of 7300 gpm, the tubes would have a life of 65 days when in fact they only lasted approximately 17 hours during the event. The inspectors conducted discussions with licensee engineering and technical staff regarding procedural precautions and limitations for the 11 salt water header alignments and corresponding flow limitations for the CC water system. The inspectors concluded that the licensee failed to appropriately incorporate the flow limitations as specified in the CC system flowrate bounding analysis in OI-29, Saltwater System, when performing the saltwater system alignment. As a result, 7300 gpm was allowed to flow through the 12 CC HX for approximately 18 hours.

Additional reviews and discussions conducted with engineering personnel revealed that the CC system flowrate bounding analysis was to be further updated by the licensee so that limiting design basis flow rates could be established in order to preclude future tube rupture events. The inspectors also learned and verified that compensatory measures were in place for both units to maintain both CC HXs continuously in service to prevent a post-safety injection actuation signal (SIAS) two pump/one HX system alignment.

The inspectors concluded that prior to this event, both Unit 1 and Unit 2 were susceptible to high flow through a CC HX following a SIAS actuation until a CC pump was secured in accordance with emergency operating procedures. In order to assess and evaluate the significance associated with this condition, a follow-up inspection is warranted after the licensee has completed the development of a CC system limiting HX flowrates. This issue is considered an unresolved item pending further review and will be documented as URI-50-317, 318/2004-05-03, Review Of Updated CC HX Design Basis Analysis.

Analysis. The performance deficiency in this event was inadequate design control (10 CFR 50, Appendix B, Criterion 3) in that the operating procedures and material specifications specified failed to prevent the CC HX from being operated at flow rates and for a length of time that allowed six of the HX tubes to fail (and 18 other tubes to crack) due to high cycle fatigue. The failure of these tubes and the resultant loss of CC water challenged both trains of this normally cross connected system to perform its shutdown safety function to support core decay heat removal.

Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or Calvert Cliffs procedures. This finding is greater than minor because it affected an attribute and the objective of the Mitigating System Cornerstone in that inadequate procedure quality resulted in degraded availability, reliability and capability of a system that responds to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter (IMC) 0609, Appendix G, Shutdown Operations SDP, Table 1, PWR Refueling Operation RCS Level Greater than 23 Feet, a regional risk analyst conducted a Phase 1 screening and determined that the finding is of very low safety significance (Green) since the safety function of the CC water system was not lost during operator actions to isolate the 12 CC HX and place the 11 CC HX in service. Throughout these operator actions, the CC water system continued to operate to support reactor core decay heat removal.

Enforcement. 10 CFR 50, Criterion III, Design Control, states in part that measures shall be established to assure that design basis are correctly translated into procedures. Contrary to this requirement, the licensee failed to incorporate a design flow calculation into an operating procedure OI-29, Saltwater System, which allowed the licensee to operate the 12 CC HX in excess of its maximum design shell side flow versus time curves. As a result, twenty-five tubes in the CC HX were damaged and the CC HX system was challenged. Because this finding is of low safety significance and has been entered into the licensee's CAP as IR4-030-055, IR4-031-212, and IR4-029-536, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC enforcement policy. NCV 050000317/2004-05-02, Failure To Implement Design Control Measures For The 12 CC HX.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

The inspectors reviewed a permanent plant modification associated with the installation of a forced oil system on the 11 AFW pump. This inspection was performed to verify the adequacy of the modification package, and to verify that design and licensing bases requirements of the system were not degraded during associated work activities. The inspectors also verified that post-modification testing was completed in accordance with established station procedures which adequately demonstrated continued reliability and satisfactory performance of the AFW system. Documents reviewed during the course of this inspection are listed in the Attachment at the end of this report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 samples)

a. Inspection Scope

The inspectors observed and/or reviewed post-maintenance tests associated with the following work activities to verify that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance. The applicable documents for this inspection are located in the Attachment. Post-maintenance testing associated with the following maintenance activities was reviewed:

- 12 Control Room HVAC Maintenance Activities
- 11 Charging Pump Repacking
- Channel "D" RPS T_{hot} Input Reading Low
- 12 CC HX Tube Removal and Plugging Activities
- 1A EDG Maintenance Activities

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 1 sample)

1. Fuel Handling/Core Refueling Activities

a. Inspection Scope

The inspectors conducted an inspection of the containment and spent fuel pool area during fuel handling operations and verified that these activities were conducted within the bounds of technical specifications, and in accordance with approved licensee procedures. The inspectors confirmed with reactor operators the operability of boration flowpaths and nuclear instrumentation detectors, and confirmed that the control room maintained constant communications with the fuel handling team. Near the conclusion of the Unit 1 outage, the inspectors interviewed nuclear fuels management personnel, reviewed core loading software programs and core loading documents, and verified that fuel assemblies were properly tracked from core offload through core reload. The inspectors concluded that the fuel assemblies were placed in the correct locations. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified.

2. Reduced Inventory Operations

a. Inspection Scope

During the Calvert Cliffs, Unit 1, Spring 2004 RFO, the water level in the reactor vessel was lowered to the middle of the RCS hot leg penetrations on the reactor vessel, commonly referred to a “mid-loop” condition, for various maintenance activities. Mid-loop is approximately 4 feet above the top of the core. The inspectors observed this infrequent evolution from the control room, reviewed the applicable operating procedures and engineering documentation associated with the additional level instrumentation required during mid-loop operations. In addition, the inspectors interviewed the designated point of contact for the evolution, personnel from the operating crew, and the Operations General Supervisor. The documentation reviewed was:

- Operating Procedure OP-7, Revision 7, Shutdown Operations
- Calvert Cliffs Unit One “RFO 2004 Water Level Plan”
- BG&E Memorandum #PDSU-96-074/DE00986, March 28, 1996, “Reduced Inventory/Mid-Loop Operations Instrumentation”
- Calculation #I-92-38, Revision 1, “Instrument Loop Uncertainty Estimate, RCS Mid-Loop Narrow Range Level Monitor Loop”
- 50.59 Safety Evaluation of #91-B-078-093-R0, “Installation of Narrow and Wide Range Reactor Vessel Level Instrumentation for Use During Mid-Loop Operation”
- Calvert Cliff’s response to Generic Letter #88-17, “Loss of Decay Heat Removal, 10CFR50.54(f)”
- IR4-030-907

b. Findings

No findings of significance were identified.

3. Containment Closeout Inspection

a. Inspection Scope

Near the end of the Unit 1 RFO, the inspector conducted a containment closeout inspection with licensee personnel, as required by OP-6, Pre-Startup Checklist. The inspection included a complete walkdown of all elevations and areas in the containment building. The inspector verified that the licensee personnel performed a thorough inspection, that equipment stored was properly secured, and that all material to be removed from containment was either removed or noted as requiring removal prior to the final closeout.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors observed and/or reviewed the six surveillance tests listed below associated with selected risk-significant SSCs to verify that technical specifications were properly complied with, and that test acceptance criteria were properly specified. The inspectors also verified that proper test conditions were established as specified in the procedures that no equipment preconditioning activities occurred, and that acceptance criteria had been met. The following surveillance tests were reviewed:

- STP O-73D-1, Charging Pump Performance Test
- STP O-73H-1, AFW Pump Large Flow Test
- STO O-36-1, Shutdown Cooling Hdr Return Isolation Valve Test
- STP-O-87-1, Borated Water Source 7 Day Operability Verification
- STP-M-212D-1, Channel “D” reactor Protective System Functional Test
- STP-O-5A-1, AFW System Surveillance Test

b. Findings

No findings of significance were identified

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed a control room simulator training exercise conducted on June 18, 2004, to assess licensed operators’ performance in the area of emergency preparedness. This training exercise focused on equipment failures and operator challenges that would typically exist during RCP seal package failures which resulted in LOCA events. The required procedural transitions and associated event classifications were observed and evaluated by the inspectors. Details pertaining to this inspection are provided in Section 1R11 of this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 14 samples)

a. Inspection Scope

The inspector reviewed radiological work activities and practices and procedural implementation during observations and tours of the facilities and inspected procedures, records, and other program documents to evaluate the effectiveness of Calvert Cliffs access controls to radiologically significant areas. This inspection activity represents the completion of fourteen (14) samples relative to this inspection area (i.e., inspection procedure sections 02.01, 02.02.a through f, 02.03.b, c, and d, 02.04.a through c, and 02.05.c) in partial fulfillment of the annual inspection requirements.

Inspection Planning (02.01)

The inspector found no licensee Performance Indicators (PIs) for the Occupational Exposure Cornerstone to review for follow-up.

Plant Walkdowns and Radiation Work Permit (RWP) Reviews (02.02.a through f)

The inspector identified exposure significant Special Work Permits (SWPs) and work areas within radiation areas, high radiation areas (<1 R/hr), or airborne radioactivity areas in the plant and reviewed associated licensee controls and surveys of these areas to determine if controls were acceptable. The reviewed SWPs included 2004-1306 to remove and replace In-Core Instrumentation (ICI) flanges, 2004-1324 for volumetric inspections of the reactor head, and 2004-1408 for steam generator nozzle dam installation and removal. The inspector walked down these areas with a survey meter and/or observed these areas via closed-circuit television and the teledosimetric results from the licensee's remote radiation safety central monitoring station. The inspector determined that the prescribed SWP, procedural, and engineering controls were in place, that the licensee's surveys and postings were complete and accurate, and that air samplers were properly located.

The inspector reviewed the SWPs used to access these and other high radiation areas and identified what work control instructions and control barriers were specified. The barriers met generic Technical Specification High Radiation Area (HRA) requirements which were reflected in the licensee's procedures. The inspector reviewed the electronic personnel dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspector verified that workers knew what actions were required when their electronic personnel dosimeters alarmed.

The inspector reviewed selected, previously-cited SWPs for airborne radioactivity areas with the potential for individual internal exposures of greater than 50 millirems cumulative effective dose equivalent (CEDE). For these selected areas, the inspector verified engineering-controls performance based on air monitoring, air sampling, and whole body counting results. Additionally, the inspector examined the licensee's physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool.

During the week of June 14, 2004, the inspector reviewed the internal dose assessment results for the current year. There were no recorded internal exposures greater than 50 millirems as committed effective dose equivalent. The inspector reviewed the dry-active-waste (DAW) analysis for the 2003-Health-Physics (HP)-DAW scaling factors used for internal dose commitment calculations.

Problem Identification and Resolution (02.03.b, c, and d)

During the week of June 14, 2004, the inspector reviewed issue reports (i.e., corrective action reports) related to access controls, including those involving high radiation areas. The inspector discussed the issue reports with radiation protection staff to determine how the issues were prioritized and to determine the adequacy of the corrective actions. The radiation protection group has performed self-assessments and trending of issue reports on a quarterly basis. There were no Performance Indicator (PI) events identified year-to-date.

Job-In-Progress Reviews (02.04.a through c)

As noted previously, the inspector selected several jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (less than 1 R/hour) for observation (i.e., SWPs to remove and replace ICI flanges, for volumetric inspections of the reactor head, and for steam generator nozzle dam installation and removal). The inspector reviewed the radiological job requirements (SWP requirements and work procedure requirements) and attended the update pre-job briefings for the reactor head visual inspections (external of the head) and for the nozzle dam installation on steam generator 12. The inspector determined that radiological conditions were adequately communicated and that the radiological controls, including surveys, radiation protection job coverage, and contamination controls were adequate.

The inspector reviewed the application of dosimetry to effectively monitor exposure to personnel for the steam generator nozzle dam installations. In this case, multiple whole body dosimeters and extremity dosimeters were utilized due to the dose rate gradients involved. The inspector verified that the licensee controls were adequate.

High Risk Significant, High Dose Rate HRA and VHRA Controls (02.05.c)

The inspector reviewed the licensee's procedures for posting and locking of entrances to high-dose-rate high radiation areas and Very High Radiation Areas (VHRAs). The inspector verified that the entrance to the cavity under the reactor vessel was adequately posted and locked.

Related Activities

On April 13 and 14, the inspector observed Radiologically Controlled Area (RCA) entries and exits being made by radiation workers at the primary RCA access control point to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of alarming electronic radiation dosimeters. The inspector toured various elevations in the auxiliary buildings and in the Unit 1 reactor containment building to verify the adequacy of the radiological controls which were being implemented during this Unit 1 2004 RFO. The inspector reviewed observed work activities for compliance with the special work permit (SWP) requirements. During these observations and tours the inspector reviewed, for regulatory compliance, the posting, labeling, barricading, and level of radiological access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas. On April 13, 14, and 15, the inspector observed the morning turnover meetings for the Health Physics (HP) technicians.

The inspector performed a selective examination of documents (as listed in the List of Documents Reviewed section) to evaluate the adequacy of radiological controls.

The review in this area was against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts D, F, G, H, I, and J), Technical Specifications, and procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 8 samples)

a. Inspection Scope

The inspector reviewed the effectiveness of the licensee's program to maintain occupational radiation exposure as low as is reasonably achievable (ALARA). This inspection activity represents the completion of eight (8) samples relative to this inspection area (i.e., inspection procedure sections 02.01.b through d, 02.02.a, b, and c, 02.03.b, and 02.07) in partial fulfillment of the annual inspection requirements.

Inspection Planning (02.01.b through d)

The inspector reviewed the Unit 1 2004 RFO work scheduled during the inspection period and associated work activity exposure estimates and selected work activities which were likely to result in the highest personnel collective exposures. The inspector also examined the site-specific trends in collective exposures for refueling outages. Additionally, the inspector reviewed the site-specific procedures associated with maintaining occupational exposures ALARA, including the processes used to estimate and track work-activity-specific exposures.

Radiological Work Planning (02.02.a, b, and c)

Using a list of outage work activities ranked by estimated exposure obtained from the licensee, the inspector selected work activities that were in progress which represented the highest exposure significance. Having done that, for the selected work activities, the inspector then reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The selected work activities included the following: the ISI/NDE activities (SWP 2004-1002), the removal and replacement of ICI flanges (SWP 2004-1306), the volumetric inspections of the reactor head (SWP 2004-1324), the reactor head visual inspections (external of the head)(SWP 2004-1325), the installation and removal of the steam generator nozzle dams (SWP 2004-1408), and the steam generator ECT, sleeving, and tube plugging activities (SWP 2004-1409). The inspector determined that the licensee had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that were ALARA.

During the week of June 14, 2004, the inspector compared the results achieved (i.e., person-rem used) during the recent Unit 1 RFO with the intended doses established in the licensee's ALARA planning for these work activities. The inspector reviewed the ALARA packages for several specific Special Work Permits (SWPs) which are identified in the List of Documents Reviewed section. The inspector also reviewed the draft post-outage report for 2004 for HP.

Verification of Dose Estimates and Exposure Tracking Systems (02.03.b)

During the week of June 14, 2004, the inspector reviewed the licensee's method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work are encountered. The inspector examined the ALARA packages for several specific SWPs to review the ALARA in-process/post-job reviews. The inspector examined the criteria in Procedure RSP-1-200, ALARA planning and SWP preparation, for initiating ALARA in-process/post-job reviews.

Declared Pregnant Workers (02.07)

The inspector, during the week of June 14, 2004, reviewed the licensee's process for monitoring declared pregnant workers with respect to the requirements of 10 CFR 20.1208. The inspector reviewed the monitoring results for the pregnant workers who declared in 2004. The inspector also reviewed documentation concerning area surveillance doses used for demonstrating some monitoring results for such workers.

Related Activities

The inspector performed a selective examination of documents (as listed in the List of Documents Reviewed section) for regulatory compliance and for adequacy of control of radiation exposure.

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701 (Use of process or other engineering controls), and procedures.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 3 samples)

a. Inspection Scope

The inspector reviewed the program for HP instrumentation and protective equipment to determine the accuracy and operability of the instrumentation and equipment. This inspection activity represents the completion of three (3) samples relative to this inspection area (i.e., inspection procedure sections 02.01 and 02.06.a and b) in partial fulfillment of the annual inspection requirements.

Inspection Planning (02.01)

The inspector reviewed the plant's Updated Final Safety Analysis Report (UFSAR) to identify applicable radiation monitors associated with transient high and very high radiation areas including those used in remote emergency assessment. The inspector identified the appropriate installed area and process radiation monitors, emergency assessment instrumentation, and portable radiation instruments that are used to identify changing radiological conditions such that actions to prevent an overexposure may be taken. The identified monitors, instrumentation, and instruments will be examined in future inspections.

Self-Contained Breathing Apparatus (SCBA) Maintenance and User Training (02.06.a and b)

During the week of June 14, 2004, the inspector reviewed the status and surveillance records of SCBA staged and ready for use in the plant. The licensee stated that there were sufficient staged air bottles to last for ten hours in an emergency; however, the inspector examined the licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions; there were two breathing-air compressors on the site. The inspector reviewed records to selectively verify that control room operators and other emergency response and radiation protection personnel (assigned in-plant search and rescue duties or as required by emergency response training) were trained and qualified in the use of SCBA (including personal bottle change-out). The inspector also reviewed records to verify that personnel assigned to refill bottles are trained and qualified for that task.

Also, during the week of June 14, 2004, the inspector selectively verified that only personnel who possess manufacturer-certified training/qualifications were allowed to perform maintenance and repairs on SCBA components vital to the unit's function. The inspector also selectively reviewed records for repairs on the vendor-designated vital components of several SCBA units and on the periodic air cylinder hydrostatic testing. The licensee used the vendor's maintenance procedures and test program for the vital components.

Related Activities

The inspector performed a selective examination of documents (as listed in the List of Documents Reviewed section) for regulatory compliance and adequacy in this area.

The review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, Technical Specifications, and procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

40A1 Performance Indicator Verification (71151 - 10 samples)

1. Initiating Events

a. Inspection Scope

For the period from April 2003 through March 2004, the inspectors examined the licensee's PI data and plant records associated with the PIs listed below for both units, including licensee event reports, NRC inspection reports, operator narrative logs, and associated issue reports. The review was against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment PI Guideline, Revision 2.

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

b. Findings

No findings of significance were identified.

2. Mitigating Systems Cornerstone

a. Inspection Scope

For the period from April 2003 through March 2004, the inspectors examined the licensee's PI data and plant records associated with the PI listed below for both units, including licensee event reports, selected operator narrative logs, maintenance work orders, system health reports, and associated issue reports. The review was against the applicable criteria specified in NEI 99-02, Regulatory Assessment PI Guideline, Revision 2.

- Safety System Unavailability, High Pressure Injection System

b. Findings

No findings of significance were identified.

3. Barrier Integrity

a. Inspection Scope

For the period from January 2003 through March 2004, the inspectors examined the licensee's PI data and plant records associated with the PI listed below for both units, including RCS nuclide data. The review was against the applicable criteria specified in NEI 99-02, Regulatory Assessment PI Guideline, Revision 2. Additionally, the inspectors observed chemistry technicians obtain and analyze a Unit 2 RCS sample in accordance with CP-401, Nuclear Steam Supply System Sampling, and CP-935, Determination of Reactor Coolant Isotopic Activity.

- RCS Activity

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 samples)

1. Inservice Inspection (ISI) - Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a sample of issue reports listed in Attachment 2, under 4OA5, Other, which identified flaws and other nonconforming conditions discovered during this and the previous outage. The inspectors verified that the nonconforming conditions identified were reported, characterized, evaluated and appropriately dispositioned and entered into the CAP.

b. Findings

No findings of significance were identified.

2. Identification and Resolution of Problems - Occupational Radiation Safety

a. Inspection Scope

During the week of April 12, 2004, the inspector selected three issues identified in the CAP for detailed review (i.e., IR4-023-768, IR4-003-922, and IR4-031-118). The issues were associated with source checking of radiation detection instrumentation, entries into LHRAs, and HRA boundary conditions.

During the week of June 14, 2004, the inspector examined ten issues identified in the CAP for detailed review (i.e., IR4-002-488, IR4-003-866, IR4-012-452, IR4-015-259, IR4-023-604, IR4-028-244, IR4-029-737, IR4-032-691, IR4-032-799, and IR4-032-975). These issues were associated with lack of procedural adherence by radiation protection technicians and by radiation workers involving air sampling, checking electronic dosimeters on a periodic basis, and work-in-process ALARA activities.

The documented reports for the issues were reviewed to determine whether the full extent of the issues was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

b. Findings

No findings of significance were identified.

3. Corrective Action Review by Resident Inspectors

a. Inspection Scope

Continuous Review

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s CAP. This review was accomplished by reviewing each condition report (CR), attending daily screening meetings, and accessing the licensee’s computerized database. This review was also performed during the 1^{st} quarter 2004 inspection period.

Semi-Annual Problem Identification and Resolution Review

The inspectors performed an in-depth, semi-annual, PI&R review of licensee documents written from January 2004 through June 2004 to verify that the licensee is identifying issues at the appropriate threshold, entering them into the corrective actions program, and ensuring that there are no significant, adverse trends outside of the corrective actions program which would indicate the existence of a more significant safety issue. The inspectors reviewed licensee PIs, self-assessment reports, quality assurance audit/surveillance reports, corrective action reports, and systems health reports and compared the results of the review with results reported in the NRC baseline inspection program. Additionally, the inspectors evaluated the reports against the requirements of the Constellation Nuclear’s CAP as delineated in QL-2, Self-Assessment/Corrective Action Program.

a. Findings

No findings of significance were identified.

4. 11B Reactor Coolant Pump Lower Seal Failures (71152 - 1 sample)

a. Inspection Scope

The inspectors assessed the corrective actions associated with repetitive failures of the lower seal on the #11B RCP. The inspectors reviewed the issue reports for each of the failures of the lower seal, the associated causal evaluations, and actions to correct the problem. The inspectors also reviewed the engineering evaluation for the reasonable expectation for continued operation, the probabilistic risk analysis assuming the lower seal was failed, and the Maintenance Rule (a)(1) status action plan to return the system to (a)(2) status. In addition, the inspectors interviewed the system engineer and the risk analyst. This inspection represented one inspection sample. The documentation reviewed was:

- (a)(1) Evaluation, Revision 0, “Corrective Action and Goal Setting Plan for System 064B, Reactor Coolant Pumps”
- CCNPP Reliability Engineering Report, Revision 1, “QSS Evaluation of RCP Seal 1-of-4 Stages Unavailable (Remaining Seals Not Affected)”
- ESP #ES200400043, Revision 0, “Functional Evaluation of 11B RCP to Determine If Acceptable for Continued Operation”
- Root Cause Analysis #IR200300458 - “Repeat Failures of 11B RCP Lower Seal Stage”
- Issue Reports: IR1-043-156, IR3-036-934, IR3-063-312, IR4-019-662, IR4-022-943, IR4-026-711

b. Findings

No findings of significance were identified. In 1996, the licensee replaced the impeller and the pump cover HX. After the first failure in 1996 (26 days after the plant was started up), the licensee found metallic debris in the seal and assumed that it was from an inadequate flush of the HX. The causal analyses for the second and third failures (1999 and 2002) resulted in the same result; i.e., debris from the HX, even though it had been flushed several times. After the failure in December 2003, the licensee conducted a detailed root cause analysis. Since the failures started after changes to the pump in 1996, and since cleaning of the HX had not prevented further failures, the licensee determined that the HX was a possible but improbable cause. The root cause was determined to be the new impeller. The impeller on the 11B pump was attached to the shaft differently than the other pumps; specifically, it was welded to the shaft, the impellers on the other pumps were all bolted to the shaft. The corrective actions include replacement of the impeller during the 2006 RFO. The inspector determined that the final root cause analysis was detailed and thorough, and the planned corrective actions should prevent recurrence.

5. 480 Volt Breaker Inspection (71152 - 1 sample)

a. Inspection Scope

The inspector reviewed corrective actions associated with the maintenance, troubleshooting and repair of 480v breakers. Specifically, corrective actions and cause determinations for numerous IR issued on 480v DS and DSL breakers were reviewed and examined in detail. The inspector selected four CRs for review: CR-IP3-2003-00195, CR-IP3-2003-00196, CR-IP3-2003-00200, and CR-IP3-2003-00288. Additionally, the apparent cause evaluations from CR-IP3-2003-00196 and CR-IP3-2003-00200 were reviewed in detail.

b. Findings

b.1 Breaker (CX Relay) Testing Deficiency

Introduction. The inspectors identified an NCV of very low safety significance (Green) for the licensee’s failure to comply with 10 CFR 50, Appendix B, Criterion XVI Corrective Action. The licensee failed to correct a 480v breaker (CX relay) testing deficiency discovered in 1998. When action was taken in 2001, the actions were not effective in preventing two subsequent failures, one in December 2003 and one in February 2004.

Description. On December 4, 1998, IR3-019-254 reported a problem with breaker 52-1106 for #11 CC Pump. The cause of the problem was the failure to test (since original plant construction) the CX relay associated with this electrically-operated 480v DS breaker. The breaker failure occurred when a normally closed CX relay contact was found to be open. The corrective actions for this condition lead to the specification of a contact resistance test with an acceptance criteria of less than 1 ohm resistance. IR3-019-254 specified that a corrective action be taken to change procedures FTE-52 and FTE-53 to include testing of the CX relays with the normal breaker preventive maintenance. This action was to be tracked by AIT IH19990006 MS004. This corrective action was closed; however, the procedures were not changed and the CX relays were not tested.

On October 23, 2001, IR3-071-275 was written to revise maintenance procedures to begin testing the CX relays when performing maintenance on the 480v DS and DSL breakers. Thus, the condition, reported in 1998 under IR3-019-254, which existed since original plant construction continued to exist until October 23, 2001. At this time the maintenance procedures were changed to include a contact resistance test, with an acceptance criteria of less than 1 ohm resistance. No test hold time was specified in this test.

On December 22, 2003, the #22 control element drive mechanism (CEDM) cooling fan failed to start. The failure was determined to be due to breaker 52-2319 failing to close due to a failed (1427 ohm resistance) CX relay contact. This failure was reported via IR4-027-382 (12/22/03). However, the cause of the relay failure was initially attributed to the age of the relay and not the inadequacy of the test.

On February 19, 2004, IR4-002-246 reported the failure of a pressurizer heater breaker 52-1427 to close. This failure was attributed to a failed CX relay. Again, the cause, was initially attributed to the age of the relay.

The causal analysis for IR4-027-382 which was completed on 2/27/04 determined that the contact resistance test (which was enacted in October 2001) was not adequate to determine if the relay was fully operable. Further investigation indicated the need to hold the test conditions for a longer period of time to ensure that the coil was de-energized and the relay carrier returned full travel. Upon discovery of this condition, the licensee immediately launched a program to replace all CX relays on the 480v breakers. For those which could not be replaced (due to lack of replacement relays) compensatory actions were put in place to ensure operation.

There are 85, 480v electrically operated breakers which depend on CX relays for their operation. These breakers were exposed to potential failure to operate due to lack of testing of the CX relays. All of these breakers are in safety related applications.

Analysis. This performance deficiency is more than minor because it affects the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affects the availability and reliability of the 480v electrical distribution system.

This finding is not a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18. The finding does not represent an actual loss of safety function of a system. Since the finding does not represent an actual loss of safety function of one or more Technical Specification trains for greater than its Technical Specification Allowed Outage Time a Phase 2 SDP Evaluation was not necessary. Because the finding did not screen as potentially risk significant due to a seismic, fire, flooding or severe weather initiating event, the final result of the Phase 1 SDP is a finding of very low safety significance, Green. This finding, which involved the failure to promptly identify the cause for the 480v breaker deficiencies, was associated with the cross-cutting area of Problem Identification and Resolution.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action requires, in part, "...Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies...are promptly identified and corrected. In the case of significant conditions adverse to quality, the measure shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to this requirement, the licensee failed to promptly identify the cause of the 480v breaker malfunctions resulting from the CX relay testing deficiency discovered in 1998. Because this finding is of very low safety significance and has been entered into the licensee's CAP (IR4-028-182), it is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. NCV 05000317, 05000318/2004-05-04, Inadequate Corrective Actions For 480v Breaker Testing Deficiency.

b.2 Inadequate Breaker Maintenance Procedures

Introduction. The inspectors identified an NCV of very low safety significance (Green) for the licensee's failure to comply with Technical Specification 5.4.1.a. when procedures to ensure that planned, scheduled maintenance was actually being performed were not implemented. This performance deficiency resulted in the failure of several breakers to operate when required.

Description. Calvert Cliffs procedure FTE-53 , Revision 14, Westinghouse DS-206 Circuit Breaker and Cubicle Inspection and procedure FTE-52, Westinghouse Circuit

Breaker and Cubicle Inspection, Rev. 4, Section D, Lubrication, Step 1. States INSPECT, CLEAN AND LUBRICATE the following: a) IF needed, CLEAN AND LUBRICATE sparingly with Molykote BR2+, the following points:... This procedure was discussed with electrical maintenance supervision and electrical technicians. Persons interviewed said that this procedure gave them the latitude to determine, on their own judgement whether or not to clean and re-lubricate the specified points in the breaker operating mechanism.

Interviews with Engineering and Maintenance Management revealed that they were under the impression that scheduled maintenance in accordance with these procedures was actually being performed. In actuality the maintenance was being scheduled, however, no record exists as to what was actually being done to the breakers.

The inspector identified approximately 18 Issue Reports (IR) on circuit breaker failures which had been attributed to dirty mechanisms, mechanism gummed up with grease, lubrication needed, excessive grease on the operating mechanism, dirty drive link, dried grease or which operated successfully after performing the maintenance specified in FTE-52 and FTE-53 during subsequent troubleshooting. This problem affected both safety related and non-safety related breakers. The inspector reviewed maintenance records back to 1996 and found the same wording in FTE-52 and FTE-53.

Analysis. This performance deficiency is more than minor because it affects the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affects the availability and reliability of the 480v electrical distribution system.

This finding is not a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18. The finding does not represent an actual loss of safety function of a system. Since the finding does not represent an actual loss of safety function of one or more Technical Specification trains for greater than its Technical Specification Allowed Outage Time a Phase 2 SDP Evaluation was not required to be completed.

Since the finding did not screen as potentially risk significant due to a seismic, fire, flooding or severe weather initiating event, the final result of the Phase 1 SDP is a finding of very low safety significance, Green.

Enforcement. Calvert Cliffs Unit 1 Technical Specifications, 5.4.1. requires "Written procedures be established, implemented, and maintained covering the following activities: a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 , Section 9 Procedures For Performing Maintenance, paragraph a) requires, "Maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures..." Paragraph b) requires, "Preventive Maintenance schedules should be developed to specify lubrication schedules, inspections of equipment..." Contrary to this requirement, the licensee's procedures did not ensure that the specified maintenance was actually being performed as required. This condition has existed since, at least, 1996. Because this performance deficiency was determined to be of very low significance and has been entered into the CAP (IR4-028-183), this finding is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. NCV 05000317/2004-05-05, Failure to Implement Planned, Scheduled Maintenance.

6. Cross-Reference for Cross-Cutting Findings

Section 1R13 describes a human performance finding that outage management failed to properly schedule a maintenance activity that resulted in the loss of one train of shutdown cooling.

Section 2OA2 describes a corrective action finding which involved the failure to promptly identify the cause for the 480v breaker deficiencies

4OA5 Other Activities

1. TI 2515/150, Revision 2 - Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Bulletin 2002-02)

a. Inspection Scope

The inspectors reviewed the licensee's examination activities performed in response to NRC Order EA-03-009, "Establishing Interim Inspection Requirements for RPV Heads at Pressurized Water Reactors," dated February 11, 2003, and NRC Bulletin 2002-02, "RPV Head and Vessel Head Penetration Nozzle Inspection Programs." These examination activities were inspected using Temporary Instruction (TI) 2515/150, Revision 2. In addition, the inspectors reviewed examination plans and test method commitments provided by the licensee in their response to NRC Bulletin 2002-02. The inspectors reviewed the licensee's request for relaxation from NRC Order EA-03-009 dated January 30, 2004. The licensee had requested relaxation of the requirement to ultrasonically examination (UT) each RPV head penetration nozzle (i.e., nozzle base material) up to two inches above the J-groove (weld) because of limited accessibility for insertion of the UT probe in some vessel head penetrations (VHP).

The inspectors reviewed the licensee's inspection methods to detect evidence of leakage and/or cracking of RPV head penetration (CEDM, in core instrumentation and the vessel head vent) nozzles. The licensee performed a VT of selected samples of penetrations to evaluate the integrity of vessel head and penetration intersections to confirm the absence of flaws and boric acid deposits. The inspectors interviewed examination personnel, data analysts and engineering personnel and reviewed training and qualification records to verify that licensee personnel were properly trained to perform the reactor vessel head inspections. The inspectors also reviewed the examination procedures to determine whether they provided adequate guidance and acceptance criteria to perform the examinations.

The inspectors selected seven CEDM nozzle penetrations to observe and evaluate the effectiveness of the VT and UT to verify that the test methods could reliably detect flaws, actual leakage or identify a “leak path” from a failure in the vicinity of vessel head penetrations. The inspectors 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 inspectors verified that the examination procedures required that anomalies, deficiencies and discrepancies identified during the examination process be evaluated and documented in accordance with the licensee’s CAP. The applicable documents for this inspection are located in the Attachment.

b. Findings

No findings of significance were identified.

2. TI 2515/156, Offsite Power System Operational Readiness

a. Inspection Scope

The inspector performed Temporary Instruction 2515/156, Offsite Power System Operational Readiness. The inspector collected and reviewed information pertaining to the offsite power system specifically relating to the areas of the maintenance rule (10 CFR 50.65), the station blackout rule (10 CFR 50.63), offsite power operability, and corrective actions. The inspector reviewed this data against the requirements of 10 CFR 50 Appendix A General Design Criterion 17, Electric Power Systems, and Plant Technical Specifications. This information was forwarded to the Office of Nuclear Reactor Regulation (NRR) for further review.

b. Findings

No findings of significance were identified

4OA6 Meetings, Including Exit

On July 2, 2004, the inspectors presented the inspection results to Kevin Neitmann and other members of his staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Technical Specification 5.4.1.a, Calvert Cliffs Procedure RP-1-100 (Radiation Protection) Section 5.2.3.u, and SWP No. 2004-1016 (Activity 1) require that continuous HP coverage be implemented for entries into posted locked high radiation areas (LHRAs). Contrary to the above, on April 10, 2004, a radiation worker entered a posted LHRA without such coverage in violation of the SWP. This event is documented in the licensee's CAP as IR4-003-922. This finding is of very low safety significance because it did not involve a personnel overexposure, a substantial potential for the same, or compromise the ability to assess dose.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:

J. Blankenship, Health Physics Work Leader (Technology)
S. Brown, Health Physics Work Leader (Radiological Engineering)
R. Caredo, Probalistic Risk Analyse
B. Dansberger, Health Physics Work Leader (Materials Processing Facility)
D. Demore, Health Physics Work Leader (Operations)
E. Deogracias, Health Physics Work Leader (Dosimetry)
H. Evans, Health Physics Work Leader (Operations)
C. Fabricante, Materials Processor
R. Greene, Health Physics Technician
J. Guidotti, Health Physics Work Leader (Radiation Instruments)
J. Johnson, Health Physics Technician
V. Johnson, Health Physics Technician
P. Jones, Senior Plant Health Physicist
T. Kirkham, Health Physics Operations Supervisor
J. Lenhart, Health Physics Work Leader (Operations)
M. Lewis, System Engineer, Reactor Coolant Pumps
R. Lopez, Health Physics Technician
R. Martin, Mid-Loop May 3, Dedicated SRO
J. Meyers, Materials Processor
K. Mills, Operations General Supervisor
R. Pace, Shift Manager
I. Rice, Health Physics Technician
S. Sanders, Health Physics General Supervisor
P. Suter, Designated Lead Pont of Contact for Mid-Loop Evolution
J. York, Health Physics Support Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

50-317,318/2004-05-03	URI	Review Of Updated CC HX Design Basis Analysis. (Section 1R15)

Opened and Closed

50-317/2004-05-01	NCV	Failure to Assess and Manage Risk Associated With Unit 1 RPS Power Supply Replacement Activities During Reduced Inventory (Section 1R13)
50-317/2004-05-02	NCV	Failure To Implement Design Control Measures For The 12 CC HX (Section 1R15)

50-317, 318/2004-05-04	NCV	Inadequate Corrective Actions For 480v Breaker Testing Deficiency. (Section 4OA2)
50-317/2004-05-05	NCV	Failure to Implement Planned, Scheduled Maintenance (Section 4OA2)

TI 2515/150, Revision 2 - REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATION NOZZLES (NRC Order EA-03-009) Penetration Nozzles - Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel with certification to the ASME, Section XI, Level II and Level III for visual examiners. In addition, Level II and Level III examiners had received training for this type of inspection. The training included a review of industry experience, lessons learned, prior inspection results and procedural 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 training and experience on reactor head penetration (RHP) examination techniques and analysis.

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.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies.

a.4. The examination performed was capable of identifying the primary water stress corrosion phenomena described in Order EA-03-009.

b. 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 difficulty in removal of the insulation panels and ALARA considerations, all nozzle penetrations, including the vent line, were visually examined for a full 360-degree view using a high resolution video probe snorkel that was manually inserted and manipulated from the top of the mirror finish insulation panels. The remote video snorkel was also used to perform the VT of the bare metal of the vessel head. The inspectors verified that the visual inspection was accomplished using this method.

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

d. No material deficiencies were identified with the exception of minor scratches on the outside diameter of two penetrations. The indications were identified on IR4-031-631 and dispositioned as “accept as is” within the licensee’s CAP. The inspectors viewed the indications and reviewed the disposition of IR4-031-631.

e. There were no impediments to an effective VT of the vessel head. There were three mechanically pressed, equally spaced pads on the outside diameter of the vessel head penetrations thermal liner and a counter bore step which interfered with the ultrasonic examination of a full two inches above the highest point of the root of the J-groove weld on some penetrations (refer to licensee relaxation request dated January 30, 2004).

f. The basis for the temperature used (593.7° F) in the susceptibility ranking calculation was an analysis documented in Combustion Engineering (CE) 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 head temperature and the effective full power years are the only inputs to the susceptibility ranking calculation. 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.

g. No indications were identified during the non-visual examinations.

h. Procedures were available to identify potential boric acid leaks from pressure-retaining components above the RPV head.

i. The licensee performed appropriate examinations for indications of boric acid leaks from pressure-retaining components above the RPV head.

LIST OF DOCUMENTS REVIEWED

<u>Section 1R01: Adverse Weather Protection</u>

<u>Tornado Watch Preparations & Tornado Watch Event</u>

ERPIP 3.0 Attachment 20, Severe Weather
ERPIP 3.0 Attachment 21, Personnel Recall for Severe Weather
Operations Administrative Policy OAP 00-01, Severe Weather Operations
SA200300121 - EP Unit Self Assessment; Severe Weather

Section 1R04: Equipment Alignment

11 Charging Pump During Pump Repacking Activity

Clearance ID#: 1200400660

11/12 Component Cooling Water system Heat Exchangers (Compensatory alignment following Unit 1 tube rupture event)

Clearance ID#: 1200400554

Unit 1 12 Service Water Header

Clearance ID#: 1200400702
Drawing # 60708SH0003 Circulating Salt Water Cooling System
Drawing # 60706SH0002 SRW Cooling System Auxiliary Building and Containment

Unit 1 12B Service Water Heat Exchanger

Clearance ID#: 1200400702
Drawing # 61080SH0020S 12A/12B Service Water HTEX Strainers Control Panel 1C201
Drawing # 60712SH0007 Compressed Air System Instrument Air and Plant Air

Section 1R05: Fire Protection

Manual:

SA-1-101, Fire Fighting
Fire Fighting Strategies Manual

Section 1R08: Inservice Inspection (ISI) Activities

IR4-028-633 Indication in Component Support 16-FW-1218-R3
IR4-030-055 #12 CC Water HX Tube Leak
IR4-000-256 Removal of Brackets from Containment Liner in Fuel Transfer Tube Pit
IR4-000-255 Perform Liner Inspection of Fuel Transfer Pit Location
IR4-009-374 Bolt Hole Discrepancies, System 11, SRW Support, ES200300178
IR4-002-336 SRW Cooling, Rod Hanger Support 13600-6862, ES200300221

Examination Procedures:

NDE-5419-CC Ultrasonic Examination of Vessels Greater Than 2 Inches in Thickness
STP-M-574B-1 Eddy Current Exam of CCNPP Unit 1 Steam Generator #12
Steam Generator Eddy Current Analysis Guidelines
54-ISI-400-12 Multi-Frequency Eddy Current Examination of Tubing
Screening Guidelines for In-Situ Tube Inspection Criteria
ISI Program Plan ASME Section XI, IWE/IWL (Class MC and Metallic Liners of CC Components)
Steam Generator Degradation Assessment, Unit 1

NDE-5110-CC Dry Powder AC Yoke MT Examination of Nuclear Components and Welds
Eddy Current Examination Technique Specification Sheet (ETSS) #1, Procedure 54-ISI-400-12
Eddy Current Examination Procedure Qualification (54-PQ-400-31-12) for Procedure 54-ISI-400-12

Examination Reports

CC04-IM-008 MT Report Lug Attachment to Main Feedwater Hanger, 16-FW-1218-R3
VT 2002BV155 Examination for Leakage at 1CKCVC-186 and Piping, Welds H1, H2 and H3.
VT 2003BV104, 105 and 106 Containment Liner Attachment Areas
Liquid Penetrant Examination 2002BP235, 236, 141, 285 and 286 Examination of Check Valve 1-CKVCVC-186, Weld W-1, 2, 3, 2R1and 3R1.
Radiographic Examination 2002BR005A, 006A, 006B,006C and 006D (Pipe Welds), Weld-1, 2, 3, and 2R1and 3R1 (40 film strips total)
MT Examination CC04-IM-008 Integral Attachment to 16-FW-1218-R-3
MT Examination 2003BM110 Exam of Containment Liner Removal Area
MT Examination 2003BM111 Exam of Liner in Fuel Transfer Tube Pit
Ultrasonic Examination CC04-IU-014, Nozzle to Pressurizer Bottom Head Weld, C69-PZR
Ultrasonic Calibration CC04-ICA-023, 027, 028 and 029 for Component C69-PZR
2002BR005 Radiographic Exam Summary Report Weld-1, System 041
2002BR006 Radiographic Exam Summary Report Weld 2, 3, 2R1 and 3R1, System 41
2003BU049 Ultrasonic Thickness Measurement Report Liner Thickness Measurements in Area of Attachment Removal

Work Orders

MO1200003552 Work Order for The Installation of Saltwater Pump Into #11 Cavity-Replacement Activity
MO1200002340 Work Order for Replacement of Check Valve 1CKVCV186, System 41, Chemical and Volume Control System
MO2200301118 Removal of Two Brackets and One Channel from Containment Liner

Welding Procedures

Welding Procedure Specification P8-T, P8 to P8, GTAW
Welding Procedure Specification P8-T/LH, P8 to P8, GTAW/SMAW
Welding Procedure Qualification Records #1, 2, 3A, 4, 11, 12, 15, 16, 48, 72 and 107

Miscellaneous

Welder Performance Qualification Reports for Welders 42363 and 42528
NDE Personnel Certifications (Eddy Current and Ultrasonic), Level II and Level III
Engineering Evaluation ES200000318 Corrosion of Unit 1 Containment Liner
Engineering Evaluation ES199900382 Corrosion of Unit 2 Containment Liner
Engineering Evaluation ES200300128 Disposition of Containment Liner Plate Wall Thickness

Section 1R11: Licensing Operator Requalification Program

Procedures

ERPIP 3.0, Immediate Actions

Other

Simulator Operator Examination For The Licensed Operator Training Program At The Calvert Cliffs Nuclear Power Plant, approved on May 28, 2004.

Section 1R12: Maintenance Effectiveness

Unit 1 11 AFW Forced Oil System Modification Boring Error

IR4-032-002 While machining the 11 AFW pump lower turbine casing a new hole was bored oversized
MO# 1200102925 - AFW Pump Testing
ESP ES200100565 - Forced Feed Lubrication System for 11 AFW Pump

Unit 1 12 CC HX Tube Failures

IR4-030-055 - 12 CC HX Tube Failure
IR4-029-536 - 21/22 CC HX's may be susceptible to same failure as 12 CC HX
AIT IR200400254
Category I Root Causal Analysis - #12 CC HX Tube Failures Due to High CC Flow Rates
OI 16 CC System

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Unit 1 'B' RPS Channel Power Supply Replacement

IR4-030-018, Powered up B RPS channel with resulting ground
IR4-030-584, Found wiring improperly landed
DWG. 60710, SH0001, SH0002, CC System
DWG. 60731, SH0001, SH0002, SH0003, Safety Injection & Containment Spray Systems
DWG. 61022 SH0001, Single Line Diagram 120V AC Vital System
NO-1-103, Conduct Of Lower Mode Operations
MO 1200204155, Replace channel B WRNI RMSP power supplies
MO 1200401550, Troubleshoot/repair ground on B RPS channel
Maintenance Rule Scoping Document, system 058A, Reactor Protective
Calvert Cliffs Nuclear Power Plant Preventive Maintenance Basis, PM Basis 463, System 18, Vital Instrument AC Fuse and Fuse Holders, approved September 22, 1999
ES-013, 1Y0213 Loss of Power Effect/Load List Engineering Standard Manual - Appendix C

12 Turbine AFW Pump Trip and Throttle Valve

MO 1200401980, Replace friction washer and split washer on 1CV3988, 12 AFW Turbine Trip and Throttle Valve
MO 1200400364, Perform STP-0-005A-1, AFW system quarterly surveillance test

12B Service Water Heat Exchanger/Header Maintenance

MO 1200400259, Replace elastomers in 1CV5159, SW strainer diverter valve, actuator
MO 1200400193, Perform STP M-031-0 on relief valve 1-RV-5212
MO 1200400173, Perform inspection of 12B basket strainer 1BS5159
MO 1200400273, Calibrate 12B SRW HX salt water flow and D/P instruments
MO 1200303532, Replace the handswitch for 1-HS-1592 which controls 1-CV-1592

22 Salt Water Pump Discharge Check Valve Replacement

QSS Week 0424 Risk Evaluation rev 7
QSS Software Converter Program
Shift Turnover Sheet for June 16, 2004

1A Emergency Diesel Generator

STP O-8A-1 Test of 1A DG and 11 4KV Bus LOCI Sequencer
MO 1200303083 Perform a 15 minute run of the Diesel Room Supply Fans
MO 1200303108 Perform a Breaker Operational test on BKP 52-1103/11-17 4KV Bus
MO 1200104791 Perform a Slow or Fast Speed Start on 1A DG
MO 1200303099 Inspect Dampers
MO 1200302779 Replace 1A2 DG Engine Driven Fuel Oil Pump

Section 1R14: Operator Performance During Non-Routine Evolutions and Events

Generator Temperature Monitoring System (GTMS) Indication Failure and U-1 Emergent Power Reduction

AOP-7E, Main Turbine Malfunction
OP-3, Normal Power Operation
Unit 1 Reactor Operator Log

12 Component Cooling Water Heat Exchanger Tube Failure

AOP 7C, Loss of CC Water
AOP 7C, Loss of CC Water Basis Document
Unit 1 Reactor Operator Log

Section 1R15: Operability Evaluations

22 Vital Battery Cracked Post Seal Rings

IR3-048-632, Cracked post seal rings on battery 22
IR4-002-241, 125V battery technical review has not been performed on post seal nut cracking
MO 2200003755, Replace battery 22 post seal rings on cells 40, 7, and 10
GNB Industrial Battery Company Specification, Attachment #1, ES 200100611-000

21/22 Component Cooling Water Heat Exchangers

IR4-029-536 - 21/22 CC HX's may be susceptible to same failure as 12 CC HX

13 Charging Pump Degraded Backup Selector Switch

RECO for 13 charging pump degraded condition, dated June 17, 2004
IR4-033-248, 13 charging pump electrically cycled
MO 1200402737, 13-charging pump inadvertently started for less than 1 second
DWG. 61-075-C SH 23C, Schematic Diagram Charging Pump - 13
DWG. 61075 SH000336B, Schematic Diagram Pressurizer Lvl Channel L-110

11 Component Cooling Water Heat Exchanger

Operability Determination Dated 4/30/04 - CCW Heat Exchangers 11, 12, 21, and 22 CCW HX
OI -29 Section 6.7 Shutdown of 11 Salt Water Header
IR4-030-512 - Evaluation of tube failures

12 Component Cooling Water Heat Exchanger

Failure Analysis of No. 12 CCW Heat Exchanger Tubes
Operability Determination Dated 4/30/04 - CCW Heat Exchangers 11, 12, 21, and 22 CCW HX
Cat I IR 12 CC HX Tube Failure
IR4-030-055 12 CC HX Tube Failure
IR4-031-212 An evalution of 11 CC HX was performed to assess susceptibility to tube failure
IR4-029-536 21/22 CC HX may be susceptible to tube failure

Section 1R17: Permanent Plant Modifications

11 AFW Pump Forced Oil System Modification

MO 1200102925 Disconnection of Instrumentation and Tagging of Leads
ESP ES200100565 Forced Feed Lubrication System for 11 AFW Pump
IR4-032-002 While machining #11 AFW pump lower turbine casing a new hole was bored oversized

Section 1R19: Post-Maintenance Testing

12 Control Room HVAC Maintenance

MO 0200302231, Inspect/replace belts for 12 CR room HVAC supply fan
MO 0200302228, Inspect belts, lubricate #12 C/R HVAC condenser fans A & B
MO 0200302207, Inspect 12 control room HVAC breaker and controls, inspect local control panel for age degradation
MO 0200302229, Inspect/replace #12 control room AC supply fan filters
Functional operational 15 minute test in accordance with MO 0200302228

11 Charging Pump Repacking

STP O-73D-1, Charging Pump Performance Test
MO 1200401379, Replace 1HVCVC-1057, 11 charging pump vent
MO 1199604281, Belzona coat 11 charging pump box plunger internals
MO 1200402307, Repack the 11 charging pump

Channel D RPS T_{hot} Input Reading Low

IR4-033-630, T_{hot} input low for Channel D RPS
STP M-212D-1, RPS Channel D Test
MO 1200402328, T_{hot} on Channel D RPS is reading low

12 CC HX Tube Repair Activities

Operability Determination Dated 4/30/04 - CCW Heat Exchangers 11, 12, 21, and 22 CCW HX
Cat I IR 12 CC HX Tube Failure
IR4-030-055 12 CC HX Tube Failure
MO 1200401591, 12 CC HX Tube Removal and Plugging

1A EDG Maintenance Activities

STP O-8A-1 Test of 1A DG and 11 4KV Bus LOCI Sequencer
MO 1200202613 Perform 1A1/1A2 Lube Oil HX Inspections
MO 1200402842 Perform PMT on 1A Generator Bearings
MO 1200303094 Replace the Seals and Bearings in 1A1/1A2
MO 1200303083 Inspect/Lube 1A EDG Supply Fans

Section 1R20: Refueling and Other Outage Activities

Fuel Handling/Refueling Activities

Fuel Handling Procedure 305 (FH-305) Core Alterations
Attachment 6 to FH-305 Boron Concentration Sheet
Fuel Move Log
EN-1-311, Rev 5, Special Nuclear Material Movement and Tracking
OI-25A, Spent Fuel Handling Machine
ESP 200200274, Supplement 018, Rev 1, for crediting CEA's during an incore shuffle

MO 1200203390 Clean and Inspect Reactor Vessel Studs
MO 1200301872 Setup from ICI removal
MO 1200203390 Reactor Vessel Fuel Shuffle
MO 1200301872 Integrated Prejob brief for ICI wire removals
MO 1200301872 Remove ICI's
MO 1200203390 Decon, Clean and Inspect Flanges

Section 2OS1: Access Control To Radiologically Significant Areas

Procedure NO-1-117, Rev. 11, Integrated risk management
Procedure RP-1-100, Revs. 6 and 8, Radiation protection
Procedure RSP-1-101, Rev. 22, Routine radiological surveys
Procedure RSP-1-104, Rev. 17, Area posting and barricading
Procedure RSP-1-106, Rev. 9, Special work permit administration
Procedure RSP-1-132, Rev. 8, Job coverage in radiologically controlled areas
SWP 2004-1002, Rev. 00, ISI/NDE activities for the Unit 1 RFO,
Activity 1 - medium risk - perform, Activity 2 - medium risk -verify, Activity 3 - low risk - support
SWP 2004-1008, Rev. 00, Inspections/FME to support Unit 1 RFO, Activity 1 - medium risk
SWP 2004-1306, Rev. 00, Remove and replace ICI flanges, Activity 1 - medium risk
SWP 2004-1324, Rev. 01, Volumetric inspections of the reactor head, Activity 1 - high risk
SWP 2004-1325, Rev. 01, Reactor head visual inspections (external of head), Activity 1- high risk
SWP 2004-1408, Rev. 00, Installation and removal of steam generator nozzle dams, Activity 1 - high risk
SWP 2004-1409, Rev. 00, Steam generator eddy current, in-situ, sleeving, and tube plugging, Activity 1 - high risk
Radiation survey of Unit 1 reactor head in head lay down area on February 26, 2002 (with shroud in place)
Radiation survey of Unit 1 reactor head in head lay down area on February 26, 2002 (with shroud on stanchion poles and ICI flange shield plugs installed)
Radiation surveys of Unit 1steam generator (12) cold leg and hot leg bowls on April 14, 2004
Unit 1 RFO health physics high impact team (2004)
Health physics contingency plans for Unit 1 2004 RFO
Internal contamination uptake, retention, and dose commitment assessment calculations for year-to-date
DAW analysis for 2003 HP DAW scaling factors
Self-assessment, SA 200200149, Evaluate and identify areas of improvement for RP knowledge
and skill at CCNPP, March 23, 2004
Benchmarking report, Emergency preparedness dose assessment, June 8, 2004
Trend analysis of Issue Reports for RP for 1st quarter of 2004

Section 2OS2: ALARA Planning and Controls:

Procedure PHP-3-301, Rev. 2, Plant health physicist instructions for internal dose calculations
Procedure RP-1-100, Rev. 8, Radiation protection
Procedure RP-1-101, Rev. 3, ALARA
Procedure RSP-1-106, Rev. 9, Special work permit administration
Procedure RSP-1-200, Rev. 21, ALARA planning and SWP preparation

Listing of Unit 1 RFO SWPs and estimated and actual doses
Pre-outage ALARA review packages for SWPs 2004-1002, 2004-1306, 2004-1324, 2004-1325, 2004-1408, and 2004-1409
Post-outage review of:
ALARA review package for SWP 2004-1000, Rev. 03, Health physics activities for Unit 1 RFO, Activity 1
ALARA review package for SWP 2004-1013, Rev. 00, Remove and replace CVC check valves, Activity 1
ALARA review package for SWP 2004-1306, Rev. 00, Remove and replace ICI flanges, Activity 1 - medium risk
ALARA review package for SWP 2004-1324, Rev. 01, Volumetric inspections of the reactor head, Activity 1 - high risk
ALARA review package for SWP 2004-1325, Rev. 01, Reactor head visual inspections (external of head), Activity 1- high risk ALARA review package for SWP 2004-1400, Rev. 00, Major machinery minor maintenance in RCP bays and turbine building
Unit 1 RFO health physics high impact team plan (2004)
Health physics contingency plans for Unit 1 2004 RFO
Site PI for source term reduction
Review of chemistry and operations from unit off-line through refuel pool flood-up for Unit 1 RFO 2004, May 2004
Draft post outage report for 2004 for health physics
Memorandum dated June 14, 2004, concerning declared pregnant workers in 2004
Memorandum dated June 16, 2004, concerning surveillance dose for declared pregnant workers in 2004

<u>Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment:</u>

Procedure EP-1-306, Rev. 1, Emergency response organization training
Procedure RSP-1-102, Rev. 19, Pre-operational checks of the portable survey instruments
Record for monthly MSA 4500 SCBA inspections for June 2004
Record for monthly regulator accounting for June 2004
Record for breathing air compressor training on June 16, 2004
SCBA user qualification list by department for June 15, 2004
Materials processing qualification matrix report for March 31, 2004
Certificate of attendance at Certified Air Mask Repair Education (CARE) class for MSA BMR air masks
MSA SCBA performance evaluation form (including requirement for removing and replacing air bottle)
Updated Final Safety Analysis Summary for Units 1 and 2, Volume 5, Chapter 11, Waste Processing and Radiation Protection

Section 4OA2: Identification and Resolution of Problems

Semi-Annual Problem Identification and Resolution Review

Self Assessment Report 1st Quarter 2004
Operations PIs
SA# SA200300051 - Assessment of Operations Self-Assessment Program
Nuclear Oversight Surveillance Report 2004-087 - Compensatory Measures
Nuclear Oversight Surveillance Report 2004-090 - Power Supply Reliability Management
Nuclear Oversight Surveillance Report 2004-095 - SSC Evaluations
Self Assessment AI# SA200400073 - EP 2003 Drill Deltas Roll Up
Self Assessment AI# SA200300121 - EP 2003 Severe Weather
Self Assessment AI# SA200000015 - EP 2003 Corrective Actions Program
Maintenance Report Backlog
Maintenance PIs
QL-2, Self-Assessment/Corrective Action Program
QL-2-100, Issue Reporting and Assessment
QL-2-104, Self-Assessment Standard/Benchmarking

480 Volt Breaker Inspection

Drawings
61001SH0001, Rev. 42; Electrical Main Single Line Diagram, FSAR Fig. No. 8-1
61009, Rev. 37; Single Line Meter & Relay Diagram 480v Unit Buses 11A, 11B, 14A & 14B, FSAR Fig. No. 8.3
61010SH0001, Rev. 36; Single Line Meter & Relay Diagrams 480v Unit Buses 12B, 12A, 13A, 13B,15
63009, Rev. 34; Single Line Meter & Relay Diagram 480v Unit Buses 21A, 21B, 24A, 24B, FSAR Fig. No. 8-11
63010, Rev. 39; Single Line Meter & Relay Diagram 480v Unit Buses 22A, 22B, 23A, 23B, 25

Modification Packages
ES200100938-000, Supplement 000, Rev. 000, (1/18/02)
ES200100017-001, Supplement 001, Rev. 000, (2/28/01)
ES199501450, Supplement 000, Rev. 000, (10/19/95)
ES199501406, Supplement 000, Rev. 000, (6/19/95)

Issue Reports

IR1-066-312	IR3-009-033	IR3-074-989	IR4-020-682
IR1-029-247	IR3-037-534	IR3-077-191	IR4-020-810
IR1-027-923	IR3-041-816	IR3-081-251	IR4-020-811
IR1-028-003	IR3-042-098	IR3-081-877	IR4-022-461
IR1-053-644	IR3-044-612	IR4-002-245	IR4-025-422
IR3-003-365	IR3-053-962	IR4-002-246	IR4-027-382
IR3-010-311	IR3-053-663	IR4-008-309	IR4-028-152
IR3-010-312	IR3-059-951	IR4-016-425	IR4-028-163
IR3-037-266	IR3-071-275	IR4-019-025	IR4-028-164
IR3-003-385	IR3-072-536	IR4-019-595*	IR4-028-177
IR3-003-515	IR3-074-771	IR4-020-680	IR4-028-182*

IR4-028-183*
IR4-028-843
IR4-029-151
IR4-029-228
IR4-029-432
IR4-029-585
IR4-030-336*
IR4-031-426

* indicates the IR was generated by this inspection

Work Orders
2200302762, 12/23/03; Inspect 52-2319 (22 CEDM CLG FAN)
1200300097, 7/7/03; Inspect 52-1115 (11 Chg. PP) And Control Circuit Per EPM 05000
1200100381, 4/9/01; 12 Instrument Air Compressor Breaker And Controls Inspection
1199605767, 1/26/98; Bkr. 52-1222
1199605960, 4/1/98; Bkr. 52-1301
2199803672, 11/13/98; Bkr. 52-2430
2200000078, 7/14/00; 22 CNTMT Cooler Breaker and Controls Inspection
2200104355, 6/14/02; Inspect 22 Containment Cooler Motor, Breaker and Local Controller. Calibrate Motor Ammeter and Control Circuit Agastat Relay
2199703849, 6/12/98; Inspect 22 Containment Cooler Motor, Breaker and Local Controller. Calibrate Motor Ammeter and Control Circuit Agastat Relay
2199603361, 11/1/96; Inspect Motor, Controller and Feeder Breaker for 22 Containment Air Cooler
29202461, 5/8/92; 22 CNTMT Cooling Fan Space Htr., Breaker and Control Inspection, PM 2-60-E-RQ2-3
2200100672, 1/15/02; 22 Cavity Cooling Fan Motor Feeder Breaker Inspection
0200002315, 3/30/01; 1MCC125 Feeder Breaker Inspection
2199901443, 10/18/1999; Inspect Motor And Feeder Breaker For 22 Cavity Cooling Fan
2199701160, 11/25/97; Inspect Motor And Feeder Breaker For Cavity Cooling Fan
2199503532, 1/30/96; Inspect Motors And Feeder Breakers For Cavity Cooling Fan And It's Dampers
1200103402, 12/27/01; Inspect MCC-111 Feeder Breaker 52-1427
1199903249, 4/11/00; Inspect MCC-111 Feeder Breaker 52-1427
1199704312, 5/29/98; Inspect MCC-111 Feeder Breaker 52-1427

Maintenance Procedures
CCNPP Technical Procedure FTE-52, Westinghouse DS-416 Circuit Breakers And Cubicle Inspection, Rev. 4, (6/28/02)
CCNPP Technical Procedure, FTE-29, Acceptance Test And Calibration Of Amptectors, Rev. 8, (6/9/04)
CCNPP Technical Procedure FTE-53, Westinghouse DS-206 Circuit Breakers And Cubicle Inspection, Rev. 14, (1/20/00)
CCNPP Technical Procedure E-32, DS-206/416 CIRCUIT BREAKER OVERHAUL PROCEDURE, Rev. O, 9/6/02
CCNPP Technical Procedure, FTE-76, WESTINGHOUSE DS-632 LOW VOLTAGE BREAKER INSPECTION, TESTING, ADJUSTMENT AND CUBICLE CLEANING, Rev. 3, 6/28/02
CCNPP Technical Procedure, FTE-77, WESTINGHOUSE DS-416H CIRCUIT BREAKER AND CUBICLE INSPECTION, Rev. 1, 9/29/00

Procedures
CCNPP Administrative Procedure, ISSUE REPORTING AND ASSESSMENT, OL-2-100, Rev. 18, 3/16/04

CCNPP Administrative Procedure, CAUSAL ANALYSIS, QL-2-101, Rev. 8, 3/16/04
Calvert Cliffs Engineering Standard, ES-034: ELECTRICAL DEVICE SETTINGS, Rev. 00, 10/11/95
CCNPP Administrative Procedure, INTEGRATED RISK MANAGEMENT, NO-1-117, Rev. 12, 5/27/04
CCNPP Administrative Procedure, PLANT HEALTH COMMITTEE, ER-1-101, Rev. 1, 5/15/04

Root Cause Analyses
Category I Root Cause Analysis IR299400111(IR4-002-245)

Miscellaneous Documents
Calvert Cliffs Maintenance Rule Scoping Document, Rev. 22, 1/19/04
RECO for IR4-028-152 CX Relay Operability
Westinghouse NSD-TB-75-2 (2/20/75)
Westinghouse NSD-TB-83-03 (3/24/03)
Westinghouse NSD-TB-84-02, Rev 1. (4/13/84)
Westinghouse NSD-TB-85-17, (8/14/85)
Westinghouse NSD-TB-87-11 (12/1/87)
Westinghouse NSID-TB-88-04 (8/9/88)
Westinghouse NSID-TB-88-05 (8/9/88)
Westinghouse NSID-TB-88-07 (12/16/88)
Westinghouse NSID-TB-91-06-RO (9/24/91)
Westinghouse NSD-TB-92-06-RO (6/12/92)
Westinghouse NSD-TB-93-05-RO (1/10/94)
Westinghouse NSAL-94-024, Rev. 1 (2/15/96)
Westinghouse ESBU-TB-98-02 (7/22/98)
Westinghouse TB-00-01-RO (4/24/00)
Westinghouse TB-01-01 (2/9/01)
Westinghouse TB-04-7 (4/6/04)
Westinghouse TB-04-8 (5/3/04)
Westinghouse Westector Technical Summary
AIT #IR200400220(IR4-017-436(6/13/04)), 3/26/04
Westinghouse I.B. 33-790-1G (12/89); Instructions For Low Voltage Power Circuit Breakers Types DS and DSL
Westinghouse I.B. 33-790-1E (7/81); Low Voltage Power Circuit Breakers Types DS and DSL
Cutler-Hammer Instructions For Low-Voltage Power Circuit Breakers Types DS and DSL; I.B. 33-790-11 (10/89)

Section 4OA5: Other Activities

Associated Documents:

Order Establishing Interim Inspection Requirements for RPV Heads at Pressurized Water Reactors (February 11, 2003)
NRC Bulletin 2002-02, RPV Head And Vessel Head Penetration Nozzle Inspection Programs
NRC Bulletin 2002-01, RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity
Effective Degradation Years, Unit 1, Calculation as Required by NRC Order EA-03-009
Calvert Cliffs Response for Additional Information Regarding Interim Inspection Requirements for RPV Head

Drawings

DWG 6030777D-0 Calvert Cliffs Unit 1 Inspection Nozzle Penetration Map
DWG 5023649E Calvert Cliffs 1&2 CEDM Guide Replacement

Issue Reports

IR4-031-631 External Scratches on CEDM Nozzles

Examination Procedures

54-ISI-367-06 VT for Leakage of Reactor Head Penetrations
54-ISI-100-11 Remote Ultrasonic Examination of Reactor Head Penetrations

LIST OF ACRONYMS

AFW	Auxiliary Feedwater
ALARA	As Low As Is Reasonably Achievable
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CC	Component Cooling
CE	Combustion Engineering
CEDE	Cumulative Effective Dose Equivalent
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CR	Condition Report
DAW	Dry Active Waste
ECT	Eddy Current Testing
ERPIP	Emergency Response Plan Implementation Procedures
GTMS	Generator Temperature Monitoring Systems
HP	Health Physics
HRA	High Radiation Area
HX	Heat Exchanger
ICI	In-Core Instrumentation
IMC	Inspection Manual Chapter
IR	Issue Report
ISI	In-Service Inspection
LHRA	Locked High Radiation Area
MT	Magnetic Particle
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
OAP	Operations Administrative Policy
PI	Performance Indicator
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RHP	Reactor Head Penetration
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SCBA	Self-Contained-Breathing Apparatus
SDC	Shutdown Cooling
SDP	Significance Determination Process
SIAS	Safety Injection Actuation Signal
SRW	Service Water
SSC	Systems, Structures, and Components
STP	Surveillance Test Procedure
SWP	Special Work Permit
UT	Ultrasonic Examination
VHP	Vessel Head Penetration
VHRA	Very High Radiation Area

VT	Visual Examination
UFSAR	Updated Final Safety Analysis Review