May 3, 2002

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Constellation Nuclear
Calvert Cliffs Nuclear Power Plant, Inc.
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

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

Dear Mr. Cruse:

On March 30, 2002, the NRC completed an inspection at your Calvert Cliffs Nuclear Power Plant Units 1 & 2. The enclosed report documents the inspection findings which were discussed on April 24, 2002, with Mr. Katz and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of this inspection, the inspectors identified one finding of very low safety significance (Green). This issue did not involve a violation of NRC requirements.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate Calvert Cliffs Nuclear Power Plant, Inc. compliance with these interim requirements.

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

Sincerely,

/RA/

Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 50-317/02-02 and 50-318/02-02

Attachments: (1) Supplemental Information
(2) NRC Inspection Results for Section 5 of Temporary Instruction 2515/145, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

cc w/encl: M. Geckle, Director, Nuclear Regulatory Matters (CCNPPI)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-317, 50-318
License Nos.:	DPR-53, DPR-69
Report Nos.:	50-317/02-02 50-318/02-02
Licensee:	Calvert Cliffs Nuclear Power Plant, Inc.
Facility:	Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Location:	1650 Calvert Cliffs Parkway Lusby, MD 20657-4702
Dates:	February 17, 2002 - March 30, 2002
Inspectors:	David Beaulieu, Senior Resident Inspector Leonard Cline, Resident Inspector Ronald Nimitz, Senior Health Physicist E. Harold Gray, Senior Reactor Inspector Kenneth Kolaczyk, Reactor Inspector Thomas Burns, Reactor Inspector
Approved by:	Michele G. Evans, Chief Projects Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000317-01-02, IR 05000318-01-02, on 2/17-3/30/2002, Calvert Cliffs Nuclear Plant, Inc.; Calvert Cliffs Nuclear Power Plant, Units 1 & 2. Operability Evaluations.

The inspection was conducted by resident inspectors, a senior health physicist, and regional specialist inspectors. The inspection identified one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/reactors/operating/oversight.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified that the Updated Safety Analysis Report, Section 14.6, analysis for a loss of main feedwater did not assume a single failure of the auxiliary feedwater (AFW) system as specified in NUREG 0737, Three Mile Island Action Item II.E.1.1.

  This finding was of very low safety significance because the licensee's re-analysis demonstrated acceptable results by crediting operator action to increase AFW flow from the operating AFW pump. (Section 1R15.2)

B. Licensee Identified Findings

None

Report Details

Unit 1 was shutdown for a refueling outage and steam generator replacement for the entire inspection period. Unit 2 operated at or near 100 percent power for the entire inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

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

- 1B emergency diesel generator (EDG) was inspected on March 25, 2002, while the 1A EDG was out of service for maintenance.

The inspectors reviewed Calvert Cliffs Nuclear Power Plant Procedure OI-21B-1, “1B Diesel Generator,” to support the walkdown of the EDG.

b. Findings

No findings of significance were identified.

1R05 Fire Protection - Fire Area Tours

a. Inspection Scope

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

The areas inspected included:

- 1A emergency diesel generator room.
- 1B emergency diesel generator room.
- 2A emergency diesel generator room.

b. Findings

No findings of significance were identified.

1R08 In-Service Inspection

.1 Welding and nondestructive examination requirements

a. Inspection Scope

The inspector reviewed samples of nondestructive examination (NDE) and American Society of Mechanical Engineers (ASME) Code, Section XI repair/replacement activities based on the inspection procedure objectives and the risk significance of the subject components and systems. The inspector also reviewed a sample of issue reports initiated as a result of problems identified during the conduct of in-service inspection examinations.

To evaluate the licensee's effectiveness in monitoring degradation of risk significant systems, structures and components, the inspector performed a review of three types of NDE activities, volumetric, surface, and visual examinations. In addition, the inspector evaluated the disposition of non-conforming conditions the licensee had identified and verified that, in accordance with the ASME Code, the licensee had performed analyses for acceptance and continued operation without repair. To verify compliance with ASME Section XI requirements for NDE, the inspector reviewed the ultrasonic and magnetic particle test reports for the Unit 1 reactor pressure vessel head to flange weld and the penetrant test for the valve to pipe weld for safety injection system valve 1-SI-401. The inspector also reviewed the examination and issue reports for the visual inspection of the containment liner (conducted during this outage) for compliance with ASME Section XI requirements for class MC and metallic liners of class CC components.

The inspector reviewed welding activities associated with the replacement of selected components to verify the activities were performed in accordance with the requirements of ASME Section IX and XI. The inspector reviewed licensee weld procedures and personnel qualification records. The inspector conducted interviews with the welding supervisor and personnel responsible for radiographic activities and the management of welding activities. Radiographs of welding activities performed under maintenance order MO 1200001173 were reviewed for field welds 1, 2, 3, 4 and the repair of field weld 2 for the ASME Section XI replacement of main steam piping and components. The review verified flaws were appropriately located, identified, sized, recorded, and evaluated for compliance with the requirements of ASME Section XI. The radiograph review verified that repair welding was satisfactorily completed within the ASME Code requirements.

b. Findings

No findings of significance were identified.

.2 NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."

a. Inspection Scope

The licensee's activities performed in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," were reviewed. The inspection scope and results are described in Section 4OA5 of this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

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

- Unit 1 containment air coolers. The licensee has appropriately classified this system as (a)(1) primarily due to high unavailability caused by frequent cleaning of the service water heat exchangers. The inspector evaluated the acceptability of the licensee's corrective action plan as documented in Issue Report IR3-026-604.

- Unit 2 high pressure safety injection (HPSI) main header motor-operated isolation valve, 2-SI-616-MOV. The inspector evaluated the troubleshooting performed to determine why the valve failed to open and the subsequent performance of maintenance order MO 2200201261, "Replace hand switch 2HS3616 for valve 2-SI-616-MOV at control room panel 2C08." The inspector assessed the licensee's evaluation of this functional failure in accordance with maintenance rule criteria.

- Low pressure safety injection (LPSI) and HPSI pumps. The inspector reviewed design engineering calculation CRMP 203 which provided a basis for not considering the LPSI and HPSI pumps unavailable when the associated room cooler is removed from service.

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

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

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

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

- MO 1199904707, Disassemble, clean, and inspect No. 11 vital ac bus.
- MO 1200101723, Unit 1 No. 13 auxiliary feedwater (AFW) pump. The inspector verified that the licensee appropriately considered Unit 2 risk because the No. 13 AFW pump can be cross-connected to supply Unit 2.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 High Pressure Safety Injection Header Isolation Valve

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant mitigating systems to assess: (1) technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were appropriately addressed with respect to their collective impact on continued safe plant operation; and (4) where compensatory measures were involved, whether the measures were in place, would work as intended, and were appropriately controlled. The following evaluation was reviewed:

- Operability Determination 02-003, No. 21 high pressure safety injection header motor-operated isolation valve, 2-SI-616-MOV.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 50-317 & 318/2001-008-01: The inspector noted that the anticipated operational occurrences (AOOs), evaluated in the Updated Final Safety Analysis Report (UFSAR) Chapter 14, did not assume a single failure of auxiliary feedwater (AFW) or any other system. As an interim measure, the licensee prepared

Operability Determination No. 01-016 which satisfactorily bounded conditions involving a single failure of AFW. This unresolved item was opened to allow further review of the licensing basis regarding whether the licensee is required to assume a single failure.

Chapter 14.1 of the UFSAR describes the classification of transients (including the definition of AOOs), acceptance criteria for each type of transient, assumed protective system actions, and core and system performance requirements. Except for the AFW system which is discussed below, there was no licensing basis document that specifies what single failures are required to be assumed in accident analyses. The practice at the time Calvert Cliffs was licensed was to not assume a single failure for AOOs while having a stringent acceptance criteria for no fuel failures. The inspector notes that later vintage plants were licensed using NUREG 0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” which specifies assuming a single failure. However, the acceptance criterion was that fuel failure was permitted as long as offsite dose consequences were a small percentage of 10CFR100 limits.

The inspector found that the current licensing basis for the AFW system requires the licensee to assume a single failure for the loss of main feedwater analysis even though this transient is an AOO. NUREG 0737, “Clarification of Three Mile Island (TMI) Action Plan Requirements,” Item II.E.1.1(3), required licensees to reevaluate the AFW system flowrate design bases and criteria. NUREG 0737 Item II.E.1.1(3) also states that licensees that required AFW system upgrades (which included Calvert Cliffs) were sent a letter requesting additional information on the AFW system flowrate design basis and criteria. In a letter to Calvert Cliffs, dated November 7, 1979, the NRC stated that the AFW system design basis and criteria were not well documented or defined, so the licensee was asked to provide additional design basis information including a verification that the AFW pumps will supply the necessary flow to the steam generators considering a single failure.

Contrary to this, the inspector noted that the current UFSAR did not assume a single failure for a loss of main feedwater event. In addition, the inspector found that a license amendment request dated December 20, 2000, (which had not yet been approved by the NRC) that requested a technical specification change to support steam generator replacement also did not assume a single failure of AFW for the loss of main feedwater event. As a result, on November 19, 2001, the licensee submitted a license amendment request for the re-analysis of the loss of main feedwater event which addressed several non-conservative assumptions including considering a single failure of AFW. The NRC approved the November 19, 2001, license amendment request on February 26, 2002, thereby dispositioning TMI Action Item II.E.1.1(3). This allowed for NRC approval of the December 20, 2000, licensee amendment request on March 1, 2002.

The failure to assume a single failure of AFW for the UFSAR, Section 14.6, loss of main feedwater event as specified in TMI Action Item II.E.1.1(3) is considered more than minor because it had a credible impact on safety. Notwithstanding, the licensee’s re-analysis demonstrated acceptable results by crediting operator action to increase AFW flow from the operating AFW pump; therefore, this inspector finding was determined to be within the licensee response band (Green). A non-conformance with a TMI Action Item does not constitute a violation of NRC requirements.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- MO 1200101090, Remove, inspect, and reinstall saltwater control valve 1-SW-5156-CV. The retest that was conducted on April 1, 2002, consisted of a timed stroke test with position indication verification in accordance with surveillance test procedure STP-66I-1, “Saltwater Emergency Overboard Valves Cold Shutdown Operability Test,” and a saltwater system flow verification test.

- MO 1200103640, New bench set adjustment for valve 1-CV-4550, the Unit 1-to-Unit 2 auxiliary feedwater cross-connect valve. The retest that was conducted on February 27, 2002, consisted of a timed stroke test, position indication verification, and flow verification testing in accordance with procedure STP O-005A-1, “Auxiliary Feedwater System Quarterly Surveillance Test.”

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the following activities related to the Unit 1 refueling outage for conformance with the applicable procedures, and witnessed selected activities associated with each evolution:

- Defueling operations.
- Shutdown risk evaluations.
- Contingency Plan 02-11 for the 120 Vac vital electrical panel, 1Y01, maintenance outage.
- MO 1199904707, Disassemble, clean and inspect No. 11 vital ac bus.
- MO 1200005040, 1A1 diesel engine pistons A2 and B4 have small cracks.
- Contingency Plan 02-07 for the 1A diesel generator outage.
- MO 1200201026, Remove and inspect cylinder heads on 1A1 and 1A2 diesel generators.

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

- IR3-080-066, Found cracked cylinder heads in 1A diesel generator during inspections.
- IR3-075-135, Reactor coolant system level lowered from 43.1 feet to 41.65 feet when the reactor head vent flange was broken.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk-significant systems, structures, and components (SSCs) to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report, Technical Requirements Manual, and licensee procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- STP O-66I-1, Saltwater Emergency Overboard Valves Cold Shutdown Operability Test.
- STP O-005A-1, Auxiliary Feedwater System Quarterly Surveillance Test.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification No. 1-02-003, which installed a temporary hose to cross-connect plant air and instrument air in Unit 1 containment, to assess: (1) the adequacy of the 10 CFR 50.59 evaluation; (2) that the installations were consistent with the modification documentation; (3) that drawings and procedures were updated as applicable; and (4) the adequacy of the post-installation testing.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector conducted the below listed activities and reviewed the associated documents to determine the effectiveness of radiologically controls, including access controls to radiologically significant areas. The inspector's review focused on areas and activities with higher radiological risk.

The inspector toured portions of Unit 1 and Unit 2 Auxiliary Buildings and the Unit 1 Containment Building and reviewed access controls to high radiation areas. High radiation area access points were physically inspected to determine if posting and barricading were adequate and to determine if access controls were sufficient to preclude unauthorized entry.

The inspector reviewed radiologically controlled area (RCA) access and egress controls including personnel contamination monitoring practices during RCA egress.

The inspector reviewed the radiological controls provided and accrued occupational radiation doses received by workers who conducted: Unit 1 reactor cavity decontamination activities; removal and installation of the Unit 1 reactor vessel head; Unit 1 reactor coolant pump seal activities; and Unit 1 incore instrument removal during the current Unit 1 outage. The inspector reviewed radiological surveys (i.e., airborne radioactivity, beta and gamma radiation dose rates, and loose surface contamination) associated with the work, and reviewed conformance with applicable special radiation work permits. The inspector also reviewed the fractional mix of radionuclides present to determine the adequacy of internal exposure controls.

The inspector reviewed occupation exposures for workers to determine: if workers received unplanned internal or external exposures; if doses were consistent with initial estimates; and if applicable radiological controls were adequate for conditions present. The inspector also reviewed the licensee's detection and monitoring practices for dose rate gradients that could produce non-uniform occupational exposures.

The inspector reviewed implementation of new respiratory protection equipment, including training provided to workers.

The reviews in this area were against criteria contained in 10 CFR 19, 10 CFR 20, Technical Specifications, applicable site radiation procedures, and the Radiation Safety Section Outage Plan for the Unit 1 2002 outage.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

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

- The inspector reviewed the licensee's performance in the area of occupational exposure reduction for the Unit 1 outage through March 15, 2002. As part of this review, the inspector attended the March 13, 2002, outage planning and status meeting. The inspector reviewed occupational exposures sustained relative to applicable goals. Tasks reviewed included: radiation safety coverage; maintenance activities; reactor assembly and disassembly; and vessel head penetration work. Also reviewed were the results achieved for the licensee's shutdown chemistry control, including enhanced flow rates for cleanup of reactor coolant letdown.

- The inspector toured the Unit 1 containment, observed ongoing work activities and interviewed workers to ascertain if worker and radiation protection personnel performance were consistent with occupational reduction practices specified in applicable ALARA reviews.

The review was against criteria contained in 10 CFR 19, 10 CFR 20, Technical Specifications, and applicable site procedures, including the Radiation Safety 2002 ALARA Pre-Outage Report.

b. Findings

No findings of significance were identified.

2SO3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector selectively reviewed elements of the radiation monitoring instrumentation calibration program to evaluate the adequacy and effectiveness of the program. Specifically, the inspector reviewed the use, calibration, and source checking of radiation survey and monitoring instruments used for risk significant radiological work tasks. The instruments reviewed were: Air sampler (SN. No. 5267), RO-2A (SN. No. 2576), E-600 (SN. No. 1133/280), Teletector (SN. No. 106911), and Electronic Dosimetry (SN. Nos. 6372 and 6161).

The review was against applicable station procedures and 10 CFR 20.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstones: Mitigating Systems

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed data for the High Pressure Safety Injection System Unavailability performance indicator for Units 1 and 2. This inspection examined data and plant records through the first quarter of 2002, including Performance Indicator Data Summary Reports, Licensee Event Reports, operator narrative logs, and maintenance rule records.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Temporary Instruction 2515/145 - Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

a. Inspection Scope

In accordance with Temporary Instruction (TI) 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” the inspector reviewed the licensee’s activities to detect circumferential cracking of reactor pressure vessel head penetration nozzles. This included interviews with responsible personnel, reviews of qualification records and procedures, observations of visual inspections in-progress and review of selected video tape records of the reactor vessel closure head penetration visual examination. The inspector independently viewed a sample of 22 out of the total 74 penetrations examined by the plant staff. The NRC inspection results for the items listed in Section 5 of TI 2515/145 are documented in Attachment 2 of this report.

b. Findings

No findings of significance were identified.

.2 Steam Generator Replacement Project

a. Inspection Scope

The inspector reviewed radiological controls for ongoing steam generator replacement work activities. Licensee activities were evaluated using the criteria contained in 10 CFR 19, 10 CFR 20, Technical Specifications, and applicable site and project procedures, including the Steam Generator Replacement Radiation Protection Plan. The following matters were reviewed:

- Current occupational exposure reduction performance relative to goals.
- Current project schedule and ALARA planning for steam generator girth cutting and primary steam generator pipe end decontamination activities.
- Radiological safety plans for transport and storage of the steam generators, including radiological environmental monitoring.
- Surveillance and planned audits of work activities (Vendor Assessment Audits QAO-01-150 and QAO-02-015, dated December 19, 2001, and February 12, 2002, respectively, and Unit 1 Outage Weekly Oversight Reports through March 13, 2002).
- The inspector attended the March 13, 2002, integrated high radiological risk briefing for cutting the steam generators.

This inspection included a review of welding plans, procedures, controls, and the welder training/qualification process. The preparations and procedures for nondestructive examination of piping and cone-to-cone circumferential welds were reviewed, as well as, the procedure for control of weld filler material. The shell cone weld joint preparation on the replacement steam generator assembly was examined. The results of dye penetrant and helium leak testing of the tube-to-tubesheet welds on the replacement steam generators were reviewed. To ensure weld filler material was being controlled in accordance with procedures, the inspectors interviewed personnel, and reviewed weld material records.

Several safety analyses and work packages prepared to support steam generator replacement activities, were reviewed by the inspectors to determine if the proposed activities required NRC review and approval before implementation. The inspectors also compared the safety analysis associated with the replacement steam generators to licensed operator lesson plans to ensure operators were adequately trained regarding expected changes in plant performance following installation of the replacement steam generators. Finally, the inspectors reviewed the licensee's disposition of several non-conformance reports associated with the steam generator replacement project.

b. Findings

No findings of significance were identified.

.3 Institute of Nuclear Power Operations Plant Assessment Report

The inspectors reviewed the Institute of Nuclear Power Operations final report documenting their December 2000 plant evaluation of Calvert Cliffs Nuclear Power Plant.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on April 24, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

a. Key Points of Contact

C. Cruse, Vice President
M. Geckle, Director, Nuclear Regulatory Matters
M. Haney, Radiation Protection Supervisor
D. Holm, Superintendent, Nuclear Operations
D. Jordan, Radioactive Waste Supervisor
P. Katz, Plant General Manager
T. Kirkham, Radiation Protection Supervisor
M. Korsnick, Superintendent, Work Management
K. Nietmann, Manager, Nuclear Performance Assessment Department
T. Pritchett, Manager, Nuclear Engineering Department
S. Sanders, General Supervisor Radiation Safety
J. Spina, Superintendent, Nuclear Maintenance
R. Szoch, General Supervisor, Plant Engineering
L. Weckbaugh, Manager, Nuclear Support Services
R. Wyvill, ALARA Supervisor

b. List of Items Opened, Closed, or Discussed

Closed

50-317; 318/2001-008-01	URI	UFSAR Single Failure Assumptions (Section 1R15.2)

c. List of Documents Reviewed

Steam Generator Replacement Project

Welder Status report dated 2/26/02
Welding Procedure Specification GT/3.3-1, Rev. 0
QEP 18.1, “Quality Assurance Audits,” dated 3/14/01
QEP 20.1, “Control of Welding,” dated 7/16/01
QEP 20.2, “Welding Procedure Qualification,” dated 8/14/01
QEP 20.3, “General Welding Requirements for American Society of Mechanical Engineers,” dated 12/30/01
QEP 20.4, “Welder Performance Qualification,” dated 11/30/01
QEP 20.5, “Control of Weld Filler Material,” dated 1/30/02
QEP 20.6, “Control of Preheat and Posted Heat Treatment,” dated 5/25/01
CCNPP Replacement Steam Generator Drawing - Weld Map, Small Bore Non-destructive examination, Rev 1
CCNPP Replacement Steam Generator Drawing - Weld Map, Large Bore Non-destructive examination, Rev 1
SGRP-UT-2, “Ultrasonic Test of Class 1 and 2 Vessel Welds,” Rev. 0
SGRP-UT-3, “Ultrasonic Test of Ferritic Piping Welds,” Rev. 0
ES199601526-100, Replacement of steam generators
ES199601526-101, Temporary lifting device

ES199601526-102, Hatch transfer system
ES199601526-103, Installation of temporary platforms
ES199601526-104, Unit 1 reactor cavity decking
ES199601526-105, Miscellaneous rigging and handling
ES199601526-106, Unit 1 reactor coolant system temporary supports
ES199601526-109, Unit 1 feedwater
ES199601526-111, New steam generator recirculation piping
ES199601526-116, Unit 1 replacement steam generator modification
ES199601526-119, Unit 1 component cooling water
ES199601526-120, Unit 1 miscellaneous interferences
NCR101, Rebar clearance violation
NCR126, Weld oven temperature and operation is inadequate
NCR130, Welder not certified for two inch pipe weld

Temporary Instruction 2515/145 - Reactor Pressure Vessel Head Penetration Nozzles

- Procedure 54-ISI-367-03, "Visual Test for Leakage of Reactor Head Penetrations," Rev. 3, dated 11/15/01.
- Letter, C. Cruse to NRC dated 9/4/01, Response to NRC Bulletin 2001-001, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."
- Letter, C. Cruse to NRC dated 2/7/02, Alternate Techniques for Reactor Vessel Head Repair.
- EPRI Report 1006284, (MRP-48), on NRC Bulletin 2001-001, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."

d. List of Acronyms

AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DBT	Design Basis Threat
EDG	Emergency Diesel Generator
HPSI	High Pressure Safety Injection
IR	Issue Report
LPSI	Low Pressure Safety Injection
MO	Maintenance Order
NDE	Nondestructive examination
NRC	Nuclear Regulatory Commission
RCA	Radiological Controlled Area
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
TI	Temporary Instruction
TMI	Three Mile Island
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 2

NRC Inspection Results for Section 5 of Temporary Instruction 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.”

a.1. The examination was performed by qualified and knowledgeable personnel. Although the visual examination performed was to determine leakage, the inspectors found that the plant staff invoked the additional requirements of a “qualified” visual examination for personnel, equipment and technique, as described in Temporary Instruction 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.”

a.2. The visual examination was in accordance with approved and adequate procedures.

a.3. The examination of the 65 control element drive mechanism penetrations, eight incore instrumentation penetrations, and the head vent line connections were adequate to identify, disposition, and resolve deficiencies.

a.4. The examination performed was capable of identifying the primary water stress corrosion cracking phenomenon described in the NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.”

b. The general condition of the reactor vessel head was mostly clean bare metal with some localized staining and minor debris. The insulation design and configuration did not provide easy access for examination; however, the visual obstructions were overcome by the use of a video probe delivered through guide tubes and manipulation of the insulation. There were two in-core instrumentation penetrations where borated water from prior flange leaks left some deposits. These deposits were removed and the areas were re-examined. The videotape inspection showed no boron deposits that were considered to result from leakage through the control element drive mechanisms.

c. Small boron deposits, as described in NRC Bulletin 2001-01, could be identified and characterized by the visual examination technique used. None were found during this visual inspection.

d. No material deficiencies associated with concerns in NRC Bulletin 2001-01 were found.

e. The ALARA radiation exposure controls for the visual examination process were effective. The total dose received was approximately 2 person-rem, which was under the projected estimate of 3 person-rem. (These values do not include exposure associated with re-insulation.)

Temporary Instruction 2515/145, Section 04.04c, requires that inspectors report lower-level issues concerning data collection and analysis, and issues deemed to be significant to the phenomenon described in NRC Bulletin 2001-01. No lower-level issues were identified by the inspector.