Peter E. Katz
Vice President
Calvert Cliffs Nuclear Power Plant
Constellation Generation Group, LLC

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 495-4455
410 495-3500 Fax

Constellation Energy Group

January 16, 2003

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to Request for Additional Information Regarding Calvert Cliffs Nuclear Power Plant's 60-Day Response to NRC Bulletin 2002-01 (TAC Nos. MB4533 and MB4534)

REFERENCES:

(a) Letter from D. M. Skay (NRC) to P. E. Katz (CCNPP), dated November 19, 2002, Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60–Day Response for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Request for Additional Information (TAC Nos. MB4533 and MB4534)

(b) Letter from Mr. C. H. Cruse (CCNPP) to Document Control Desk (NRC), dated May 15, 2002, 60-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

(c) NRC Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, dated March 18, 2002

The purpose of this letter is to provide Calvert Cliffs Nuclear Power Plant, Inc.'s responses to Nuclear Regulatory Commission's request for additional information (Reference a) regarding the CCNPP's 60-day response (Reference b) to Nuclear Regulatory Commission Bulletin 2002-01 (Reference c).

Attachment (1) to this letter provides the requested information.

A095

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Peter E. Katz

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Peter E. Katz, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this response on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Peter E. Katz

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 16 day of January, 2003.

WITNESS my Hand and Notarial Seal:

Michelle D Hall
Notary Public

My Commission Expires:

February 1, 2006
Date

PEK/GT/bjd

Attachment: (1) Response to NRC request for Additional Information Regarding Calvert Cliffs Nuclear Power Plant's 60-Day Response to Bulletin 2002-01

cc: J. Petro, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
D. M. Skay, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING CALVERT CLIFFS NUCLEAR POWER PLANT'S 60-DAY RESPONSE TO NRC BULLETIN 2002-01

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING CALVERT CLIFFS NUCLEAR POWER PLANT'S 60-DAY RESPONSE TO NRC BULLETIN 2002-01

NRC REQUEST 1:

Provide detailed information on, and the technical basis for, the inspection techniques scope, extent of coverage, and the frequency of inspections, personnel qualifications and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection locations where reactor coolant leaks have potential to come in contact with and degrade the subject material (e.g. reactor pressure vessel (RPV) bottom head).

CCNPP RESPONSE:

As stated in Reference (1), the Calvert Cliffs Nuclear Power Plant (CCNPP) Boric Acid Corrosion Inspection (BACI) Program procedure contains:

1. Requirements for visual examinations for leakage following every reactor shutdown which is 30 days or more since the last examination or any reactor shutdown due to Reactor Coolant System (RCS) leakage. This examination is to be performed as soon as possible after attaining Mode 3, Hot Standby conditions, and access to Containment is authorized. This examination is performed to identify any boric acid leakage or residue in the areas of the Containment where RCPB components are found. During a non-refueling outage these examinations would not require any insulation removal.

2. During each refueling outage or forced outage where Mode 5 (cold shutdown)/Mode 6 (refueling) conditions are attained the BACI Program requires more specific components be examined. These items are believed to be the principal areas where RCS leakage could begin or be located. These areas include reactor vessel closure head penetrations, incore instrumentation flange studs, reactor coolant pump suction elbows, reactor coolant pump studs/seals/controlled bleedoff lines, pressurizer penetrations (120 heater sleeves and 7 instrument nozzles), pressurizer manway studs, steam generator manway studs, valves and mechanical joints in the Chemical and Volume Control System, Safety Injection, Containment Spray, Pressurizer Safety/Relief and Reactor Coolant Systems in the Containment, and RCS and pressurizer surge line piping. During a non-refueling outage none of these examinations require insulation removal. During a refueling outage bare metal visual examinations would be required of the intersection of the Alloy 600 penetration and carbon steel base metal for all Alloy 600 penetrations.

3. The BACI Program requires personnel performing the examinations be certified VT-2, Level II or III Nondestructive Examiners. These personnel are qualified and certified in accordance with CCNPP's American Society of Mechanical Engineers Section XI and American Society for Non-Destructive Testing CP-189 programs.

4. The nondestructive examination procedures require the examiner to the extent possible, to locate the source of the leakage, quantify the leak rate, establish the leakage path of the coolant, and any RCPB contacted.

5. The BACI Program requires evidence of boric acid leakage to be documented via an Issue Report to ensure its inclusion in the site corrective action program. If boric acid leakage is identified, the boric acid residue must be removed and the underlying steel must be evaluated for wastage. If corrosion is noted, the component is required to be evaluated for suitability for continued service. The corrective action program also requires corrective actions to prevent recurrence.

Table A shows the Alloy 600 pressure boundary material, dissimilar metal Alloy 82/182 welds/connections and the principal carbon steel components in the RCPB at CCNPP. Table A also describes the inspection techniques, personnel qualifications, extent of coverage, frequency, degree of insulation removal/insulation type for examinations required by the CCNPP BACI Program.

Also as described in Reference (1), CCNPP has other programs that are intended to help ensure the integrity of the RCPB is maintained. One of these programs is the Alloy 600 Program Plan.

The CCNPP Alloy 600 Program Plan systematically determines actions necessary to eliminate nuclear safety risks and minimize economic risks associated with the potential for primary water stress corrosion cracking (PWSCC) of all Alloy 600 nozzles and pressure boundary components in the RCS. For each group of Alloy 600 components in the RCS, detailed analyses were performed. These analyses addressed susceptibility to PWSCC, safety significance, possible inspection techniques, potential repair/replacement techniques, potential mitigation techniques, regulatory issues, feasible options, economic analysis, and recommendations. The program plan makes short and long-term recommendations as necessary to address potential nuclear safety and economic issues. An Issue Report is generated for any potential nuclear safety issues identified to enter them in the corrective action program.

Under the Alloy 600 Program Plan, a detailed analysis was performed for each Alloy 600 penetration and other full penetration dissimilar metal Alloy 82/182 welds in the CCNPP RCS. An analysis was performed for each of the following nozzle groups:

- Reactor Vessel Head Major Nozzles - Control Element Drive Mechanism Nozzles and Incore Instrumentation Nozzles
- Reactor Coolant Piping Instrumentation Nozzles
- Pressurizer Instrumentation Nozzles
- Pressurizer Heater Sleeves
- Reactor Vessel Small Nozzles – Head vent and o-ring leakage monitor nozzle
- Full Penetration dissimilar metal Alloy 82/182 Welded Nozzles
- Steam Generator Primary Instrument Nozzles
- Weld Metal in J-groove Welds

Each of the nozzle groups identified above was assessed with regard to susceptibility to PWSCC. The fabrication history for each nozzle was researched to gather information with respect to fabrication sequence, machining techniques, heat treatment, and nonconformances resulting in rework. The chemical composition, mechanical properties, heat treatment, and processing was documented. The following critical variables affecting PWSCC were determined for each nozzle:

1. Thermomechanical processing of the Alloy 600 material - includes heat treatment, degree of hot or cold work.
2. Stress level – Total of residual plus applied stress.
3. Chemical environment.
4. Temperature.

An evaluation was performed of potential nuclear safety significance for each nozzle based on the expected failure mode, likelihood of failure and failure consequences. Potential inspection, repair/replacement, and mitigation techniques were identified for each nozzle group. Feasible options (e.g., proactively replace nozzles, augmented inspections, continue with status quo) for each type of penetration were identified.

Based on the results of the analysis, short-term and long-term recommendations were developed for each group of nozzles taking into consideration the safety and economic significance and the effect of regulatory issues.

Some of the actions taken as a result of the evaluations performed by the Alloy 600 Program Plan include:

1. Augmented eddy current examinations of the Unit 1 reactor vessel head vent line.
2. Pre-emptive installation of mechanical nozzle seal assemblies on most of the pressurizer instrument taps.
3. Development of a volumetric examination technique for the Unit 2 Pressurizer inner-to-outer heater sleeve welds

NRC REQUEST 2:

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to PWSCC (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

CCNPP RESPONSE:

Based on industry events and evaluations performed by the Alloy 600 Program Plan, CCNPP has determined that bare metal visual examinations for leakage should be performed on Alloy 600 penetrations. The evaluations performed by the Alloy 600 Program Plan determined that currently the full penetration dissimilar metal Alloy 82/182 Welded Nozzles could continue to be examined in accordance with ASME Section XI (i.e., VT-2 while insulated and periodic surface and/or volumetric examination) requirements. See Table A for the type of insulation on these points.

NRC REQUEST 3:

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

CCNPP RESPONSE:

The evaluations performed by the Alloy 600 Program Plan have determined that the frequency of examinations performed by the Boric Acid Corrosion Program are suitable to ensure the continued integrity of the RCPB. During a refueling outage CCNPP has no inaccessible areas for determining leakage.

NRC REQUEST 4:

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis to establish the acceptance criteria. In addition,

a. *If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or*
b. *If observed leakage is not determined to be acceptable, describe what actions are taken to address the leakage.*

CCNPP RESPONSE:

Calvert Cliffs Nuclear Power Plant procedures require an Issue Report be generated if boric acid leakage is found. We do not allow active boric acid leakage inside the Containment Building. If active leakage is identified, it must be corrected. If dry boric acid residue is found, the source of the leakage is required to be identified. If the leakage is at a bolted connection, the VT-2 procedure requires a bolt nearest the source of the leakage be removed and VT-1 examined. When the removed bolt has evidence of degradation, all remaining bolting in the connection are removed and examined. Any corrosion found which exceeds the lesser of 10 percent of section thickness or 1/32", requires documentation in accordance with the VT-2 procedure. If corrosion exceeds 10 percent of the section thickness, the component must be evaluated to determine if the component is acceptable for continued service or repair/replacement is necessary.

NRC REQUEST 5:

Explain the capabilities of your program to detect the low levels of RCPB leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components in the leak path.

CCNPP RESPONSE:

Calvert Cliffs Nuclear Power Plant Units 1 and 2 are Combustion Engineering pressurized water reactors that do not have bottom reactor pressure vessel head incore instrumentation nozzles. However, we do perform a VT-2 examination of the reactor pressure vessel bottom head every refueling outage. Our reactor vessel bottom head has no penetrations and is uninsulated. In Reference (2), we described our intention to perform non-visual examinations of our incore instrumentation nozzles, which are located in the reactor vessel closure head.

NRC REQUEST 6:

Explain the capabilities of your program to detect the low levels of RCPB leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your *program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components in the leak path.*

CCNPP RESPONSE:

Calvert Cliffs Nuclear Power Plant performs trending of parameters which could be indicative of possible RCPB leakage such as reactor coolant leakage, containment sump discharge frequencies, containment humidity, containment atmosphere activity and iodine levels, containment surface contamination, and containment air cooler performance.

NRC REQUEST 7:

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

CCNPP RESPONSE:

As described in our responses to Questions 1, 2, and 3 above, evaluations performed as part of the Alloy 600 Program Plan were used to determine appropriate inspections. All Alloy 600 is considered susceptible to PWSCC, so all Alloy 600 RCPB penetrations and welds are examined for leakage every outage. We use susceptibility models to decide to perform bare metal visual examination on all penetrations each outage. We use consequential model only for economic analyses. In most cases the economic consequences of PWSCC support deployment of mitigation, modification, or replacement activities prior to occurrence of through wall leakage.

NRC REQUEST 8:

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

CCNPP RESPONSE:

To date, the manufacturer of Combustion Engineering Nuclear Steam Supply Systems has made inspection recommendations for pressurizer heater sleeve penetrations and reactor vessel head control element drive mechanism (CEDM) penetrations only. The pressurizer heater sleeve penetrations recommendation calls for visual examination of the interface between the pressurizer lower head and the heater sleeve. The reactor vessel head CEDM penetrations recommendations are for non-visual examinations required to detect cracking in the CEDM penetrations.

The examinations being performed by the CCNPP Boric Acid Inspection Program, as described in response to Question 1 and in the exams described in Reference (2), exceed the requirements of the recommendations of the Nuclear Steam Supply System vendor.

NRC REQUEST 9:

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

CCNPP RESPONSE:

Calvert Cliffs is in compliance with plant Technical Specifications and 10 CFR 50.55(a) because we do not allow active boric acid leaks from RCPB components. Our VT-2 examination procedure requires the source of any leakage to be identified and any general corrosion caused by boric acid greater than 1/32" to be evaluated. We will remain in compliance with these regulatory requirements because our BACI Program would promptly identify any leakage, should it occur, and would cause the leak to be repaired, the boric acid to be removed, and the underlying steel to be evaluated for evidence of boric acid corrosion.

REFERENCES:

1. Letter from Mr. C. H. Cruse (CCNPP) to Document Control Desk (NRC), dated May 15, 2002, 60-Day Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

2. Letter from Mr. P. E. Katz (CCNPP) to Document Control Desk (NRC), dated October 9, 2002, 30-Day Response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Penetration Nozzle Inspection Programs"

TABLE A

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Reactor Vessel Closure Head Penetrations	Qualified VT-2	VT-2	100%	Every RFO	Unit 1- Inspection was achieved by going under insulation with minimal removal required/Rigid metallic Unit 2-Insulation is planned to be removed 1st bare metal exam will be performed in 2003/Blanket type	See response to Request #4
Reactor Vessel Lower Head	VT-2	VT-2	100%	Every RFO	N/A/No Insulation	See response to Request #4
Reactor Coolant System Instrument Taps	VT-2	VT-2	100%	Every RFO(1)	Enough Insulation is removed to perform 100% inspection of the interface between the Alloy 600 penetration and the carbon steel piping/ Rigid metallic and Blanket type	See response to Request #4
Pressurizer Instrument Taps	VT-2	VT-2	100%	Every RFO(1)	Enough Insulation is removed to perform 100% inspection of the area of the penetration / Rigid metallic and Blanket type	See response to Request #4
Pressurizer Bottom/Head Heater Sleeves penetrations	VT-2 with VT-1 requirements for distance and lighting	VT-2	100%	Every RFO(1)	Enough Insulation is removed to perform 100% inspection of the interface between the Alloy 600 penetration and the carbon steel vessel Blanket type	See response to Request #4
Reactor Coolant Piping	VT-2	VT-2	100%	Every RFO(1)	Only when required for ASME XI surface/volumetric examinations	See response to Request #4

TABLE A

Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Steam Generator Primary Instrument Nozzles	VT-2	VT-2	100%	Every RFO[1]	Enough Insulation is removed to perform 100% inspection of the interface between the Alloy 600 penetration and the carbon steel steam generator/rigid metallic and Blanket type	See response to Request #4
Full Penetration dissimilar metal Alloy 82/182 Welded Nozzles	VT-2	VT-2	100%	Every RFO[1]	Only when required for ASME XI surface/volumetric examinations	See response to Request #4
Steam Generator Primary Head	VT-2	VT-2	100%	Every RFO[1]	Only when required for ASME XI volumetric examinations	See response to Request #4
Pressurizer Vessel	VT-2	VT-2	100%	Every RFO[1]	Only when required for ASME XI volumetric examinations	See response to Request #4

[1] These components are examined following every reactor shutdown which is 30 days since the last examination or any reactor shutdown due to RCS leakage during a non-refueling outage no insulation is removed for performance of these examinations.