EXPEDITE!

CALVERT CLIFFS
LICENSE RENEWAL DOCUMENTS
(Incoming BG&E documents and NRC meeting summaries.)

- Process IMMEDIATELY upon receipt.

- Upon completion, IMMEDIATELY hand deliver to next step in process.

Charles H. Cruse
Vice President
Nuclear Energy

Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 495-4455

A Member of the
Constellation Energy Group

December 30, 1999

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
Revised Line Items for the List of Specific Information Needed to Support the 10 CFR 54.29 Finding for the License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2

REFERENCES:

(a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated December 6, 1999, "Specific Information Needed to Support the 10 CFR 54.29 Finding for the License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2"

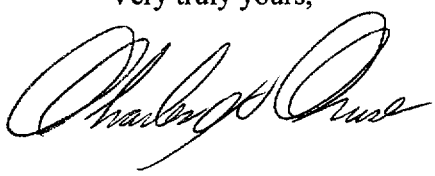
(b) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated October 22, 1999, "Additional Information Regarding the License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2"

Reference (a) forwarded a list from Baltimore Gas and Electric Company (BGE) to the NRC staff for the development of the basis of information needed to support the 10 CFR 54.29 finding for the BGE License Renewal Application. During a teleconference call between NRC Staff and BGE on December 14, 1999, BGE agreed to certain revisions to Item Nos. 8, 16, 46, 106, 112-116, 262, and 273 from Reference (a). Those eleven revised line items are contained as Attachment (1) to this letter and replace the corresponding line items in Reference (a). Baltimore Gas and Electric Company understands from NRC Staff that, with this action, the exercise to generate this list, which will be incorporated as Appendix E of NRC's Safety Evaluation Report (SER) on BGE's License Renewal Application, is complete.

Baltimore Gas and Electric Company understands that at this point in time, the SER has been finalized as NUREG 1705. Any needed changes BGE discovers from this point would be made as part of the incorporation of Appendix E into the Calvert Cliffs Updated Final Safety Analysis Report in the next annual update. For instance, BGE has found an editorial error in Item No. 46 of the list, regarding an alternative involving replacement of Cast Austenitic Stainless Steel (CASS) components with components that contain no CASS. The body of the SER and Item No. 6 in Reference (b) are clear that the alternative simply involves replacement, with no particular additional qualifiers.

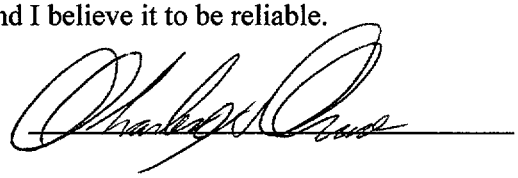
Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,



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

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy Division, Baltimore Gas and Electric Company (BGE), and that I am duly authorized to execute and file this response on behalf of BGE. 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 BGE employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.



Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 30 day of December, 1999.

WITNESS my Hand and Notarial Seal:

Michelle D. Hall
Notary Public

My Commission Expires:

February 1, 2003
Date

CHC/RCG/dlm

Attachment (1): Revised Line Items for List of Information Needed to Support the 10 CFR 54.29 Finding for the License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
S. R. Peterson, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
C. I. Grimes, NRC
D. L. Solorio, NRC
Resident Inspector, NRC
R. I. McLean, DNR
J. H. Walter, PSC

ATTACHMENT (1)

REVISED LINE ITEMS FOR LIST OF INFORMATION NEEDED TO SUPPORT THE 10 CFR 54.29 FINDING FOR THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
December 30, 1999

ATTACHMENT (1)

REVISED LINE ITEMS FOR LIST OF INFORMATION NEEDED TO SUPPORT THE 10 CFR 54.29 FINDING FOR THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
8.	3.2	Cranes, Reactor Vessel Cooling Shroud	Reactor vessel cooling shroud structural support members	Corrosion Due to Boric Acid	Boric Acid Inspection Program (BACI) Program (MN-3-301)	The BACI is credited for discovery and management of general corrosion/oxidation and corrosion due to boric acid for the reactor vessel cooling shroud structural support members by performing visual inspections. This program requires visual inspection and investigation of any leakage or corrosion that is found. A visual examination of external surfaces is performed for components in a borated water system, including the reactor vessel head penetrations. This program will be modified to specify examinations during refueling outages of: (a) the reactor vessel cooling shroud anchorage to the reactor vessel head for evidence of boric acid leakage; and (b) all reactor vessel cooling shroud structural support members for general corrosion/ oxidation. The scope of the program is threefold in that it: (a) identifies locations to be examined; (b) provides examination requirements and procedures for the detection of leaks; and (c) provides the responsibilities for initiating engineering evaluations and the necessary corrective actions. The examinations will be performed each refueling outage using qualified inspectors and inspection techniques.	The program will be modified by midnight, July 31, 2014 (Unit 1) or midnight, August 13, 2016 (Unit 2).

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Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
16.	3.3A	Containment	Containment tendons	Prestress Losses	Containment Tendon Procedures (STP-M-663-1 for Unit 1 and STP-M-663-2 for Unit 2)	Procedure STP-M-663-1/2 provides instructions for the Containment Tendon Surveillance which includes: • Determining that for a representative sample of dome, vertical, and hoop tendons, each tendon retains a lift-off force equal to or greater than its lower limit expected range for the time of the test. • Removing one wire from each of a dome, vertical and hoop tendon checked for lift off force, and determining the extent of corrosion and the minimum tensile strength. • Performing a chemical analysis of the sheath filler grease from the selected surveillance tendons to detect changes in its chemical properties. For the selected tendon, a measurement of the lift-off point pressure is made and converted to lift-off force. This value is compared against a lower bound individual lift-off value. Selected wires are also removed for visual examination and testing. The testing determines the yield strength, ultimate tensile strength, and elongation at ultimate tensile strength. The visual inspections include an examination of the selected surveillance tendon ends to determine the extent of coverage of the sheathing filler and to detect the presence of water, an examination of all anchorage components for indications of corrosion, pitting, cracking, distortion, or damage, an examination of the surrounding concrete, and an examination of the removed tendon wire for signs of gross corrosion or damage. The existing tendon lift-off force curves will be re-evaluated to reflect the required prestress levels for the period of extended operation by monitoring the parameters specified in 10 CFR 50.55a(b)(2)(ix)(B), and applying the acceptance criteria for expected trends in prestressing forces and taking appropriate corrective action as specified in Section IWL of the ASME Code.	Program to be modified by midnight July 31, 2014 (Unit 1) or midnight August 13, 2016 (Unit 2).

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Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
46.	4.1	RCS	PP, PUMP and PZV (surge nozzle safe end)	Thermal Embrittlement	Cast Austenitic Stainless Steel (CASS) Evaluation Program	A new program will be developed to manage the effects of thermal embrittlement by identifying those components that may be susceptible to the effects of thermal embrittlement. The CASS Evaluation Program will: (1) screen components; (2) review operating experience; (3) utilize enhanced VT-1 inspection [a visual examination capable of ½ mil resolution] (for reactor vessel internals [RVI] only); and (4) follow industry programs to evaluate thermal embrittlement and change the program accordingly. Susceptibility of individual components to thermal embrittlement will be determined based upon the delta ferrite content of the component, the casting method (static or centrifugal), and the molybdenum content. When delta ferrite content is not documented for a component, it will be estimated using actual material data and Hull's equivalent factors. For components that fail the screening and are deemed susceptible to thermal aging embrittlement, the preferred alternative will be a flaw tolerance evaluation and augmented inspection. Fracture toughness properties used for the flaw tolerance evaluation will be estimated using the method in NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," and/or NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems." The intent of the analysis will be to determine if the respective component has adequate fracture toughness, based on its material properties, in order to be capable of performing its pressure boundary function under CLB conditions. A second alternative will be to replace the components with those that contain no CASS. The second alternative will be used if a component cannot be qualified for the license renewal term by flaw tolerance analysis, or leak-before-break analysis, or if it is more cost effective to replace rather than perform an analysis. Replacement of the component will make the age-related degradation mechanism (ARDM) non-plausible for the respective component. The corrective actions taken as part of the CASS Evaluation Program will ensure that the CASS components remain capable of performing their pressure boundary function under all CLB	Program to be implemented by midnight July 31, 2014 (Unit 1) or midnight August 13, 2016 (Unit 2).

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
						conditions. If the components do not meet the screening criteria described above, then: 1. There will be an augmented inspection combined with a flaw tolerance evaluation; or 2. A full leak-before-break evaluation will be performed to prove that current inspection requirements are adequate to prevent catastrophic failure; or 3. They will be replaced. If an augmented inspection flaw tolerance approach is chosen, the acceptable flaw size for the inspection will be determined as follows: • For non-niobium containing components having less than 25% delta ferrite, a component specific J-R curve will be generated. If a component contains niobium or 25% or greater delta ferrite, the actual fracture toughness properties will be determined on a case-by-case basis. • An elastic-plastic fracture mechanics analysis will be performed for the component to determine the critical flaw size that is stable under all anticipated normal and accident loadings. This analysis may be component specific, or an analysis that bounds a group of components may be referenced. • The critical flaw size will be used to determine the inspection acceptance criteria. The critical flaw size minus an allowance for flaw growth during operation until the next inspection will equal the allowable flaw size. If available inspection technology permits, the inspection for RCS components will be conducted using a volumetric examination appropriate to a pressure-retaining weld on an ASME Section XI category B-L-1, B-M-1, or BJ component. If available inspection technology does not permit a volumetric examination, an alternative approach similar to that described in Code Case N-481 will be used.	Continued From Previous Page

ATTACHMENT (1)

REVISED LINE ITEMS FOR LIST OF INFORMATION NEEDED TO SUPPORT THE 10 CFR 54.29 FINDING FOR THE LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
106.	5.2	CVCS	CVOPs and PCVs with air internal environment	General Corrosion	Check Unit 1(2) IA Quality (IPM 10000 [10001])	The quality of IA System components that are within scope is verified approximately quarterly, based on operating experience and applicable industry standards, in accordance with Checklist IPM10000 and IPM 10001. These checklists assure that the system is being maintained in accordance with industry standards for moisture (dewpoint) and particulate contamination. This procedure is used to mitigate the effects of general corrosion on control valve operators and pressure control valves.	Existing Program
112.	5.3	CC	PP, automatic vents, CKVs, CVs, HVs, PUMP casings, REs, RVs, SVs, TEs, TIs, TICs, TKs	Crevice Corrosion, Pitting	ARDI Program	The ARDI Program is described in Section 4.1.	Program to be performed prior to, and near, the end of the current license period (e.g., no sooner than five years prior to expiration of the current license term).
113.	5.3	CC	PP	Erosion Corrosion	ARDI Program	The ARDI Program is described in Section 4.1.	Program to be performed prior to, and near, the end of the current license period (e.g., no sooner than five years prior to expiration of the current license term).
114.	5.3	CC	PP, CKVs, CVs, HVs, PUMP casings, RVs, TEs, TIs, TKs	General Corrosion	ARDI Program	The ARDI Program is described in Section 4.1.	Program to be performed prior to, and near, the end of the current license period (e.g., no sooner than five years prior to expiration of the current license term).
115.	5.3	CC	Automatic vents, CVs, HVs, RVs, SVs	Selective Leaching	ARDI Program	The ARDI Program is described in Section 4.1.	Program to be performed prior to, and near, the end of the current license period (e.g., no sooner than five years prior to expiration of the current license term).
116.	5.3	CC	CKVs, CVs	Wear	ARDI Program	The ARDI Program is described in Section 4.1.	Program to be performed prior to, and near, the end of the current license period (e.g., no sooner than five years prior to expiration of the current license term).

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Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
262.	6.1	Cables	EPR/XLPE insulated power cables EPR/XLPE/XLPO instrumentation cables in critical service	Thermal and Radiative Effects Insulation Resistance Reduction of Instrumentation Cables	Cables Aging Management Program	BGE is committed to managing the cable commodity during the period of extended plant operation as follows: • The ethylene-propylene-rubber (EPR)/ crosslinked polyethylene (XLPE) insulated cables that are in power service, routed with maintained spacing, and subject to plausible thermal aging are going to be monitored with an aging management program. This program will ensure that cable replacement would occur prior to any cable functional failure. BGE is working to define a conditioning monitoring program that will provide acceptance criteria and testing/analysis to determine at what point the cables should be replaced. • The EPR/XLPE insulated cables in power service that run inside of Containment and subject to plausible synergistic thermal and radiative aging are also going to be monitored by an aging management program. This program will ensure that cable replacement would occur prior to any cable functional failure. BGE is working to define a conditioning monitoring program that will provide any necessary acceptance criteria and the testing and analysis that would be required to determine at what point the cables would actually be replaced. • EPR/XLPE/crosslinked polyolefin (XLPO) insulated instrumentation cables in critical service are subject to plausible insulation resistance reduction, which is managed by the existing MN-1-211 instrument calibration program. • EPR cable terminations associated with 4 kV SW and SRW pumps are subject to thermal aging. The aging will be managed by existing EPMs 04000, 04003, and 05135.	Program to be implemented by midnight July 31, 2014 (Unit 1) or midnight August 13, 2016 (Unit 2).

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273.	6.3	EQ Equipment	All long-lived components on the EQ Master List	Thermal, Radiative, and Kapton-Unique Aging Effects	CCNPP 10 CFR 50.49 Program	BGE will continue to execute its 50.49 Program during the period of extended operation. The Calvert Cliffs 50.49 Program will continue to be administered in accordance with the regulatory requirements of 10 CFR 50.49, the Division of Operating Reactors Guidelines as transmitted by NRC Bulletin 79-01B, and NUREG-0588. The program includes requirements for determining the components in scope per 10 CFR 50.49 and options for management of thermal and radiative aging of components found to be in the scope of 10 CFR 50.49. The program contains the elements necessary to ensure that 50.49 equipment will remain qualified to execute the required 50.49 function(s) under all design loading conditions should a design basis event occur at the end of extended plant life.	Existing Program