CHARLES H. CRUSE
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410 495-4455

BGE

November 4, 1998

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 for the Review of the Calvert Cliffs Nuclear Power Plant, Units 1 & 2, Integrated Plant Assessment Report for the Fuel Handling Equipment and Other Heavy Load Handling Cranes

REFERENCES:

(a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated November 22, 1997, "Request for Review and Approval of System and Commodity Reports for License Renewal"

(b) Letter from Mr. D. L. Solorio (NRC) to Mr. C. H. Cruse (BGE), August 26, 1998, "Request for Additional Information for the Review of the Calvert Cliffs Nuclear Power Plant, Units 1 & 2, Commodity Report for the Fuel Handling Equipment and Other Heavy Load Handling Cranes"

(c) Letter from Mr. D. L. Solorio (NRC) to Mr. C. H. Cruse (BGE), September 24, 1998, "Renumbering of NRC Requests for Additional Information on Calvert Cliffs Nuclear Power Plant License Renewal Application Submitted by the Baltimore Gas and Electric Company"

Reference (a) forwarded four Baltimore Gas and Electric Company (BGE) system and commodity reports for license renewal. Reference (b) forwarded questions from NRC staff on one of those four reports, the Integrated Plant Assessment Report for the Fuel Handling Equipment and Other Heavy Load Handling Cranes. Reference (c) forwarded a numbering system for tracking BGE's response to all of the BGE License Renewal Application requests for additional information and the resolution of the responses. Attachment (1) provides our responses to the questions contained in Reference (b). The questions are renumbered in accordance with Reference (c).

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Should you have further 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 4 day of November, 1998.

WITNESS my Hand and Notarial Seal: Michelle D. Hall
Notary Public

My Commission Expires: February 1, 2002
Date

CHC/DLS/dlm

Attachment: (1) Response to Request for Additional Information; Integrated Plant Assessment Report for the Fuel Handling Equipment and Other Heavy Load Handling Cranes

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
S. S. Bajwa, NRC
A. W. Dromerick, NRC
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D. L. Solorio, NRC
Resident Inspector, NRC
R. I. McLean, DNR
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ATTACHMENT (1)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION; INTEGRATED PLANT ASSESSMENT REPORT FOR THE FUEL HANDLING EQUIPMENT AND OTHER HEAVY LOAD HANDLING CRANES

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
November 4, 1998

NRC Question No. 3.2.1

Provide the basis for excluding the spent fuel shipping cask wash down pit, a structural component in the Spent Fuel Storage System, and the fuel transfer tube from the scope of license renewal (*in Baltimore Gas and Electric Company's [BGE's] License Renewal Application [LRA]*).

BGE Response

The spent fuel shipping cask wash down pit and the fuel transfer tube are within the scope of license renewal and subject to aging management review (AMR). The fuel transfer tube is addressed in Section 3.3A, "Primary Containment Structure," of BGE's LRA. The reinforced concrete of the spent fuel cask wash down pit is addressed in Section 3.3E, "Auxiliary Building and Safety-Related Diesel Generator Building Structures," of the LRA.

The spent fuel cask wash down pit liner is constructed of stainless steel and primarily experiences the ambient atmospheric conditions of the Auxiliary Building spent fuel area. The stainless steel liner is wetted with borated and/or demineralized water during spent fuel storage cask loading operations. Due to the relatively mild atmosphere and infrequent wetting, there are no plausible age-related degradation mechanisms (ARDMs) for the stainless steel liner and, therefore, no discussion of aging management programs in the LRA.

NRC Question No. 3.2.2

Section 3.2.1, pages 3.2-1 and 3.2-2, briefly discuss the spent fuel stainless steel storage racks. While the spent fuel storage racks are not specifically identified as subcomponents within the Spent Fuel Storage System that are within scope of license renewal, they are identified as components subject to an AMR. Please clarify the scoping conclusion for the spent fuel storage racks, and provide a cross reference to where the discussion is provided in the LRA.

BGE Response

The Spent Fuel Storage System was identified as within the scope of license renewal in Section 3.2.1 on page 3.2-1. On page 3.2-4, the spent fuel storage racks were identified as components performing one of the structural intended functions determined in accordance with 10 CFR 54.4(a)(1) and (2) and, therefore, within the scope of license renewal and subject to AMR. The spent fuel storage racks are addressed in Section 3.3E of the LRA.

NRC Question No. 3.2.3

Section 3.2.1, under New Fuel Storage and Elevator, states that the new fuel elevators are part of the fuel handling system discussed in a subsequent paragraph in Section 3.2. Please explain why the system is called the new fuel storage and elevator system, yet the new fuel elevators are described as not being part of the system.

BGE Response

System and component naming schemes attempt to describe functional relationships between equipment but are often arbitrary. Also, component system assignments are subsequently changed to more appropriately align the components' function with the function of a particular system. The Fuel

Handling System encompasses those components used to move fuel from the time of receipt of new fuel to the storage of spent fuel in the spent fuel pool.

NRC Question No. 3.2.4

Section 3.2.1, page 3.2-5, includes a statement that there are components in the crane system that are not subject to the guidelines in NUREG-0612 because (1) there is adequate separation between the lift points and safe shutdown equipment, and (2) the load does not qualify as a heavy load. Please provide the distance BGE considers as adequate separation and the basis? Also, explain how adequate separation between lift points and irradiated fuel is considered when scoping the components in the crane system that are subject to the guidelines in NUREG-0612.

BGE Response

The Nuclear Regulatory Commission (NRC) issued NUREG-0612, "Control of Heavy Loads at Nuclear Power Facilities," in July 1980, to document the results of Unresolved Safety Issue A-36, "Control of Heavy Loads." NUREG-0612 provided a set of guidelines intended to minimize the possibility of load drops on safe shutdown or decay heat removal systems. In response to Generic Letters (References 1 and 2), BGE submitted a two-phase report reviewing provisions for handling and control of heavy loads at Calvert Cliffs, and evaluating these provisions with respect to the guidelines of NUREG-0612. The NRC accepted our Phase I evaluation in an NRC Safety Evaluation Report (Reference 3), as supplemented in Reference (4). In Generic Letter 85-11 (Reference 5), the NRC declined to review Phase II responses and released all the respondents from any commitments made in them. No distance was specified in the documents BGE provided to NRC. Reference (3) contains a list of "overhead handling systems" excluded from NUREG-0612 because "lift points and safe shutdown equipment are sufficiently separated."

NRC Question No. 3.2.5

Is the spent fuel shipping cask wash down pit reinforced concrete subject to AMR? If not, provide the basis for excluding it from an AMR.

BGE Response

Please refer to the response to Question No. 3.2.1 above.

NRC Question No. 3.2.6

Is the spent fuel shipping cask wash down pit stainless steel liner subject to AMR? If not, provide the basis for excluding it from an AMR.

BGE Response

Please refer to the response to Question No. 3.2.1 above.

NRC Question No. 3.2.7

Provide the basis for concluding there are no potential or plausible ARDMs warranting aging management for the fuel transfer tube.

BGE Response

As stated in Section 3.2.1 on page 3.2-5, the fuel transfer tube is addressed for its structural intended function(s) as part of the building in which it is housed. The fuel transfer tube, including identification of potential and plausible ARDMs, is addressed in Section 3.3A of the LRA.

NRC Question No. 3.2.8

The potential and plausible ARDMs for the Fuel Handling Equipment (FHE) and Heavy Load Handling Crane (HLHC) systems have been listed in Table 3.2-1 of Section 3.2 of the LRA. Fatigue, wear, and mechanical degradation/distortion has been considered a plausible degradation mechanism for the wire ropes. When bent over a sheave, a wire rope's load-induced stretch can cause it to rub against the groove, causing wear on the sheave or drum. Discuss the results of your evaluation of the wear of the sheaves and drums in contact with the wire ropes. Also indicate whether or not the sheaves and drums in contact with the wire ropes are subject to an AMR.

BGE Response

As stated in Section 3.2.1 on page 3.2-5, ". . . Structures and components subject to an AMR shall encompass those structures and components (i) That perform an intended function, as described in §54.4 without moving parts or without a change in configuration or properties . . ."

The scoping process determined that some structural devices, such as drums and sheaves, performed their intended function(s) while in motion. Such devices were considered to be active subcomponents and were, therefore, not subject to AMR.

NRC Question No. 3.2.9

Indicate why fatigue, wear, and mechanical degradation/distortion are not considered plausible ARDMs for the clips, bolts and stops in the Spent Fuel Cask Handling Crane, Polar Crane (PC), and Intake Structure Semi-Gantry Crane subcomponents. Additionally, include in the response a discussion as to why mechanical degradation/distortion of clips, bolts and stops is not plausible in light of the fact that these cranes are subject to accidental loadings during normal operations as described in Section 3.2 on page 3.2-23.

BGE Response

Fatigue is not a plausible aging mechanism for the carbon steel clips, bolts, and stops listed in the question due to the following:

- Potential low-cycle fatigue due to localized elevated temperatures are not anticipated to be significant (temperatures are not expected to exceed 120°F).
- The carbon steel components of the FHE and HLHC are designed such that their stresses are well below the yield stress range.

- S-N (stress versus cycles) curves for American Society for Testing and Materials (American Society of Mechanical Engineers) A36 steel (normal low carbon steel used in structural steel members) show that the fatigue limit stress is greater than the allowable stress where stress reversals do not occur. A review of the carbon steel components used in the FHE and HLHC concludes that they normally resist only gravity loads (or slight impact loads for crane components), and for the most part do not undergo stress reversals. (An exception is the PC carbon steel crane rails.)

- From the Crane Manufacturers Association of America Specification #70 for overhead traveling cranes, the service level for the Spent Fuel Cask Handling Crane, the Intake Structure Semi-Gantry Crane, the Transfer Machine Jib Crane and the PC is "A" (does not frequently lift the rated load and with a load cycle less than 200,000). All the above cranes typically only lift heavy loads occasionally. Service level "A" allows a stress range up to 40 ksi. Since the crane components (with the aforementioned exception) will not undergo stress reversals (stress would only go from "nothing" to a maximum), the maximum stress range would be, conservatively, the allowable stress for A36, which is 21.6 ksi. Therefore, there is no limit on the number of stress cycles allowed for the cranes.

Wear results from relative motion between two surfaces (adhesive wear), from the influence of hard, abrasive particles (abrasive wear), or fluid stream (erosion), and from small, vibratory or sliding motions under the influence of a corrosive environment (fretting). Since none of these conditions apply to the fixed structural components of the FHE and HLHC, wear is not a plausible ARDM for these components.

Mechanical degradation/distortion as utilized in Section 3.2 of the LRA encompasses damage to wire rope (hoist cable) structure such as kinks, birdcaging, crushing, etc., and as such this ARDM is not applicable for other carbon steel structural components.

NRC Question No. 3.2.10

Low cycle fatigue is considered plausible for the PC rails and fatigue has been identified as a potential ARDM for this item. It is stated in Section 3.2 that this ARDM, if unmanaged, could result in unstable crack growth under design loads at the flame-cut hole locations. Discuss your plans for mitigating the potential failure at flame cut holes and the potential fatigue damage in the PC trolley rails and in other FHE and HLHC components where flame cut holes might exist.

BGE Response

As stated in Section 3.2.2 on page 3.2-20, periodic inspection of the PC trolley rails for the effects of fatigue and the effectiveness of corrective actions is controlled through the existing Preventive Maintenance Program. Preventive Maintenance Tasks 10992001 and 20992000, "Perform NDE on Polar Crane Rails," are automatically scheduled and implemented in accordance with Administrative Procedure MN-1-102, "Preventive Maintenance Program." These tasks direct visual inspection of the PC rails, and subsequent non-destructive examination if there is evidence of cracking. Currently, inspection of the PC rails is performed on a four-to-six-year interval. Results are evaluated against prior inspection records to verify adequacy of weld repairs, identify trends, and determine the necessity for future inspections. The Calvert Cliffs Corrective Actions Program is used to take the necessary corrective actions. No other FHE and HLHC components at Calvert Cliffs have flame cut holes.

NRC Question No. 3.2.11

In Section 3.2.2 of the LRA, Table 3.2-1 lists those FHE and HLHC related structural components and subcomponents (the spent fuel shipping cask stainless steel support platform, incore instrumentation trash racks stainless steel structural members, spent fuel pool platform stainless steel structural members, spent fuel elevator subcomponent stainless steel structural members, and fuel transfer carriage subcomponent stainless steel structural members) that are subject to the AMR and the potential and plausible ARDMs for these systems. This table also indicates that the aging effects are not plausible for most of these structural components and subcomponents. Provide a summary of the basis upon which you concluded that the aging effects such as pitting/crevice corrosion, elevated temperature, irradiation, stress relaxation, fatigue, wear, mechanical degradation/distortion, corrosion due to boric acid, are not plausible for those structural components and subcomponents.

BGE Response

Pitting/Crevice Corrosion is not a plausible aging mechanism for the FHE stainless steel components, due to the following:

- The level of corrosion causing contaminants in the surrounding fluid (spent fuel pool and refueling canal water) is not sufficient to perpetuate pitting/crevice corrosion.
- Due to strict water control, the FHE stainless steel components could not experience localized build up of debris (e.g., crud), which is required for this type of corrosion.
- Also, pitting/crevice corrosion is not an issue for the FHE stainless steel components not in a wetted environment.

Elevated Temperature is not a plausible aging mechanism for the stainless steel components of the FHE due to normal operating temperatures within pressurized water reactor Containment and Auxiliary Building structures maintained less than 120° F, which is well below the level at which the structural integrity of steel begins to be significantly affected.

Irradiation is not a plausible aging mechanism for the stainless steel components of the FHE because the cumulative radiation exposure experienced by the FHE stainless steel components throughout the license renewal term is expected to be far below the level of 1E17 neutrons/cm^2 (E> 1MeV) below which the effects are negligible.

Stress Relaxation is not a plausible aging mechanism for the stainless steel bolts of the FHE, due to the following:

- The miscellaneous connection bolts/fasteners of the FHE are not exposed to high temperatures or high radiation.
- The miscellaneous stainless steel connection bolts of the FHE components do not depend on preload for functionality (i.e., even if a connection bolt were to lose its preload, it would still resist its intended loading).

Fatigue is not a plausible aging mechanism for the FHE stainless steel components, due to the following:

- Potential fatigue due to localized elevated temperatures are not anticipated to be significant (expected temperatures are quite low - see "Elevated Temperature" section above).
- The stainless steel components of the FHE are designed such that their stresses are well below the yield stress range.
- A review of the stainless steel components used in the FHE concludes that they normally resist only gravity loads and do not undergo stress reversals. (One exception is the fuel upender SS structural members. Stress reversal loads caused by normal operation have been responsible for fatigue crack growth on these components in the past. Coincident factors such as a marginal original design, poor original installation, and/or subsequent modifications have contributed to the degradation. Modifications to the affected areas to correct this deficiency have eliminated the likelihood of fatigue degradation of the items.)

Wear is not a potential aging mechanism for the FHE stainless steel components because wear results from relative motion between two surfaces (adhesive wear); from the influence of hard, abrasive particles (abrasive wear) or fluid stream (erosion); and from small, vibratory or sliding motions under the influence of a corrosive environment (fretting). Since none of these conditions apply to the fixed structural components of the FHE and HLHC, wear is not a potential ARDM.

Mechanical degradation/distortion as utilized in Section 3.2 of the LRA encompasses damage to wire rope (hoist cable) structure such as kinks, birdcaging, crushing, etc., and as such this ARDM is not applicable for other stainless steel structural components.

NRC Question No. 3.2.12

As described in the first paragraph of page 3.2-3 in Section 3.2.1, during the 1996 Unit 1 outage, four fillet welds connecting structural members on the fuel upending machine in the refueling pool failed due to low-cycle fatigue. After the implementation of corrective actions, BGE concluded that fatigue will not be plausible for these FHE subcomponents. Provide the basis for concluding that low cycle fatigue is not plausible aging mechanism for other welds in stainless steel members such as fuel transfer tube supports, new fuel elevator subcomponents and other components listed in Table 3.2-1.

BGE Response

As stated in Section 3.2.1 on page 3.2-3, it was determined that original joint design, original fabrication, and subsequent changes to the upending machine operations led to low-cycle fatigue failure of the welds. A nodal analysis of the fuel upending machine design determined that addition of stiffeners at the weld joints and use of dual hydraulic cylinders for machine operation would make future fatigue failure of these welds unlikely. These recommendations were implemented for both fuel upending machines in Unit 2, and will be completed for the fuel upending machines in Unit 1 prior to their next scheduled use in moving fuel.

Fatigue is not plausible for other stainless steel FHE subcomponents since: (1) normal service loads result in stresses that are far below the allowable stress range; (2) potential fatigue due to localized elevated temperatures are not anticipated to be significant; and (3) stress reversals do not occur (an exception being the upending machines).

NRC Question No. 3.2.13

Provide the basis for concluding that (1) only the PC rails need to be covered under Group 3 Aging Management, but not other crane rails, and (2) Group 4 Aging Management is applied only to wire ropes, but not to other crane components and subcomponents.

BGE Response

- As depicted on Table 3.2-1 of the LRA, fatigue of crane rails is plausible only for the rails of the PC; therefore, programs to manage the effects of fatigue described under Group 3 are only required for those particular crane rails.
- As depicted on Table 3.2-1 of the LRA, Group (4) only addresses the unique plausible ARDMs applicable for wire rope.

NRC Question No. 3.2.14

Discuss to what extent "loose bolts" (loose bolts at the connection of steel members, loose anchor bolts at cracked reinforced concrete members, etc.) were considered as aging effects for some of the FHE and HLHC systems?

BGE Response

Stress relaxation is unloading of preloaded components caused by long-term exposure of materials to elevated temperature and/or neutron irradiation. Stress relaxation is a potential aging mechanism for components with substantial preload. Since some FHE and HLHC components have bolted connections, stress relaxation was considered a potential aging mechanism. However, since the miscellaneous connection bolts/fasteners of the FHE and HLHC are not exposed to high temperatures or high radiation and do not depend on preload for functionality (i.e., even if a connection bolt were to lose its preload, it would still resist its intended loading), stress relaxation was determined to be not plausible.

NRC Question No. 3.2.15

Provide a summary of the visual inspection procedures applied for the FHE and HLHC systems, including the scope, method, acceptance criteria, frequency, and documentation. Alternatively, describe the process for establishing these attributes.

BGE Response

The description is provided in Section 3.2.2 on pages 3.2-12, -13, -16, 17, -19, -22, and -23 for the referenced visual inspection procedures. These programs direct periodic activities that will discover degradation and invoke the site Corrective Actions Program when degradation is discovered. Detailed information concerning the inspection procedures is available onsite for review.

NRC Question No. 3.2.16

Provide a summary of the coatings inspection program that is intended to supplement the existing preventive maintenance tasks associated with the load handling equipment. Cite any Steel Structures Painting Council guidance that is used in the coatings inspection procedures. Generally describe the repair practices that are used for degraded coating conditions on the load handling equipment, and summarize the past experience with degraded coatings.

BGE Response

In lieu of a supplemental coatings inspection program for FHE and HLHC, the existing preventive maintenance program tasks will be modified, as indicated in Table 3.2-2, to incorporate explicit inspection requirements for the discovery of degraded coatings and material loss due to general corrosion/oxidation of carbon steel structural elements. Since the coating does not contribute to the components' intended functions, degradation of the coating provides an alert condition that triggers corrective action before corrosion effects impacts the components' ability to perform its intended function. Corrective action for degraded coatings and any actual metal degradation will be carried out under the Calvert Cliffs Corrective Actions Program as necessary. Operating experience regarding degraded protective coatings of the Intake Structure Semi-Gantry Crane is provided in Section 3.2.2 on page 3.2-11 of the LRA.

References

1. NRC Generic Letter, "Control of Heavy Loads," dated December 22, 1980 (Un-numbered)
2. NRC Generic Letter 81-07, "Control of Heavy Loads," dated February 3, 1981
3. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated May 27, 1983, "Safety Evaluation, Control of Heavy Loads -- Phase 1"
4. Letter from Mr. S. A. McNeil (NRC) to Mr. G. C. Creel (BGE), dated August 7, 1989, "Supplement to 'Phase I' Safety Evaluation of the Control of Heavy Loads - MPC C-10 (Unit 1 TAC 07978; Unit 2 07979)
5. NRC Generic Letter 85-11, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, dated June 28, 1985