CHARLES H. CRUSE
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Calvert Cliffs Nuclear Power Plant
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BGE

November 9, 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 Containment Spray System, and Errata

REFERENCES:

(a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated January 21, 1998, "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), September 3, 1998, "Request for Additional Information for the Review of the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 & 2, Integrate Plant Assessment Reports for the Containment Spray System"

(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"

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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 on the Containment Spray System. 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). Attachment (2) provides errata to Section 5.6, Containment Spray System, of the BGE License Renewal Application. The questions are renumbered in accordance with Reference (c).

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

Very truly yours,

Charles H. Cruse

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.

Charles H. Cruse

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 9th day of November, 1998.

WITNESS my Hand and Notarial Seal: Denise D. Sankis
Notary Public

My Commission Expires: 2/2/2002
Date

CHC/KRE/dlm

Attachments: (1) Response to Request for Additional Information; Integrated Plant Assessment Report for the Containment Spray System

(2) Errata to Section 5.6, Containment Spray System; License Renewal Application

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

ATTACHMENT (1)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION;

INTEGRATED PLANT ASSESSMENT REPORT FOR THE

CONTAINMENT SPRAY SYSTEM

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
November 9, 1998

NRC Question No. 5.6.6

Section 5.6.2 *[of the Baltimore Gas and Electric Company (BGE) License Renewal Application (LRA)]* states that some components in the shutdown cooling (SDC) flowpath experienced significant thermal transients during SDC operations. Please identify these components and characterize the extent of the thermal transients they experienced. Identify the parameters and specific criteria that are used to monitor and manage thermal cyclic fatigue for these components.

BGE Response

The components in question are described and the transients are characterized in LRA Section 5.6.2. Since the SDC flowpath is bounded by the SDC and safety injection (SI) nozzles (both considered part of the Reactor Coolant System [RCS]), no components in the SDC flowpath are directly monitored for thermal cyclic fatigue. Since fatigue is considered not plausible for the Containment Spray (CS) System, no programs are credited for managing thermal cyclic fatigue for these components.

NRC Question No. 5.6.7

Section 5.6.2 indicates that core spray system *[Containment Spray System]* components in the SDC flowpath, namely SDC heat exchangers, the associated piping, temperature instruments and valves, have fatigue usage factors which are bounded by the fatigue usage of the SDC and SI nozzles that connect the SI System piping to the RCS. Clarify the technical justification for this conclusion. Also, describe the fatigue criteria used in the design of the CS System components in the SDC flowpath and justify the applicability of that criteria to the period of extended operation.

BGE Response

As part of the development of the Fatigue Monitoring Program (FMP), design analysis documents were reviewed. The SI nozzles undergo a step change from 300°F to 45°F upon initiation of SDC as the ambient temperature water resident in the SDC piping moves through the SI nozzles. (The calculation conservatively assumes a minimum Auxiliary Building temperature of 45°F). After the injection of all of the resident water, the nozzles then undergo a step increase to the temperature of the SDC fluid exiting the SDC heat exchangers. The reviews determined that the transients experienced by the SDC heat exchangers are similar to the SI nozzles, but much less severe and the SI nozzles bound the SDC heat exchangers for fatigue usage.

All CS components in the SDC flowpath are designed to American National Standards Institute (ANSI) B31.7 Class 2. American National Standards Institute B31.7 refers to ANSI B31.1 for the fatigue evaluation. The fatigue evaluations for the CS spray components used a fatigue reduction factor multiplier of 1. A multiplier of 1 assumes a maximum of 7000 full range stress cycles for the life of the components. Since our design limits Calvert Cliffs to 500 initiations of shutdown cooling transients, the assumed maximum of 7000 full range stress cycles will not be attained during 60 years of operation.

NRC Question No. 5.6.8

Section 5.6.2 indicates that based on in service inspections and additional examinations, it was concluded that the integrity of welds in the CS pump discharge piping and the high pressure safety injection piping from the SDC heat exchanger discharge, have not been affected by the service environment and residual stresses that have induced pipe cracking elsewhere in the industry. It is further stated that, since these portions of the CS System may not have any flow due to flushing or performance testing for periods of at least 30 days during normal reactor operation, they were recognized as portions of the CS System which has a high likelihood of containing stagnant oxygenated borated water, an environmental condition which has induced cracks in welds elsewhere in the industry. On the bases of this information, justify the conclusion that similar cracking of welds due to residual stresses and fatigue will not occur in this portion of the CS System during the period of extended operation.

BGE Response

Nuclear Regulatory Commission IE Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," addressed this subject of piping containing stagnant oxygenated borated water. This Bulletin required licensees to evaluate certain systems' pipe welds for the potential for the stress corrosion cracking degradation mechanism and to perform visual and ultrasonic inspections of these potentially susceptible areas. The Bulletin identified residual welding stresses in the heat affected zone of Type 304 stainless steel weld material combined with chloride and oxygen impurities as the causes of the degradation. These preconditions were not expected to be present at Calvert Cliffs as a result of the use of procedures during construction that minimized sensitization of the heat affected zone of Type 304 stainless steel, and also by the strict adherence to chemistry control during plant operation. Baltimore Gas and Electric Company had had no instances of stress corrosion related cracking in stainless steel pipe containing stagnant or essentially stagnant borated water at that time. The results of the required inspections confirmed the expectations from the operating experience as no reportable observations were discovered. Based on the favorable results of the extensive examinations, BGE concluded that the integrity of the identified susceptible welds had not been affected by the service environment and residual stresses. Repeated inspection and examination, and 19 years of operating experience supporting the absence of this form of degradation at Calvert Cliffs, reinforces the conclusion originally reached in response to Bulletin 79-17, and justifies the aging management review conclusion that stress corrosion cracking is not plausible for the CS System in the period of extended operation. Additionally, since fatigue is not plausible for the CS System (also see the response to Question No. 5.6.6 above), and, as discussed above, substantial residual stresses are not expected to exist, any cracking of the welds due to some combination of residual stresses and fatigue is not expected to occur.

NRC Question No. 5.6.9

Section 5.6.2 indicates that the SDC and SI nozzles that connect the SI System piping to the RCS are among the 11 fatigue-critical locations selected for monitoring under the Calvert Cliffs FMP. Describe the specific criteria used for selecting these nozzles for the FMP and indicate the reason the FMP calls for an engineering evaluation of these nozzles.

BGE Response

Per a telephone conference with NRC staff, BGE understands that the intent of this question is for BGE to describe the criteria for why the nozzles were selected, as stated, and describe any engineering evaluation that was performed in conjunction with that selection process.

The criteria for choosing these nozzles is their design analyses cumulative usage factor. These nozzles have the highest design analyses cumulative usage factors by virtue of experiencing the largest temperature transients during operation and, thus, bound the remaining SI and SDC flowpath components.

NRC Question No. 5.6.10

It is stated in the License Renewal Application that the FMP monitors and tracks low-cycle fatigue usage for the selected components of the Nuclear Steam Supply System and the steam generators. Describe the parameters that are monitored by the FMP that are applicable to the SDC and SI nozzles in the CS System. Also describe how the monitored parameters are compared to the fatigue analysis of record, and the criteria used to initiate corrective action.

BGE Response

The SDC outlet and SI nozzles are part of the RCS piping and are discussed in Section 4.1, "Reactor Coolant System," of the application. Pages 4.1-29 through 31 describe the FMP and the specific transients that are monitored for the SDC outlet and SI nozzles.

The critical transient for the SDC outlet nozzles is RCS cooldown following Mode 1 operations (≥ 5% power). The specific parameters monitored are RCS cold leg temperatures, pressurizer pressure and reactor power. The number of cooldowns are compared to the number of transients in the analyses of record every six months.

The critical transients for the SI nozzles are plant cooldown with initiation of SDC and the SI check valve test. The specific parameters monitored for SDC initiation are SDC flow, SI actuation signal, low pressure SI header temperatures, RCS cold leg temperatures, containment temperatures, and pressurizer pressure. The SI check valve tests are manually logged by plant operators. The specific parameters recorded for the SI check valve tests are charging temperature, RCS cold leg temperatures, and pressurizer pressure. The number of SDC initiations and SI check valve tests are compared to the number of transients in the analyses of record every six months.

For both locations, all other transients analyzed in the analysis of record are assumed to have occurred and the corresponding fatigue contribution is accounted for as "initial" fatigue usage in the FMP.

ATTACHMENT (2)

ERRATA TO SECTION 5.6, CONTAINMENT SPRAY SYSTEM;

LICENSE RENEWAL APPLICATION

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
November 9, 1998

ATTACHMENT (2)

ERRATA TO SECTION 5.6, CONTAINMENT SPRAY SYSTEM; LICENSE RENEWAL APPLICATION

The following change applies to Section 5.6 of the BGE LRA:

- On page 5.6-5, in the middle of the paragraph preceding "5.6.1.2 Component Level Scoping," the acronym "ANSI" should be defined as "American National Standards Institute," vice "American Nuclear Standards Institute."