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1650 Calvert Cliffs Parkway
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BGE

January 28, 1997

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
120-Day Response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

REFERENCES:

(a) Letter from Mr. B. K. Grimes (NRC) to Mr. C. H. Cruse (BGE), dated September 30, 1996, NRC Generic Letter 96-06: "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

(b) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated October 30, 1996, 30-Day Response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

The purpose of this letter is to forward our 120-day response to Reference (a). In our 30-day response (Reference b), we committed to provide a written summary report of our evaluation to:

(1) determine if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions; and,

(2) determine if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

The postulated accident conditions include a loss-of-coolant accident or a main steam line break event that each include a loss of offsite power. Attachment (1) contains our detailed 120-day response to the information requested.

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Should you have 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 hereby certify that on the 28th day of January, 1997, before me, the subscriber, a Notary Public of the State of Maryland in and for Calvert County, personally appeared Charles H. Cruse, being duly sworn, and states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information, and belief; and that he was authorized to provide the response on behalf of said Corporation.

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

My Commission Expires: 2/2/98
Date

CHC/JMO/dlm

Attachment: (1) Baltimore Gas and Electric Company's 120-Day Response to GL 96-06: "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
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ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S

120-DAY RESPONSE

TO GL 96-06:

"ASSURANCE OF EQUIPMENT OPERABILITY AND

CONTAINMENT INTEGRITY

DURING DESIGN-BASIS ACCIDENT CONDITIONS"

Calvert Cliffs Nuclear Power Plant
Units 1 & 2
January 28, 1997

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S 120-DAY RESPONSE TO GL 96-06: "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"

BACKGROUND

The Service Water (SRW) System is a closed system with a vented head tank. It uses plant demineralized water containing a corrosion inhibitor. The SRW System removes heat from the turbine plant components, blowdown recovery heat exchangers, containment air coolers (CACs), spent fuel pool cooling heat exchangers, and three emergency diesel generators. The SRW System for each unit is divided into two subsystems or loops by manually positioned valves. Each loop contains an SRW pump which pumps hot return fluid through the SRW heat exchanger to each of the system heat loads. A third pump, which can be aligned to either of the loops, is available for increased system reliability.

Pressure in each loop is maintained by an atmospheric head tank located on the 69-foot elevation of the Auxiliary Building. Each tank has a 3/4-inch vent line. The head tanks for each loop are connected to each other by an overflow line.

Inside containment, there are four CACs, two in each subsystem. Subsystems 11 and 21 serve the Nos. 11 and 12, and 21 and 22, CACs, respectively, which are located on the 45 foot-elevation of containment; while Subsystems 12 and 22 serve the Nos. 13 and 14, and 23 and 24, CACs, respectively, which are located on the 69-foot elevation of containment. For the remainder of the response, we will refer only to Unit 1 systems; however, the discussion will apply to both units.

During normal operation, both subsystems are required and are independent to the degree necessary to assure the safe operation and shutdown of the plant, assuming a single failure.

During a loss-of-coolant accident/loss of offsite power (LOCA/LOOP), numerous plant responses are occurring in rapid succession:

- During the first two seconds, the SRW pumps are assumed to coast down.
- Air-operated valves in the SRW System are actuated by the Emergency Safety Features Actuation System to place the SRW System in its initial accident line-up. In this line-up, the blowdown recovery heat exchangers, the spent fuel pool heat exchangers, and all coolers located in the Turbine Building are isolated.
- The CAC fans are also coasting down; however, the fan coast-down period will likely exceed one minute.
- The containment temperature, pressure, and steam/vapor content are rapidly increasing.
- After 10 seconds, the emergency diesel generators have reached full speed and the loss-of-coolant incident sequencer begins to restart essential plant loads.
- At 25 seconds, the CAC fans are automatically restarted, followed 5 seconds later (i.e., at 30 seconds) by the SRW pumps, to re-establish flow in the system.

**BALTIMORE GAS AND ELECTRIC COMPANY'S
120-DAY RESPONSE TO GL 96-06: "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"**

REQUIRED RESPONSE

Within 120 days of the date of this generic letter, addressees are required to submit a written summary report stating:

1) *actions taken in response to the requested actions noted above;*

Response

The requested actions are: (1) determine if CAC cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions; and (2) determine if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur. Items 2 through 5 below summarize our efforts taken in response to the requested actions.

2) *conclusions that were reached relative to susceptibility for water hammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment;*

Response

A) Water Hammer

We concluded there is a potential to create voiding in two of four CACs during the scenarios of LOCA/LOOP [concurrent] or a main steam line break with a LOOP. During the LOCA/LOOP, the SRW pumps and the CAC fans are without power for approximately 30 seconds. The stagnate water in the CACs is heated to the point of boiling. Boiling is predicted to begin as early as 15 seconds into the event. As the fluid boils, it creates large voids in the coolers.

Thermal analysis has demonstrated that the Nos. 11 and 12 CACs are not susceptible to boiling prior to restart of the SRW pumps during a LOCA/LOOP. The Nos. 13 and 14 CACs were found to be susceptible to boiling as early as 15 seconds into the event. The primary difference between the Nos. 11 and 12 CACs, and the Nos. 13 and 14 CACs, is their elevation. The Nos. 13 and 14 CACs are approximately 24 feet above the Nos. 11 and 12 CACs. The volume of steam that would be generated is sufficient to completely void the Nos. 13 and 14 CACs.

Compared to the LOCA/LOOP, the design basis small break LOCA and main steam line break represent a smaller mass/energy release into the containment atmosphere and, consequently, produce lower vapor, saturation temperatures, and condensation rates on the CACs. In all cases, the LOCA/LOOP with maximum safety injection is the bounding event (Reference 1, Chapter 14).

At 30 seconds into the event, the emergency diesel generators re-power the SRW pumps to re-establish flow in the system. This collapses the voids and, ultimately, results in a water hammer. Therefore, two of the four CACs and the associated piping system in each unit are susceptible to boiling and the resulting water hammer, as described in the generic letter.

B) Two-Phase Flow

We previously evaluated the potential to create two-phase flow conditions at the outlet of the CACs and associated downstream restrictions during worst-case LOCA conditions. The evaluation has been documented in a calculation. However, the evaluation did not fully consider the transient period immediately following the Engineering Safety Feature Actuation Signal.

In Reference (2), we stated that we would further review the transient period immediately following an Engineering Safety Feature Actuation Signal where the SRW System is reconfigured into the accident lineup through the isolation of Turbine Building loads and other automatic valve manipulations. From this review, we concluded that two-phase flow conditions would not be created during the transition period. The final documentation of this calculation has not been completed; we anticipate completing the calculation by February 28, 1997.

At Calvert Cliffs, SRW flow is throttled to the CACs on their inlet side, which is a substantial difference from the two-phase flow scenario described in the generic letter. There are no flow restrictions downstream of the CACs which would act to force the fluid closer to two-phase flow conditions. The piping on the outlet of the higher two CACs immediately drops approximately 40 feet in elevation. This drop results in pressure recovery as the fluid moves further from saturation conditions. Therefore, the most limiting condition for two-phase flow occurs at the outlet of the highest CAC. A computer model has been used to predict the fluid flow rate and pressure at the limiting location. The CAC outlet temperature has been calculated using the conservative assumption that the CACs are perfectly clean (i.e., the fouling factor is 0.0 Hr-Ft^2-°F/BTU). Based on the operation and design of our SRW System, assuming a conservative CAC fouling factor, we have concluded that two-phase flow will not exist during worst-case LOCA conditions.

C) Overpressurization

For overpressurization, we reviewed all piping that penetrates containment, and all safety-related piping internal to containment. We concluded that during a LOCA, or any other accident that results in a containment isolation signal, nearly all piping sections (i.e., safety-related and non-safety-related) that penetrate containment are isolated. Of these isolated piping sections, only six penetrations per unit have trapped fluid that could become pressurized. They are listed below.

PIPING SYSTEMS SUSCEPTIBLE TO OVERPRESSURIZATION			
SYSTEM	PENETRATION NUMBER	LINE SIZE / SCHEDULE	INSULATION
Reactor coolant pump seals controlled bleed off	1C	3/4" sch. 80s	None
Plant Water	37	3" sch. 40	None
Refueling pool recycle inlet	59	8" sch. 10s	None
Refueling pool outlet to fuel pool cooling pipe	61	8" sch. 10s	None
Containment plant heating outlet	62	3" sch. 40	1" fiberglass w/ aluminum
Containment plant heating inlet	64	3" sch. 40	1" fiberglass w/ aluminum

3) *the basis for continued operability of affected systems and components as applicable; and,*

Response

A) Water Hammer

The SRW System piping to the Nos. 13 and 14 CACs was analyzed for the anticipated water hammer loads. As discussed above, the Nos. 11 and 12 CACs are not subject to the boiling and subsequent water hammer. In addition, the piping system geometry was reviewed to ensure that Nos. 11 and 12 CACs were sufficiently isolated from the Nos. 13 and 14 CACs to ensure that there would be no interaction.

The piping system was analyzed using the force time history data generated using a fluid dynamics model. A forcing function which consists of force versus time was applied to each piping segment. For this analysis, a piping segment is the run of piping between bends, elbows or piping discontinuities such as tees or reducers. The pipe stress levels and restraint/anchor loads were reviewed against the operability limits identified in ASME Section III, Appendix F and the acceptance criteria of NRC Bulletins 79-02 and 79-14 as discussed in Reference (1), Chapters 4 and 5A, and Reference (3). The results of this analysis indicated several items would exceed design limits. All met operability limits.

The magnitude of the pressure pulse seen during the postulated water hammer event is approximately 1180 psia, without the effect of non-condensable gases. The peak pressure pulse anticipated was compared to the hoop stress limits/pressure design criteria for each component or piping segment. In all cases, faulted design bases hoop stress limits were met.

To provide a more realistic assessment of the overall magnitude of the postulated water hammer, the presence of dissolved gases was considered. The dissolved gas content is equal to the solubility of nitrogen at 95°F. Testing has shown little to no oxygen content. Low oxygen content is due to corrosion and hydrazine additions to scavenge oxygen. Accounting for dissolved gases resulted in approximately a 50% reduction in the

magnitude of the postulated water hammer. When the fluid boils, these gases, mainly nitrogen, come out of solution, and act as non-condensable gases during the period of void collapse.

The CACs were also evaluated to consider the effects of the pressure pulse on their cooler coil assemblies. The tubes in the coil assembly are 90-10 Copper-Nickel Alloy. The tubes are brazed, not rolled, to the supply and return headers. The coil assemblies were found to be sufficiently robust to withstand the pressure loads. Some deformation of the coil assemblies and associated fins could be expected. In addition, we performed a combination of qualitative and quantitative evaluations of the dynamic loads on the coolers in order to determine the potential effect of these loads on the pressure boundary and cooler functions. Based on these evaluations, we have concluded that the CACs are operable.

No detailed seismic analysis was performed as part of our water hammer evaluation. The magnitude of the water hammer completely envelopes the Operating Basis Earthquake and Safe Shutdown Earthquake (SSE) cases. As a result, if the system is shown to remain operable for the water hammer case, it will continue to be operable for the SSE case. Additionally, the loads from an SSE were not combined with the water hammer loads. Our licensing basis does not require an evaluation of the effects of an earthquake simultaneously with those of a LOCA. The effects of an earthquake must, however, be considered before and after the LOCA. Because this water hammer event occurs essentially concurrent with the LOCA (i.e., within 30 seconds), this approach is found to be acceptable.

Therefore, the CACs and SRW systems in Calvert Cliffs Units 1 and 2 were found to exceed design basis limits. However, they are operable during the postulated water hammer. Final documentation and independent review of the above analysis and evaluations are ongoing. They will be completed by February 28, 1997.

B) Two-Phase Flow

Since two-phase flow will not occur, the SRW System is operable during all anticipated scenarios.

C) Overpressurization

A thermal analysis was performed on the identified susceptible sections to determine the maximum temperature experienced by the isolated water during a LOCA event. The peak pressure that would develop in the piping was determined and the corresponding piping stress calculated. For operability evaluations, the calculated water pressures and stresses were based on elastic-plastic analyses. For design basis evaluations, the pressures and stresses were based on elastic analyses only.

For operability evaluations, plastic deformation of the pipe is taken into account in reducing the pressure of the confined water. When the pressure in the pipe causes the stresses in the pipe to exceed yield, the piping undergoes plastic deformation. This stage of deformation is characterized by large volumetric expansion without addition of load (pressure). As the pipe volume increases, the isolated water has more room to expand and, therefore, the pressure drops. The allowable stresses we used for the operability evaluation are defined in plant administrative procedures, which permit use of ASME Section III, Appendix F, for operability assessments. All the piping sections susceptible to water-filled isolation satisfy the prescribed allowables and are, therefore, operable.

4) *corrective actions that were implemented or are planned to be implemented.*

Response

For water hammer, the corrective actions are to modify the SRW System to prevent boiling during all anticipated scenarios. We are currently finalizing the scope of the specific modifications that will prevent boiling. For Unit 2, we are working on an expedited basis to complete a modification to the SRW System, which will provide a means to prevent boiling, by the end of the refueling outage which starts in March 1997. We anticipate that the modifications will require a license amendment for Units 1 and 2 in accordance with 10 CFR 50.59 and 50.90. For Unit 1, we anticipate completing modifications to prevent boiling by the end of the 1998 refueling outage. We expect to need the license amendment by the end of the 1997 refueling outage, which is scheduled to end in late April 1997.

For two-phase flow, no corrective actions are required. The modification proposed to resolve the CAC water hammer issue will further increase the degree of subcooling for the two-phase flow issue. Therefore, that modification will have no adverse impact on SRW System performance for two-phase flow.

For overpressurization, we determined that four of the six segments were found to exceed design basis requirements. In order to satisfy the design basis requirements, we will perform the corrective actions (summarized below) for each of these piping segments. The changes to Operating Instructions and any required modifications will be completed by the end of each unit's next refueling outage (Unit 1 - spring 1998, Unit 2 - spring 1997).

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S
120-DAY RESPONSE TO GL 96-06: "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"

ANTICIPATED CORRECTIVE ACTIONS FOR OVERPRESSURIZATION		
SYSTEM	PENETRATION NUMBER	POTENTIAL MODIFICATION
Reactor coolant pump seals controlled bleed-off	1C	Install pressure relief device or add insulation to the piping section that may be exposed to LOCA temperatures
Plant Water	37	Change Operating Instruction (OI 23D) to include line draining prior to plant operation
Refueling pool recycle inlet	59	Change Operating Instruction (OI 23H) to include line draining prior to plant operation
Refueling pool outlet to fuel pool cooling pipe	61	Change Operating Instruction (OI 23H) to include line draining prior to plant operation
Containment plant heating outlet	62	None required due to insulation
Containment plant heating inlet	64	None required due to insulation

REFERENCES:

(1) Updated Final Safety Analysis Report, Revision 19

(2) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated October 30, 1996, 30-Day Response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

(3) Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability (Generic Letter 91-18), dated November 7, 1991