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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

January 31, 1985

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTENTION: Mr. James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318
Request for Amendment

REFERENCES:
(a) BG&E letter from Mr. A. E. Lundvall, Jr., to Mr. J. R. Miller, dated December 22, 1983
(b) Safety Evaluation by NRC Division of Licensing, dated April 19, 1984
(c) BG&E letter from Mr. A. E. Lundvall, Jr., to Mr. R. W. Reid, dated May 1, 1978
(d) BG&E letter from Mr. A. E. Lundvall, Jr., to Mr. R. A. Clark, dated December 12, 1980
(e) Letter from Mr. D. G. Eisenhut, NRC, to all PWR Licensees, dated November 1, 1983

Gentlemen:

The Baltimore Gas and Electric Company hereby requests an Amendment to its Operating License Nos. DPR-53 and DPR-69, for Calvert Cliffs Unit Nos. 1 & 2, respectively, with the submittal of the proposed changes to the Technical Specifications. Change Numbers 1 and 2 below will be needed in time to support the Unit 1 start-up following the spring 1985 refueling outage in the middle of May 1985.

CHANGE NO. 1 (BG&E FCR 83-1044 Unit 1)

Remove existing pages 3/4 7-27, 7-28, 7-29, 7-30, 7-31, 7-46, 7-47, and 7-60 of the Unit 1 Technical Specifications and replace with attached marked up pages.

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DISCUSSION

By an earlier application for license amendment, reference (a), we requested and received approval for provisions in the list of safety-related snubbers, Table 3.7-4, to allow removal of selected snubbers and replacement with sway strut supports so long as certain conditions were met. The conditions were delineated in reference (b). This submittal forwards a list of those snubbers planned for modification under those conditions during the spring 1985 Unit 1 refueling outage. Revised stress calculations were performed by Bechtel, the Architectural Engineer, which justify outright deletion of three Unit 1 snubbers without replacement. Snubber 1-11-12 will be deleted for the following reasons:

A rigid restraint exists on the same piping line as the snubber at a distance of 4'7" from the snubber and is capable of carrying the relatively small loading this snubber would be required to carry. In addition, both the rigid restraint and snubber act to restrain axial motion of the same leg of pipe.

Snubbers 1-60-5 and 1-60-5A will be deleted because reanalysis of the piping section revealed even without the snubbers, the maximum analyzed piping stress following a design basis earthquake is still well below the code allowable value.

The stresses in the associated piping systems were analyzed by Bechtel computer code ME-101, which has been verified against an independent piping program SUPER PIPE, NRC benchmark problems, detailed manual checking, and the ASME Pressure Vessel and Piping benchmark computer program. The acceptance criteria applied is governed by the code USAS B31.1.0, 1969, referenced in the Updated Final Safety Analysis Report (FSAR) Section 5A.3.2.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed change has been evaluated against the standards set forth in 10 CFR 50.92, and has been determined to involve no significant hazards considerations, in that operation of the facility in accordance with the proposed amendment would not:

(i) involve any increase in the probability of occurrence or consequences of an accident previously evaluated in the Updated FSAR.

Removal of these snubbers will not result in any effect on the probability of occurrence of an accident previously evaluated. Because the piping system predicted stresses are within code allowable limits, as described earlier, the consequences of an accident previously evaluated will not be worsened.

(ii) create the possibility of a new or different type of accident from any accident previously evaluated.

Removing these snubbers would not induce any accident beyond those previously evaluated. The snubbers are piping restraints which serve to limit stress loadings during analyzed accidents in the associated piping to below code allowables.

(iii) involve a significant decrease in the margin of safety.

The piping stresses have been reevaluated with the snubbers removed and will remain below code allowable limits. Therefore, clearly there is no significant reduction in the margin of safety resulting from the removal of these snubbers.

CHANGE NO. 2 (BG&E FCR 83-1054) Unit 1 Only

Remove existing pages 3/4 7-26, 3/4 7-26b, 3/4 7-51, and 3/4 7-52 of the Unit 1 Technical Specifications and replace with the marked-up pages attached to this transmittal.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

We plan to modify Steam Generator Support Snubbers 1-63-13 through 1-63-28 by installing dedicated hydraulic reserviors, mounted as Seismic Category I for each of these snubbers. This represents a significant improvement to safety, since these snubbers are currently served by a common hydraulic reservoir.

References (c) and (d) forwarded our intention to perform monthly surveillance of reservoir fluid level in lieu of providing an individual dedicated reservoir for each snubber. This decision was made based on recommendations from the Architectural Engineer and ITT Grinnell (the snubber manufacturer) as well as plant staff. These snubbers are to be functionally tested during the refueling outage following June 30, 1985, and each monthly surveillance test requires a containment entry at power. Therefore, we have decided, based upon revised recommendations from ITT Grinnell, to modify the snubbers by providing the dedicated reservoirs during the Unit 1 spring 1985 refueling outage scheduled to begin April 1985.

These snubbers serve to restrain sudden lateral movement of the steam generator under seismic conditions and are described on Pages 4-8 and 4-9 of the Updated FSAR.

The Technical Specification Bases describe the visual inspection requirements of Technical Specification 3/4 7-8 as "maintaining a constant level of snubber protection to systems."

The proposed change does not involve any increase in the probability or consequences of an accident previously evaluated on the Updated FSAR. This is because with dedicated reservoirs the snubbers will no longer be subject to the possibility of common mode failure which existed with the common reservoir. Although this failure mode was improbable, the dedicated reservoirs do offer an improvement to safety.

No new or different kind of accidents from those previously evaluated in the Updated FSAR are created by this proposed change. This is because the snubbers are still available to provide the dynamic load support function during a design basis seismic event.

The proposed change does not involve a reduction in the margin of safety as described in Technical Specification Bases 3/4 7-8. This is because with the exception of the special visual inspection requirement for snubbers with common reservoirs, the Operability Action and Surveillance requirements will not be changed.

CHANGE NO. 3 (BG&E FCR 84-156)

Remove existing pages 3/4 1-17 of the Unit Nos. 1 and 2 Technical Specifications and replace with attached marked-up pages.

DISCUSSION

The proposed change would permit changing Operational **MODES** while certain Control Element Assembly (CEA) Interlocks are inoperable, so long as the CEA position is controlled to ensure proper alignment is maintained.

The CEA Motion Inhibit (CMI) is described in Section 7.5.3.3 of the Updated FSAR. The purpose of the CMI is described in Section 14.1-6. The CMI ensures that programmed CEA group overlap will be maintained and that a single CEA withdrawal event will not occur.

This proposed Technical Specification Change would permit changing Operational **MODES** while Technical Specification 3.1.3.1, Action B.2 is imposed. Current Technical Specifications permit continued plant operation in **MODES** 1 & 2 while CEA Motion Inhibit is inoperable, as long as the following condition is met:

> ". . . 2. Place and maintain the CEA drive system mode switch in either the "Off" or any "Manual Mode" position and fully withdraw all CEA's in groups 3 and 4 and withdraw the CEA's in group 5 to less than 5% insertion, or . . ."

We propose to add the statement, "While this CEA position limitation is maintained, the provisions of Specification 3.0.4 are not applicable." Technical Specification 3.0.4 is designed to ensure that entry into an Operational **MODE** must be made with a) the full complement of required systems, equipment or components **OPERABLE** and b) all other parameters as specified in the Limiting Condition for Operation (LCO) are met without regard for allowable deviations and out-of-service provisions contained in the Action Statements. In this case, since no single CEA withdrawal event can be of significance due to the withdrawal position limitation on CEA's, and the Technical Specifications currently permit this configuration during **MODE** 1 or 2 operation, the margin of safety in Technical Specification (TS) bases for TS 3.1.3.1 is not degraded by permitting entry into these **MODES**. Entry into Operational **MODE** 1 or 2 is no more severe than operation in **MODE** 1 or 2, with respect to the analyzed accidents. The TS for regulating CEA insertion limits (3.1.3.6) and CEA position indication (3.1.3.3) are not affected by this proposed change.

The ± 5% value corresponds well to the 7.5 inch value in the Limiting Conditions for Operation in TS 3.1.3.1. The ± 5% value actually corresponds to approximately ± 6.8 inches.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed change to the Technical Specifications has been reviewed against the standards in 10 CFR 50.92 regarding significant hazard considerations and has been determined to involve no significant hazards considerations, in that operating in accordance with the proposed change would not:

(i) involve any increase in the probability or consequences of an accident previously evaluated in the Updated FSAR, since no actual system modifications will be effected.

(ii) create the possibility of occurence of an accident different from an accident previously evaluated. This is because the single CEA withdrawal event is limited due to the CEA position restrictions.

(iii) involve any significant decrease in the margin of safety as described in the bases for Technical Specifications 3.1.3.1 through 6. By ensuring proper CEA configuration is maintained, and position indication is maintained **OPERABLE**, the CEA insertion limits will still remain an effective method of ensuring the TS Bases remain valid.

CHANGE NO. 4 (BG&E FCR 84-159)

Remove pages B3/4 5-1 and 3/4 9-8a of the Unit Nos. 1 and 2 Technical Specifications and replace with the marked-up pages attached to this transmittal.

DISCUSSION

The proposed change is being processed to provide clarification of two plant Technical Specifications to provide a more accurate description of plant design as described in the Updated FSAR.

The first proposed Technical Specification change relates to the Low Pressure Safety Injection System (LPSI), which is described in Section 6.3 of the Updated FSAR. The wording of Technical Specification 3/4 5.2 currently specifies that ". . . Two independent ECCS subsystems shall be **OPERABLE**, with each subsystem comprised of:

a. One **OPERABLE** high-pressure safety injection pump,

b. One **OPERABLE** low-pressure safety injection pump, and

c. An **OPERABLE** flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal. . . ."

The portions of the system that are described in the Technical Specification are independent. However, the portion of the piping downstream of the Low Pressure Safety Injection pumps is common to both headers (i.e., that portion containing CV-306 and associated piping to the RCS loops).

Single failure of CV-306 has been addressed by both the Nuclear Regulatory Commission (NRC) and Baltimore Gas and Electric Company in earlier correspondence. This valve is a locked open, fail open valve and is not required to function during or subsequent to a design basis loss of coolant accident. A key operator is provided in the control room to ensure the valve remains open under all required conditions. The valve is verified open via local indication at the valve, within four hours prior to increasing the RCS pressure above 1750 psia in accordance with Surveillance Requirement 4.5.2.d of the Technical Specifications. In addition, CV-306 is verified to be indicated open in the control room with power to the valve operator removed once per 12 hours, during **MODES** 1 through 4, in accordance with Surveillance Requirement 4.5.2.a. of the Technical Specifications. Because the single failure concerns of CV-306 are adequately addressed in the Updated FSAR and controls are maintained through the Technical Specifications, the valve design is considered acceptable.

Other major components of the system that are common include four LPSI loop isolation valves and a flow-measuring orifice. The motor operated loop isolation valves can also be manually operated, if necessary, and the flow orifice is a passive component. The high degree of LPSI reliability and availability is addressed in detail in Sections 6.3.4 and 6.3.5 of the Updated FSAR. A change is being proposed to clarify the actual plant configuration.

Recent PRA-based studies funded by the NRC and performed by Sandia National Laboratories produced results in draft form, at present, that conclude the failures of Calvert Cliffs shutdown cooling system as designed are well within the proposed NRC safety goal regarding core melt sequences probabilities. A final report will be published in the near future.

The second portion of this proposed change relates to Technical Specification 3/4 9.8.2, which specifies that:

> ". . . Two independent shutdown cooling loops shall be **OPERABLE**. . . ."
> Although the major components of the shutdown cooling loops, i.e., the LPSI pumps and shutdown cooling heat exchangers, are independent, some portions are not physically independent, but are common to both loops. The shutdown cooling loop temperature control valve CV-657 and shutdown cooling loop return isolation valve MOV-658 are common. In addition, the two shutdown cooling loop suction isolation valves 1(2)MOV-651 and 1(2)MOV-652 are common to the two loops for each unit. Therefore, we propose to delete the word "independent" from this specification to more accurately reflect the actual approved plant design.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed change has been reviewed against the standards set forth in 10 CFR 50.92 and has been determined to involve no significant hazards considerations, in that operation of the facility in accordance with the proposed amendment would not:

(i) involve an increase in the probability of occurrence or consequences of an accident previously evaluated.

The two proposed changes are simply clarifications of Technical Specifications which more closely reflect actual plant design.

(ii) create the possibility of a new or different kind of accident from any accident previously analyzed.

No new equipment, system alignments beyond those previously bounded by current Technical Specifications, or accident analyses are involved in the proposed change.

(iii) involve a significant reduction in the margin of safety.

The bases for the appropriate Technical Specifications are not being altered except to provide a description of actual plant design regarding the Low Pressure Safety Injection System.

CHANGE NO. 5 (BG&E FCR 84-157) Unit 2

Remove existing page 3/4 4-25 of Unit 2 Technical Specifications.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed change would delete the zero to two year curve from the Unit 2 Technical Specifications (Figure 3.4-2a). Unit 2 has been in commercial operation for approximately seven years and has surpassed the two Effective Full Power Years point of reactor vessel embrittlement. This proposed change is being processed to prevent any misunderstanding or misapplication of the superseded Reactor Coolant System pressure temperature limit curve.

Since the proposed change is administrative in nature, it does not affect plant operation or design in any way. It has been determined to involve no significant hazards considerations, in that in absence of the zero to two year curve, operation of the facility in accordance with the proposed amendment would not:

(i) involve any increase in the probability or consequences of any previously analyzed, or

(ii) create the possibility of an accident of a new or different kind from any accident previously analyzed, or

(iii) involve any reduction in the margin of safety.

CHANGE NO. 6 (BG&E FCR 83-007)

Remove existing pages 3/4 3-41 and 3-42 of the Unit 1 and 2 Technical Specifications and replace with attached marked up pages.

DISCUSSION

The NRC published guidance Technical Specifications for certain NUREG-0737, Post Accident Monitoring Instrumentation, in reference (e). The Baltimore Gas and Electric Company requested an amendment to its Operating Licenses for Calvert Cliffs Units 1 and 2 on February 24, 1983, by adding the Containment Wide Range Water Level Instruments to Technical Specifications Tables 3.3-10 and 4.3-10. The previous request for license amendment was submitted prior to enactment of 10 CFR 50.92 regarding determination of significant hazards considerations. With this submittal, we hereby request withdrawal of the February 24, 1983, submittal on Containment Water Level Technical Specifications and submit this proposed change to add the Containment Wide Range Water Level Instruments to the Technical Specifications.

The guidance Technical Specifications in reference (e) were deemed to be too restrictive when the Calvert Cliffs installation of Containment Water Level Instruments was reviewed in detail. The guidance Specifications specify two (2) channels as the required number of channels and one channel as the minimum channels **OPERABLE.** If BG&E adopted the guidance Technical Specifications, failure of one of the two transmitters could adversely affect plant availability. This is because the transmitters are located in the 20' 9" elevation in the containment. During normal power operation, these transmitters would be inaccessible due to neutron streaming and high gamma radiation levels. Reductions in power would reduce the dose rates, but a full unit shutdown would be required to obtain access for troubleshooting and repair.

The current Calvert Cliffs Technical Specifications for Accident Monitoring Instrumentation contain a column listing for MINIMUM channels operable, but no listing for REQUIRED NUMBER of channels. As stated in NUREG 0578, the containment wide range water level would provide "critical information (free liquid inventory in the containment building) in the diagnosis of the accident." Other instrumentation, such as the subcooled margin monitor, pressurizer water level, reactor vessel water level, and refueling water tank water level indication provide an alternate means of monitoring the free liquid inventory in the containment building. The instrumentation provides corroborative information to the operator to assess the current status of an accident which is progressing. It does not provide any automatic function, nor does its operation contribute to assumptions in any accident analyzed in the Updated FSAR for Calvert Cliffs.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

The proposed Technical Specification change has been reviewed against the standards provided in 10 CFR 50.92 regarding significant hazards considerations. This change constitutes an additional restriction, limitation, and condition not currently included in the Technical Specifications. Also, the Containment Wide Range Water Level Intruments provide only corroborative information to assist the operator in assessing the current status of an accident in progress. As such, the Technical Specification would not create the possibility of a new or different type of accident from any accident previously evaluated. Because a new restriction, limitation and condition is being added to the Technical Specifications, no reduction in the margin of safety would result. Therefore, the proposed Technical Specification change has been determined to involve no significant hazards considerations.

CHANGE NO. 7 (BG&E FCR 82-124)

Incorporate revised Table 3.6.1 as shown on attached marked-up pages 3/4 6-19 through 3/4 6-25 for Unit Nos. 1 & 2.

DISCUSSION

We have been involved in an effort to revise, upgrade, and standardize our piping and instrument diagram prints. Associated with this effort we have performed a walkdown of all affected systems to verify the accuracy of affected drawings. Table 3.6.1 as presently written lists the valve designations used on our contruction prints. The proposed change would modify this table to reflect the numbers used on our operational prints. This would result in less chance of error while performing critical valve line-ups by making Table 3.6.1 consistent with our operational procedures and drawings. The requested change is an administrative improvement to the current listing of identification numbers.

DETERMINATION OF SIGNIFICANT HAZARDS

Since this proposed change to the Technical Specifications is administrative and promotes clarity and consistency with operating procedures and operational drawings, it is consistent with examples of amendments considered not likely to involve significant hazards considerations as shown in Federal Register Notice 1487 dated April 6, 1983. For the same reasons, there is clearly no reduction in the margin of safety as a result of the proposed changes. Neither will the proposed changes result in any increase in the probability of consequence of any accident previously evaluated, nor will the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated.

CHANGE NO. 8 (BG&E FCR 83-70)

Remove old pages 3/4 4-13 and 3/4 4-14, and replace with attached marked-up pages 3/4 4-13 (with attached sheet), and 3/4 4-14 incorporating the change, which is applicable to both Unit Nos. 1 and 2.

DISCUSSION AND DETERMINATION OF SIGNIFICANT HAZARDS

This change modifies Technical Specification 3.4.6.1 and its associated Action Statement to achieve clarity concerning required action in response to inoperable leakage detection systems, and modifies Technical Specification 4.4.6.2 to achieve flexibility and consistency with the proposed leak detection systems Technical Specification.

We have on several occasions in the past, due to ambiguity in the action statement for Technical Specification No. 3.4.6.1, generated unnecessary Licensee Event Reports (LERs). The ambiguity centers around the words "and/or" in the fourth line of the action statement:

> ". . . With only two of the above required leakage detection systems **OPERABLE**, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours. . . ."

The containment atmosphere particulate monitor and the containment atmosphere gaseous monitor are separate means of leakage detection that use the same sample pump to draw the air sample through the detectors. In the event of pump failure, an installed redundant sample pump can be lined up within a short period of time, allowing repairs to be accomplished on the failed pump. Conservative interpretation of the action statement caused by the ambiguity precluded the possibility of lining up the spare pump within a reasonable time and resulted in the unnecessary prompt LERs. (Reference LER Nos. 80-46, 80-47, and 82-56, updated by reports dated 6-2-83).

Current interpretation of the Action Statement, as agreed upon in 1981 by the NRC Senior Resident Inspector at Calvert Cliffs, allows continued operation for 30 days with the particulate and gaseous detectors inoperable, provided grab samples of containment atmosphere are obtained and analyzed once per 24 hours. The proposed Technical Specification Action Statement provides the needed clarity and actions that envelope the currently approved interpretation. While the currently approved interpretation only addresses the case of inoperability of the two atmosphere monitors, a more reasonable, clear, and flexible approach shown in this change provides action requirements when any two of the three leakage detection systems are inoperable. Allowing two leakage detection systems to be inoperable for 30 days does not significantly decrease the margin of safety. When the following compensatory actions are taken, we believe there is no reduction in the margin of safety. These actions are the additional requirement of proposed TS 3.4.6.1 action statement, part b, for conducting the Reactor Coolant System water inventory balance of TS number 4.4.6.2.c once per 24 hours, in addition to the once per 24 hour containment atmosphere grab sampling requirement.

The proposed Action 3.4.6.1.c is written to require a return to operability of at least one leakage detection system with four hours to achieve consistency with the Action requirements of TS 3.4.6.2.b.

Numerous other systems and means for leak detection are described and discussed in Section 4.3 of the Updated FSAR and are used by our operators for leak detection purposes. They include: Containment Pressure and Temperature Indication, Pressurizer Pressure and Level Indication and Alarm, Containment Area Radiation Monitor Indication and Alarm, Containment Humidity Indicators, Reactor Coolant Drain Tank Level Indication, Reactor Coolant Make-up Water Flow Integrators, etc. Aside from the containment atmosphere grab samples, these additional leak detection capabilities provide further indication to the operator when any leakage detection systems are inoperable. The Updated FSAR notes a similar situation occuring during the initial plant start-up phase of operation, when both containment atmosphere radiation monitors are ineffective since there is little radioactivity in the reactor coolant system. The operator can properly identify excessive leakage and take appropriate action because he has available to him the various other systems which function normally. It is, therefore, reasonable to operate for a short period of time (i.e., 30 days) with one Reactor Coolant System leakage detection system available.

The second portion of this requested change makes several minor modifications to TS 4.4.6.2. First, containment atmosphere gaseous radioactivity is added as an alternative monitored parameter to meet surveillance requirement 4.4.6.2.a, providing additional capability and flexibility in demonstrating that Reactor Coolant System leakages are within the limits of TS 3.4.6.2. Second, a requirement to perform the containment grab sample analysis of Action Statement requirement 3.4.6.1 is added to surveillance requirement 4.4.6.2.a, if both containment atmosphere monitors are inoperable, ensuring the surveillance is consistent with proposed TS 3.4.6.1. Third, the qualifying phrase added to surveillance 4.4.6.2.b ensures consistency with current and proposed TS 3.4.6.1. The containment sump discharge frequency is determined by the operation of the containment sump level alarm. With the containment sump level alarm system inoperable, this surveillance cannot be performed. Thus, while TS 3.4.6.1 or proposed TS 3.4.6.1 would allow continued operation for 30 days, inability to perform the surveillance as presently written may require shutdown, even though several other means exist to demonstrate Reactor Coolant System leakages are within TS 3.4.6.2. Lastly, a cross reference to proposed TS 3.4.6.1, Action b, is added to surveillance requirement 4.4.6.2.c.

Our review of the Technical Specification bases and the Updated FSAR indicates the proposed change falls within their guidelines, and poses no reduction in the margin of safety. Study of the analyzed accidents in Chapter 14 of the Updated FSAR shows the change does not involve a significant increase in the probability or consequences of accidents previously analyzed, neither are any new or different accidents not previously analyzed generated by the proposed changes. The proposed change provides administrative clarification and flexibility and adds action requirements to existing Technical Specifications. The proposed change conforms with the paragraph 50.92 subsection (i) examples of amendments considered not likely to involve significant hazards considerations listed in page 14870 of the Federal Register Volume 48 dated April 26, 1983, since, as previously noted, the slight reduction in the margin of safety by allowing one **OPERABLE** leakage detection system for thirty days is compensated for by the increased required frequency of Reactor Coolant System Inventory Analysis in addition to the once per 24 hour containment atmosphere grab sampling action requirement.

SAFETY COMMITTEE REVIEW

This proposed change to the Technical Specifications and our determination or significant hazards have been reviewed by our Plant Operations and Safety and Safety and Off-Site Safety Review Committees, and they have concluded that implementation of this change will not result in an undue risk to the health and safety of the public.

FEE DETERMINATION

Pursuant to 10 CFR 170.21, we have previously forwarded BG&E Check No. B367760 in the amount of $150.00 to the NRC to cover the application fee for this request.

Very truly yours,

Arthur E. Lundvall Jr.

STATE OF MARYLAND :
: TO WIT:
CITY OF BALTIMORE :

Authur E. Lundvall, Jr., being duly sworn 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: Minnie L. Robinson
Notary Public

My Commission Expires: July 1, 1986

AEL/LES/tlm

cc: D. A. Brune, Esquire
G. F. Trowbridge, Esquire
D. H. Jaffe, NRC
T. Foley, NRC
T. Magette, DNR