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May 8, 1997

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Office of Nuclear Reactor Research

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Comments on Draft NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance"

REFERENCE: (a) Letter from Mr. C. H. Cruse (BGE) to NRC Office of Nuclear Reactor Research, dated March 27, 1997, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance

In Reference (a) above, Baltimore Gas and Electric Company submitted preliminary comments on the above referenced NUREG. We also stated our plans to submit additional comments prior to the final due date. The additional comments are provided in Attachment (1). A summary of our major concerns is provided below.

A large portion of this NUREG is dedicated to the comparison of the Individual Plant Examination numerical results between plants. Drawing conclusions without researching the design characteristics, assumptions, and quality of the Individual Plant Examinations being compared, could result in significant misinterpretation.

By publishing a document with conclusions which appear to represent comparisons of operating risk, the reader can easily be mislead. We therefore strongly recommend that comparison between plants be completely eliminated from NUREG-1560.

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

Very truly yours,

CHC/SJR/dlm

Attachment: (1) Comments on Draft NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance"

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ATTACHMENT (1)

Baltimore Gas and Electric Company

Comments on Draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance"

Calvert Cliffs Nuclear Power Plant
Units 1 & 2
May 8, 1997

ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY
COMMENTS ON DRAFT NUREG-1560 "INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE"

1. *Section 1.3, page 1-3, paragraph 3 states: "In examining the IPEs* Individual Plant Examination *and developing this report, the staff used the information as reported by the licensees. That is, the staff did not consider the quality (e.g., accuracy) of the analyses when determining the implications of the collective IPE results. Therefore, the staff used information from each IPE, even if a licensee's IPE/PRA was unacceptable (in part or overall), and no adjustment or modification was made." Page 1-3 paragraph 7 states: "The IPE Insights Program is based solely on licensee submittals, which summarize the IPE analyses and do not fully document all design characteristics, analysis assumptions, and results. This limits the ability to fully account for similarities and differences in results."*

   Baltimore Gas and Electric Company (BGE) Comment

   A large portion of this NUREG is dedicated to the comparison of the Individual Plant Examination (IPE) numerical results. Drawing conclusions without researching the design characteristics, assumptions, and quality of the individual IPEs being compared, may result in significant misinterpretation. Generic Letter 88-20 requested each licensee to " . . . perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission." NUREG-1560 should focus on the quality of systematic examinations rather than numerical results.

   The Combustion Engineering Owners Group (CEOG) Probabilistic Safety Assessment (PSA) Working Group (PSAWG) has found that the differences between the Combustion Engineering (CE) IPEs are due to many reasons, each requiring careful evaluation to determine whether the differences are due to modeling techniques or plant differences. When the differences are due to modeling, considerable effort is required to determine the most appropriate solutions in order to confirm the models.

   By publishing a document with conclusions that appear to represent comparisons of operating risk, the reader can easily be mislead. The statement, "Calvert Cliffs 1 & 2 have a CDF well above other CE plants, primarily because of a higher dependence on HVAC *[Heating, Ventilation and Air Conditioning]* and less capability (relative to other CE plants) to remove decay heat through the steam generators or feed-and-bleed cooling," implies an in-depth analysis between the CE plants, which is not true. See Comment 2 below.

2. *Page 3-47, first paragraph, states: "Calvert Cliffs 1 & 2 have a CDF well above other CE plants, primarily because of a higher dependence on HVAC and less capability (relative to other CE plants) to remove decay heat through the steam generator or feed-and-bleed cooling."*

   BGE Comment

   This conclusion appears to overly simplify the characterization of Calvert Cliffs' higher CDF. A significantly more detailed analysis is needed to determine the real reasons for the differences in the calculated CDF, including a review of many of the issues identified in Section 6.2, Characteristics of a Quality Probabilistic Risk Assessment.

Reasons for a higher CDF could include:

- Calvert Cliffs' comprehensive analysis of initiating events, including many support systems;

- Realistic and sometimes conservative success criteria;

- The inclusion of system dependencies, such as HVAC, service water and component cooling water make-up systems;

- Detailed system analyses which discovered many subtle dependencies;

- Extensive data collection which often resulted in higher failure rates and unavailability than generic data indicated;

- Plant-specific common cause analysis which often resulted in higher common cause factors than generic data; and

- A more rigorous flood analysis (see NUREG-1560, page 3-53).

These reasons have more to do with the quality of the analyses than the characteristics of the plant. This higher CDF is likely due to a complex combination of thoroughness (quality), conservatisms, and plant configuration. The more thorough the analysis, the more issues that are addressed and, therefore, the more likely that these issues contribute to core damage. And although BGE intends Calvert Cliffs' Probabilistic Risk Assessment (PRA) to be as realistic as possible, conservatisms were included. These conservatisms result from both the desire to build a defensible and thorough analysis, and the desire to complete such analysis in a cost- and time-effective manner. We also believe it prudent to use bounding analysis and/or bounding modeling techniques for issues which would otherwise result in excessive resources to resolve. Finally, plant configuration, including its design, operation, and maintenance, also plays an important role. It is, therefore, impossible to draw a conclusion about Calvert Cliffs' risk relative to other plants without first resolving the issue of consistency between the analyses.

The CEOG PSAWG is performing detailed reviews of the differences between the CE plant PSAs to determine the reasons for the differences between the CE plants' calculated CDFs. This process will require time to determine the reasons for these differences.

Baltimore Gas and Electric Company recommends that NUREG-1560 clearly characterize any comparison with the uncertainty of the consistency of approaches between the PSAs and the need to resolve such differences in a thorough and comprehensive manner. It is also important to note that the IPE submittals are a snapshot in time. Calvert Cliffs' PRA continues to improve and change as knowledge and understanding of the plant and issues are better understood.

A more appropriate statement would be:

Calvert Cliffs' higher calculated CDF is difficult to explain since its design is very similar to at least two other CE plants. Its model included a higher dependency on HVAC, which may be due to its treatment of this issue. The stated lesser capability (relative to other CE plants) to remove decay heat through the steam generators or feed-and-bleed cooling is not readily apparent and requires additional evaluation to resolve.

Specific discussion on Calvert Cliffs' HVAC and decay heat removal is provided below.

HVAC

Section 6.3, page 6-13, states: "A common example of a support system initiator that is not rigorously analyzed is the loss of Control Room heating, ventilation and air conditioning (HVAC) system." It also states: "However, the basis for eliminating some dependencies (primarily HVAC dependencies), in some cases, was qualitative and made without apparent calculation support."

In Calvert Cliffs' PRA, the HVAC systems were realistically modeled to provide the best understanding of the impact of the loss of Control Room and Cable Spreading Room HVAC, Switchgear Room HVAC, and Emergency Core Cooling System Room Cooling. The mere inclusion of these dependencies adds to accident sequences that contribute to CDF. This is an area where completeness might be the governing reason for the differences between Calvert Cliffs and other CE plants. In addition, ventilation room heat-up analyses are typically conservative. In the case of Calvert Cliffs Nuclear Power Plant's Switchgear Room HVAC, the 480 VAC and 4 kV busses were assumed failed when the action to recover ventilation failed.

Decay Heat Removal

Calvert Cliffs' Auxiliary Feedwater System (AFW) is comparable to that of other CE plants, as can be seen in Table 11.22. Its AFW System consists of two turbine-driven AFW pumps and one motor-driven AFW pump per unit. The other unit's AFW motor-driven pump may be cross-connected, as well. It is, therefore, not likely that a significant design or operational difference exists between Calvert Cliffs and other CE plants. However, Calvert Cliffs' PRA does address common cause failure between the pumps of the turbine-driven pumps and motor-driven pumps, and between the Unit 1 and Unit 2 pumps. Although it is uncertain as to whether this is a driving issue, it is likely a contributor.

It is true that Calvert Cliffs' power-operated relief valves (PORVs) are small, therefore limiting the time available to initiate feed-and-bleed for effective heat removal. However, some CE plants also have small PORVs or no PORVs. Again, it is not believed that this represents a reason for the difference in calculated CDF between Calvert Cliffs and other CE plants.

3. *Page 3-50, first bullet, states: "For example, the Calvert Cliffs units require both PORVs for successful feed-and-bleed (their PORVs are small). As a result, these units do not always credit feed-and-bleed as a possible heat removal system in response to a loss of main feedwater event because the RCS* [Reactor Coolant System] *will remain above the HPI* [high pressure injection] *shutoff head unless the reactor is tripped within 10 minutes."*

BGE Comment

The statement of tripping the reactor in 10 minutes is not true. At the time of the IPE submittal, we estimated that there was approximately 10 minutes from the loss of all feedwater (and its associated reactor trip) to the time the steam generators reach -350 inches. This assumes feedwater is lost at the time of the trip. Reaching -350 inches is the criteria the operators use to initiate feed-and-bleed. The time frame from initiating event to -350 inches was believed to be too short to allow the initiation of feed-and-bleed. The updated Calvert Cliffs PRA shows a small probability of success for this scenario of initiating feed-and-bleed, before reaching -350 inches.

4. *Page 3-50, second bullet, states: "Calvert Cliffs does not credit steam generator depressurization and use of condensate for heat removal because of the small single ADV* [atmospheric dump valve] *for each steam generator."*

BGE Comment

The updated Calvert Cliffs PRA now credits feeding the steam generators with the condensate system. The amount of improvement provided by the additional functionality is difficult to estimate due to the significant number of changes made in the model between the IPE submittal and the current update.

5. *Page 3-51, bottom bullet, states: "Configuration differences range from units with just one diesel per unit with a swing diesel between units (such as Calvert Cliffs) to two diesels per unit with additional backup capability existing or being added (such as Palo Verde). Calvert Cliffs is in the process of adding diesels to improve their onsite power reliability."*

BGE Comment

Note that the IPE submittal did not credit the two new diesels. See Comment 9.

6. *Page 3-52, fifth paragraph, states: "ATWS is a relatively low contributor to CDF (less than 5E-6/ry) for all but one plant in this group (Calvert Cliffs). In that single exception, the higher ATWS contribution reflects the analyses assumptions and the assessment that the plant has an unfavorable moderator temperature coefficient for a large fraction of the time (40%). The result is an analysis that does not credit the mitigating features mentioned above, assuming instead that an ATWS leads directly to core damage. This pessimistic stance leads to an ATWS CDF that is a factor of six greater than the next highest plant ATWS CDF among the CE plants."*

BGE Comment

Anticipated transient without scram (ATWS) recovery with a favorable moderator temperature coefficient (MTC) has been added to the model. We have also further evaluated the impact of an unfavorable MTC. Calvert Cliffs' updated PRA now assumes an unfavorable MTC 22% of the time when condenser vacuum is available, and 26% when it is not. This higher value is assumed due to the loss of turbine bypass heat removal, and the higher steam generator pressure due to the dependence on the steam generator safety valves.

Calvert Cliffs' ATWS contribution is now consistent with other CEOG plants.

7. *Page 3-53, first paragraph, states: "Calvert Cliffs performed a more rigorous analysis that accumulated the combined effects of nearly a dozen flood scenarios (rather than using a series of individual screening arguments) that collectively contribute to an internal flood CDF of about 1.5E-5/ry (a 6% contribution to total CDF). It is unclear whether the flooding contribution will be significantly higher for other plants if they use the screening approach employed by Calvert Cliffs."*

BGE Comment

This is an example of where the quality of the analysis appears to impact the bottom line number.

8. *Page 9-27, Section 9.3.2.2, first paragraph, states: "Only Calvert Cliffs identified a plant improvement in this area (RCP Seal LOCA). (Page 9-27). The implemented improvements in the CCW system should reduce the frequency of an RCP seal LOCA.*

BGE Comment

Reactor Coolant Pump (RCP) Seal Loss-Of-Coolant Accident (LOCA) importance has been significantly reduced in the updated Calvert Cliffs PRA model. This is due to more realistic modeling of the high pressure safety injection pumps' dependency on component cooling water. For RCP Seal LOCA break sizes, thermal-hydraulic analysis indicates that the recirculated Reactor Coolant System temperature will not exceed the high pressure safety injection pump seal temperature (the most limiting constraint) for which component cooling water is required. Therefore, the dominant IPE sequence of loss of component cooling water challenging both the RCP seals and the high pressure safety injection pumps is no longer applicable.

9. *Page 11-81, last paragraph, states: "The former diesel configuration at Calvert Cliffs 1 & 2 was one diesel dedicated per unit with one swing diesel between plants. The enhanced design credited in the submittal included the addition of two diesels at the site, one for each plant."*

BGE Comment

Calvert Cliffs' IPE submittal did not credit the new diesels in its results. A sensitivity study, Calvert Cliffs Nuclear Power Plant IPE Summary Report, page 3.4.1-33, was included which estimated the reduction of the overall CDF at 17.7%. The statement above implies the IPE results include this addition.

10. *Page 11-83, Section 11.3.2.5, second paragraph, states: "Examination of the Calvert Cliffs 1 & 2 submittal considers the fraction of time that the reactor cores at these units have an unfavorable moderator temperature coefficient (MTC), discussed below, to be about 40%, significantly above that normally seen in PWR* [pressurized water reactor] *PRAs. Whether this is a true design difference or a function of pessimistic analysis, can not be determined just from the submittal. Partially as a result of this consideration, the Calvert Cliffs 1 & 2 analyses did not model the mitigation features mentioned above at all. This submittal assumes that failure to scram leads to core damage, a pessimistic assumption. Even with this bounding approach, ATWS barely contributes 10% to the total plant CDFs for both plants.*

BGE Comment

See Comment 6.

11. *Chapter 14, Attributes of a Quality PRA, page 14-2, first paragraph states: "This chapter is not intended to prescribe guidelines for how to perform a quality PRA. This chapter is only intended to provide the attributes of a quality PRA such that a reader can judge if a specific PRA (e.g., IPEs) is a quality PRA.*

BGE Comment

Since Chapter 14 is not meant to be used to establish IPE quality requirements nor for establishing the quality requirements for future PRA applications, BGE recommends deleting it from NUREG-1560.

Chapter 15, Comparison of Individual Plant Examinations To A Quality Probabilistic Risk Assessment, uses Chapter 14's attributes of a quality PRA. Since these attributes do not represent those which were required to meet the intent of Generic Letter 88-20, BGE also recommends deleting this chapter.

12. *Chapter 16, Safety Goal Implications, first paragraph states: Chapter 16 provides a more in-depth discussion including the approach adopted to infer how the IPE results were compared to the quantitative health objectives.*

BGE Comment

In SECY-90-104, "Role of Individual Plant Examinations (IPE) In Assessing Industry Status with Respect to the Commission's Safety Goal Policy," dated March 20,1990, the NRC staff stated:

> "Based on the significant additional resources that would be required to make a meaningful comparison of the IPE results with the Safety Goal Policy Statement and the potential problems associated with using the as-submitted IPE data, the staff recommends that no direct comparisons be made unless the IPEs are reviewed to a greater level of detail than currently planned. Therefore, no guidance or criteria for such a comparison are being proposed at this time. This recommendation is consistent with original purpose of the IPE which was to ensure a systematic evaluation of each plant for vulnerabilities to severe accidents. Enclosure 2 provides additional background on the purpose of the IPE as it has been communicated to licensees. However, if the Commission desires to invest the resources required for such a review, the staff will prepare a plan, including guidance and criteria, for Commission review and an assessment of the impact on other planned work."

It is BGE's understanding that a review of greater detail did not occur. Therefore, BGE recommends that the direct comparison of IPE's performed in Chapter 16 be removed from NUREG-1560.