BALTIMORE
GAS AND
ELECTRIC

CHARLES CENTER • P.O. BOX 1475 • BALTIMORE, MARYLAND 21203-1475

GEORGE C. CREEL
VICE PRESIDENT
NUCLEAR ENERGY
(301) 260-4455

March 7, 1991

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
Request for Amendment

REFERENCES:

(a) Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated December 20, 1989, Request for Unit 1 Amendment

(b) Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated March 6, 1990, Issuance of Unit 1 Amendment No. 140

(c) Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated October 22, 1990, Request for Unit 2 Amendment

(d) Letter from Mr. D. G. McDonald, Jr. (NRC), to Mr. G. C. Creel (BG&E), dated December 18, 1990, Issuance of Unit 2 Amendment No. 131

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 these proposed changes to the Technical Specifications.

DESCRIPTION

The proposed amendment would revise the Technical Specifications for Unit 2 to require Safety Injection Tank operability throughout MODE 3, and revise the supporting surveillance requirements for both Unit 1 and Unit 2 to be consistent with the revised MODE applicability. In addition, editorial changes would be made to the titles of related Technical Specifications to more accurately reflect their applicability.

A001 Cert NO PA83302683
1/1

BACKGROUND

The safety injection tanks (SITs) provide water to begin the re-flooding of the core following a reactor coolant system (RCS) blowdown to minimize core damage until the safety injection pumps can provide adequate water for reactor cooling. The tanks are designed to inject large quantities of borated water into the RCS immediately following a large pipe break. In such an event, the water begins to re-flood and cool the core, thereby limiting clad-heatup and metal-water reaction until the safety injection pump flow is established. During normal pressure and temperature operating conditions, Technical Specification 3.5.1 ensures that the SITs are available to perform this function.

Reference (a) requested a revision to Unit 1 Technical Specification 3.5.1 to require the SITs to be operable during all of MODE 3. The revision was approved and issued as an amendment by Reference (b). Prior to the amendment, the SITs were required to be operable in MODE 3 only when the RCS pressure was at or above 1750 psia. However, evaluation of high pressure safety injection pump controls to prevent overpressure conditions at low operating temperatures, which were also requested and approved by the above referenced documents, indicated that the water in the SITs would be necessary for mitigation of loss of coolant accidents (LOCA) occurring during MODE 3 at RCS pressures below 1750 psia. The evaluation is also applicable to Unit 2, and BG&E therefore requests a revision to expand the applicability of Unit 2 Technical Specification 3.5.1 to include all of MODE 3 as previously approved for Unit 1.

In addition, Surveillance Requirement 4.5.1.d requires verification within four hours prior to exceeding 1750 psia that the SIT isolation valve is open. A change to the applicability to include MODE 3 below 1750 psia requires a corresponding revision to this surveillance requirement so that the surveillance continues to be conducted prior to entering the applicable operating conditions. We therefore request a revision to both Units 1 and 2 Technical Specification 4.5.1.d to require verification that the SIT isolation valve is open during startup within four hours prior to entering MODE 3.

Finally, the titles of Units 1 and 2 Technical Specifications 3/4.5.2 and 3/4.5.3 do not accurately reflect the applicability of these Specifications and are requested to be revised. Currently, the titles imply that Technical Specification 3/4 5.2 is to be used when the RCS temperature is at or above 300° F and Technical Specification 3/4.5.3 is to be used when the RCS temperature is below 300°F. This would correspond to the change between MODEs 3 and 4, but the applicability statements of the Technical Specification do not agree with this implication. The Technical Specification applicability actually changes in MODE 3 when the pressurizer pressure rises above 1750 psia. Titles that reflect the requirements of the limiting condition for operation will provide easier reference that does not conflict with the Technical Specification. Also, the title of Unit 2 Technical Specification 3/4.5.4 is requested to be revised in the Index to match the title of the actual Technical Specification. These title changes are considered to be administrative in nature.

REQUESTED CHANGE

Change page 3/4 5-1 of the Unit 2 Technical Specifications and change pages V, 3/4 5-2, 3/4 5-3 and 3/4 5-6 of both the Unit 1 and Unit 2 Technical Specifications as shown on the marked-up pages attached to this transmittal.

JUSTIFICATION

This change supports previous changes to the Unit 2 Technical Specifications which were submitted and approved in References (c) and (d) and the previous change to Unit 1 Technical Specification 3.5.1 which was submitted and approved by References (a) and (b). The evaluation supporting those changes indicates that the inventory of the SITs may be required to refill the reactor coolant system following a LOCA if the high pressure safety injection (HPSI) pumps are not available for automatic operation. Since Technical Specification 3.5.3 allows restriction of the use of HPSI pumps when the RCS temperature is below 350^{O}F, and MODE 3 includes temperatures down to 300^{O}F, the SITs are required to be operable during these operating conditions. Further, since operability is dependent on a current surveillance per Technical Specification 4.0.4, Surveillance Requirement 4.5.1.d should be revised to occur prior to entry into the applicable MODE. Each of these changes is conservative with respect to the current Technical Specifications and the plant administrative controls have been revised to require safety injection tank operability prior to entering MODE 3 on a startup.

The LOCA evaluation identified above was performed to determine the SIT pressure and volume needed to re-flood the core and the minimum time available to initiate additional RCS make-up flow to maintain core heat removal. The calculation for Unit 1 and Unit 2 determined that the operator has at least 18 minutes to initiate high pressure safety injection flow to maintain long-term core heat removal. This calculation was based on the following assumptions:

a. Reactor coolant temperature is less than or equal to 350^{O}F.

b. No limitations are applied on the potential break size or location. A full spectrum of break sizes was considered. The worst case was determined to be a large cold leg break, such that one SIT spills from the break and consequently provides no usable injection flow.

c. All the initial RCS inventory is instantaneously discharged from the break following a large break LOCA (i.e., no credit for residual inventory following blowdown).

d. Fission product decay heat is based on 120% of the 1971 ANS proposed standard per the requirements of 10 CFR 50, Appendix K. The decay heat corresponds to four hours following reactor shutdown from long-term operation at full power.

e. The core power distribution corresponds to pre-shutdown operation with a peak linear heat rate at 85% core height and at the Technical Specification limit of 15.5 kw/ft.

f. Adiabatic heat-up of the fuel hot-spot (i.e., no credit for steam cooling) once the collapsed liquid level in the core drops below the hot-spot location.

g. The SITs were evaluated at the minimum and maximum Technical Specification limits for allowable pressure and volume.

The evaluation shows that, by utilizing SIT availability in lieu of automatic HPSI pump starting, adequate protection against the effect of a LOCA while at an RCS temperature of 350^{O}F or less in MODE 3 is provided to allow time for the operator to manually initiate HPSI pump flow.

The actions necessary to manually initiate HPSI in this event include:

- verifying the indicated flow path for the selected HPSI pump (including a verification that the loop isolation valves are shut)
- starting the selected HPSI pump (using the pump handswitch; turning it from its "pull-to-lock" position to the "start" position -- spring return to center)
- throttling open HPSI header loop isolation MOV as necessary to control flow to maintain proper pressure, pressurizer level, and subcooling.

If a LOCA were to occur, the operators would use Abnormal Operating Procedure (AOP)-2A, "Excessive Reactor Coolant Leakage" to implement the above actions. This procedure addresses a minor leak, a major leak, and a loss of coolant in MODEs 3, 4, 5, and 6. The goal of this procedure is to address a loss of coolant as well as to ensure that the reactor vessel is not subject to overpressure at low temperatures. The procedure is symptom based and thus addresses many possible LOCA scenarios.

The minimum time available for operator action determined by the evaluation is based on a large pipe break with an instantaneous blowdown of the entire reactor coolant inventory and an instantaneous refill of the vessel up to the break location, using the SIT inventory. For smaller pipe breaks, the blowdown and re-flood time is longer, which provides more available time for operator action. A review of the supporting calculations for the small break LOCA analyses from full power described in the Final Safety Analysis Report (FSAR) shows that at least 300,000 lbm of primary coolant is discharged through the break before the break location is uncovered. The evaluation showed more than 18 minutes available for the worst case. Since the requested amendment affects only operating conditions below 350^{O}F, the actual system pressure will be considerably lower and the blowdown times much longer.

During this time, the operator would have positive indication of a LOCA in progress due to the reactor coolant that would be released to the containment. This would be detected by containment pressure increase, sump level increase or a rise in containment temperature. The procedures provide the operator with specific instructions which ensure that the existing LOCA analyses results are not exceeded, regardless of break size. Therefore, the minimum of 18 minutes is sufficient time for the operator to identify and appropriately respond to plant conditions.

The Surveillance Requirement change will assure that the SIT limiting condition for operation which requires an open isolation valve is met prior to entering the applicable operational conditions (MODEs 1, 2 and 3).

The title changes are considered to be administrative in nature. The titles are not part of the limiting condition for operation nor the surveillance requirements and are provided only as an aid to help the user locate the appropriate information. The changes will improve this aid by more closely reflecting the applicable operational conditions of the Technical Specifications for which they are provided.

DETERMINATION OF SIGNIFICANT HAZARDS

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendment:

(1) *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

This change to the method of operation will increase the availability of the SITs to assure sufficient water is available to provide cooling in the event of a LOCA during the applicable reduced pressure and temperature operating conditions. An evaluation was performed which concludes that the applicable consequences of the LOCA events described in the FSAR bound the results for LOCA events occurring under the subject conditions. This change does not involve equipment which was considered as an initiator for a previously evaluated accident. Also, the administrative title changes will have no effect on the safety analyses or the limiting conditions for operation. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

This change to the administrative control will potentially impact the loss of coolant event mitigation features, but no changes are being made in the plant hardware, and the change, therefore, does not introduce any new accident initiators. Also, the administrative title changes will have no effect on the safety analyses or the limiting conditions for operation. Therefore, the change does not create the possibility of a new or different type of accident from any accident previously evaluated.

(3) *Would not involve a significant reduction in a margin of safety.*

This change to the method of operation assures that sufficient cooling water is available to mitigate a LOCA. An evaluation was performed which concludes that the results of the LOCA events described in the FSAR bound the results for LOCA events occurring under the subject conditions (MODE 3 LOCA with $T_{RCS} < 350^{o}F$). No changes are being made in the plant hardware considered in this analysis. Also, the administrative title changes will have no effect on the safety analyses or the limiting conditions for operation. Therefore, the change does not involve a significant reduction in the margin of safety.

SCHEDULE

This change is requested to be approved and issued by May 31, 1991. However, issuance of this amendment is currently not identified as having an impact on outage completion or continued plant operation. Further, the additional requirements for safety injection tank operability have been conservatively incorporated into plant operating and surveillance procedures and are currently being used for operation of the plant.

SAFETY COMMITTEE REVIEW

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

Very truly yours,

G C Creel

STATE OF MARYLAND :
: TO WIT :
COUNTY OF CALVERT :

I hereby certify that on the 7th day of March, 1991, before me, the subscriber, a Notary Public of the State of Maryland in and for __________, personally appeared George C. Creel, 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 information 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 information on behalf of said Corporation.

WITNESS my Hand and Notarial Seal:

Michelle D. Hall
Notary Public

My Commission Expires:

February 2, 1994
Date

GCC/ERG/erg/dlm

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
L. E. Nicholson, NRC
R. I. McLean, DNR
J. H. Walter, PSC