UNITED STATES NUCLEAR REGULATORY COMMISSION

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-53, issued to Baltimore Gas and Electric Company (the licensee), for operation of the Calvert Cliffs Nuclear Power Plant, Unit No. 1 located in Calvert County, Maryland.

The amendment would: (1) provide changes to the Appendix A Technical Specifications (TS) to allow Cycle 7 operation, (2) revise the TS Limiting Conditions for Operation and Surveillance Requirements for the Auxiliary Feedwater System, (3) delete the Limiting Condition for Operation and Surveillance Requirements for certain post-accident monitoring instruments, (4) change the TS for the indicating ranges of the remote shutdown monitoring instrumentation (5) change the surveillance requirements for Subchannels A-3 and B-3 of the automatic actuation logic for the Containment Spray Actuation System, and (6) provide Limiting Conditions for Operation and Surveillance Requirements for the Containment Vent Isolation Valves (MOV-6900 and MOV-6901).

The above proposed changes to the TS are in accordance with the licensee's application dated August 22, 1983, as supplemented September 1, 1983, September 16, 1983, and two supplements dated September 20, 1983.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for conclusions regarding "no significant hazards considerations" by providing examples (48 FR 14870). In this regard, the TS changes resulting from the reload analysis for Cycle 7 operation conform to example (iii) in 48 FR 14870 which states that no significant hazards considerations are expected to exist if operation of the facility in accordance with the proposed amendment involves:

> For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

Accordingly, on this basis, the NRC staff proposes to determine that the TS changes associated with Calvert Cliffs Unit 1 Cycle 7 operation involve no significant hazards considerations. The remaining proposed changes to the TS do not conform to any of the examples provided in 48 FR 14870; the basis for the proposed no significant hazards consideration determination is presented herein.

Changes in the TS, associated with the Auxiliary Feedwater System, are required to reflect modifications to this system. These system modifications include the addition of an independent auxiliary feedwater train with a motor powered pump. A cross-connect is also provided between the Unit 1 and Unit 2 Auxiliary Feedwater Systems. In addition, the pre-set injection valves will be replaced with flow modulating valves.

In order to assure operability of the modified system, a period of post-startup testing will be required. During this period, the automatic start and flow modulating features of the auxiliary feedwater system may be inoperable. Accordingly, the licensee has requested a special test exception to be inserted in the TS to address startup testing of the auxiliary feedwater system.

With the automatic start and flow modulating features inoperable, the auxiliary feedwater system would consist of two, manually-started, steam driven, auxiliary feedwater pumps capable of supplying water to each of two steam generators. This configuration is identical to the auxiliary feedwater system in the as-licensed plant configuration which required operator action to initiate and adjust auxiliary feedwater flow. The licensee has performed safety analyses to evaluate the time required for operator response in situations requiring auxiliary feedwater flow.

The safety analyses performed to evaluate the effects of auxiliary feedwater flow deviations show that no flow for 10 minutes after the most severe undercooling transient, or runout flow for 10 minutes after the most severe overcooling transient, is acceptable. Consequently, adequate time would be available to manually initiate and control auxiliary feedwater flow in the event that either of these transients were to occur during the proposed 30 day period. A period of 10 minutes is considered adequate for operator response. In order to assure that operators are aware of the status of the auxiliary feedwater system and the possibility that operator control may be required, the licensee will brief the operators on at least a weekly basis concerning the condition of the auxiliary feedwater system. Since operator response can adequately substitute for the automatic start and flow modulating features of the modified auxiliary feedwater system, accidents which result in auxiliary feedwater flow will not be more severe during the 30 days following Cycle 7 startup, during which the test exception will be applicable. The licensee has proposed additional changes to the Limiting Conditions for Operation (LCOs) for the Auxiliary Feedwater System. These proposed changes to the LCOs are equivalent to existing LCOs in that, should an auxiliary feedwater pump be inoperable (among a total complement of two steam driven pumps, and one motor driven pump) the proposed LCO would require that two auxiliary feedwater pumps would be operable within 72 hours. The existing LCO is based upon two steam driven auxiliary feedwater pumps and, in the event that one pump is inoperable, both pumps must be operable within 72 hours. A second proposed LCO would allow two of the three auxiliary feedwater pumps to be inoperable for up to 72 hours provided that the auxiliary feedwater cross-connect, and the Unit 2 motor driven auxiliary feedwater pump, are verified to be operable. This proposed LCO is at least equivalent to the existing LCO in that a total of two auxiliary feedwater pumps are required to be operable; in this case one Unit 1 and one Unit 2 auxiliary feedwater pump would be required.

A change to the LCO is also proposed to address the changing of operational modes (e.g. from startup to power operation) with an inoperable auxiliary feedwater pump. Since no additional safety concerns are associated with changing operational modes with an inoperable auxiliary feedwater pump, no prohibition on such mode changes is appropriate.

The Surveillance Requirements have been proposed for modification to reflect changes to the auxiliary feedwater system. Prior to modification, auxiliary feedwater flow was controlled via preset flow control valves.

These valves were the subject of surveillance to assure that their position would permit a preset flow value. The TS required reverification of auxiliary feedwater flow control valve alignment should the auxiliary feedwater flow control valves be repositioned. Since these preset flow control valves have been replaced by air-operated modulating valves, this surveillance is no longer applicable. In place of this requirement, BG&E has proposed to perform a flow verification test each 18 months which would assure that each auxiliary feedwater pump delivers flow upon automatic initiation. In addition, BG&E has proposed a test of the dynamic head for the motor driven auxiliary feedwater pump. This proposed test would be conducted every 31 days to assure a dynamic head of at least 3100 ft on recirculation flow. This test frequency is consistent with dynamic head test currently required for the steam driven auxiliary feedwater pumps.

The above proposed changes to the Limiting Conditions for Operation and Surveillance Requirements for the Auxiliary Feedwater System do not allow a decrease in the operability or effectiveness of surveillance of the auxiliary feedwater system. Based upon the above, the proposed changes to the TS for the Auxiliary Feedwater System maintain or improve the readiness of this system to respond to emergency conditions and, therefore, the staff proposes to determine that the proposed changes do not involve significant hazards considerations.

BG&E has requested that certain post-accident monitoring instrumentation be deleted from the TS. Although these instruments would not be physically removed from their installed locations, they would no longer be the subject of TS requirements. These instruments include:

(1) Power Range Nuclear Flux Monitor - This instrument was judged to be unnecessary for post-trip reactor monitoring. The function of post-accident flux monitoring is adequately performed by the Wide Range Logarithmic Neutron Flux Monitor, which includes the power range flux indication and is addressed in the TS.

(2) Reactor Coolant Total Flow - Reactor coolant system total flow is considered a non-essential channel of instrumentation in the post-trip condition because of the availability of reactor coolant system (RCS) temperature indication, RCS subcooled margin and Reactor Coolant pumps status. In the absence of adequate core flow, RCS temperature and subcooled margin are indirect indicators of core flow conditions and provide adequate display to ensure appropriate actions are initiated by operations personnel to recover from any abnormal conditions existing in the post-trip condition.

Neither the Total Reactor Coolant Flow nor the Power Range Nuclear Flux Monitor is referenced in the emergency operating procedures. We therefore conclude that the proposed deletion of operability and surveillance requirements for the power range nuclear flux monitor and reactor coolant total flow, post accident, instrumentation would not decrease the ability of the licensee to detect and correct abnormal post-accident conditions. Accordingly, the staff proposes to determine that the proposed deletion of the TS requirements for the subject post-accident monitoring instrumentation does not involve a significant hazards consideration.

The remote shutdown instrumentation allows the reactor operator to monitor key safety parameters from outside the control room. No automatic safety features are actuated from the remote shutdown instrumentation. Changes to Reactor Coolant Cold Leg Temperature, Steam Generator Pressure and Level and Wide Range Neutron Flux instrumentation as described in the applicable LCO, have been proposed.

With regard to Reactor Coolant Cold Leg Temperature (RCS)T_c, NUREG-0737, Item II.F.2 requires the installation of instrumentation for detection of inadequate core cooling. Accordingly, subcooled margin monitors have been installed utilizing, among other inputs, existing temperature measurement channel inputs. The initial installation of the subcooled margin monitors utilized (RCS)T_c narrow range temperature inputs. The detection range of the subcooled margin monitors was limited by the measurement range provided by the cold leg temperature measurement channels, required by the LCO to be from 0-600°F. The design of the subcooled margin monitors precludes providing any representative engineering data at temperature measurement ranges less than 212°F (boiling point of water) or greater than 705°F (critical point of water). The guidance contained in Reg. Guide 1.97 suggested modifications to provide temperature measurement ranges of 150°F to 750°F. Since temperature measurement ranges below the boiling point or above the critical point of water provided no useful input to the subcooled margin monitors and produce the undesirable effect of greater inaccuracy over the extended measurement ranges, a limited T_c measurement range between the two points that would provide useful input to the subcooled margin monitors, 212°F to 705°F, was selected.

Steam generator level measurement has been modified to provide an extended range of level indication. It had previously indicated level from -116 to +63.5 inches. The modification increases the range to -401 to +63.5 inches as required by the proposed LCO. The steam generator level measurement channel provides indication at the remote shutdown panel (2C43) and does not provide any control functions at the panel. The additional level range provides the operator with more representative information of actual steam generator inventory. Finally, the range of the "Wide Range Neutron Flux" instrumentation has been increased, as indicated in the proposed LCO, from .1 counts per second (cps) - 150% power to .1 cps - 200% power.

As indicated previously the remote shutdown instrumentation is provided for monitoring purposes and does not provide inputs for automatically actuated equipment; therefore, the changes as reflected in the proposed LCO do not change the course or severity of any analyzed accidents. Moreover, since the proposed changes to the instrumentation ranges provide equivalent or improved information, the usefulness of this instrumentation to provide post-accident information has not been degraded. On this basis, the staff proposes to determine that these proposed changes to the LCO for remote shutdown instrumentation do not involve significant hazards considerations.

The licensee has proposed a change in the Surveillance Requirements for subchannels A-3 and B-3 of the Containment Spray Actuation System (CSAS). The purpose of CSAS subchannels A-3 and B-3 is to isolate the service water supply to the spent fuel pool coolers on indication of high containment pressure; the licensee has proposed a plant modification which would move these functions to other CSAS actuation channels. CSAS subchannels A-3 and B-3 as reconstituted would perform the following functions on indication of high containment pressure: trip the main feedwater, condensate booster, and heater drain pumps and close the main steam and feedwater isolation valves. These automatic actions would isolate the main feedwater system in the event of a steamline break thus preventing overpressurization of the containment. The same automatic features would also be added to the Steam Generator Isolation function.

The present surveillance requirement for subchannels A-3 and B-3, requiring monthly testing, is no longer appropriate for the reconstituted subchannels A-3 and B-3. Such testing, during reactor operation, would result in a reactor trip due to closure of the main steam isolation valves since the MSIVs cannot be bypassed during testing. The licensee has proposed that CSAS subchannels A-3 and B-3 be tested every 18 months during plant shutdown, which is appropriate considering the design of the associated equipment and the need to prevent unnecessary reactor scrams. Since the addition of automatic tripping of the pumps in the feedwater trains results in a less severe plant response to a main steamline break, the NRC staff proposes to conclude that the associated changes to the TS involve no significant hazards considerations.

The licensee has proposed changes to the TS to address LCOs and Surveillance Requirements for Containment Vent Isolation Valves. These valves are presently designated as Hydrogen Purge Outlet Valves (MOV-6900 and MOV-6901). These valves are presently non-automatic, motor operated, valves that are required by the TS to be maintained in the closed position during reactor operation (Modes 1, 2, 3 and 4). A modification to these valves would add an automatic isolation signal to close these valves on a Safety Injection Actuation Signal (SIAS). The licensee has proposed that the redesignated Containment Vent Isolation Valves be required to close automatically in less than 20 seconds as verified by periodic testing. Under reactor operating conditions Modes 1, 2, 3 and 4 the proposed LCO would require the valves to be maintained in the closed position and doubly isolated. In this case, double isolation includes removal of motive power (supply breaker open) and the use of a key-locked switch. Monthly surveillance would assure that the valves remain closed and doubly isolated. In addition, during core alterations or movement of irradiated fuel within containment, the licensee has proposed TS to require that these valves remain closed.

The proposed TS are consistent with other existing Calvert Cliffs Unit 1 TS for containment purge valves. In addition, both existing and proposed TS for valves MOV-6900 and MOV-6901 require these valves to be closed during reactor operation and during refueling operations; thus, the proposed TS would be at least as restrictive as existing TS. Accordingly, the NRC staff proposes to determine that the TS changes associated with the modified MOV-6900 and MOV-6901, the Containment Vent Isolation Valves, involve no significant hazards considerations.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Comments should be addressed to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

By NOV 14 1983 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Request for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR §2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to James R. Miller: petitioner's name and telephone number;

date petition was mailed; plant name; and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to James A. Biddison, Jr., General Counsel, G and E Building, Charles Center, Baltimore, Maryland 21203, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated August 22, 1983 as supplemented September 1, 16 and two dated September 20, 1983 which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the local public document room located at the Calvert County Library, Prince Frederick, Maryland.

Dated at Bethesda, Maryland, this 6th day of October, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by J. R. Miller

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

bcc: See next page

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