BALTIMORE GAS AND ELECTRIC

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VICE PRESIDENT
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April 30, 1990

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
Technical Specification Change Request:
Spent Fuel Cask Handling Crane

REFERENCE: Letters from A. E. Lundvall, Jr. (BG&E) to D. G. Eisenhut (NRC), dated January 4, 1982 and March 1, 1982, Control of Heavy Loads

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, in accordance with 10 CFR 50.90 and 50.91 to allow movement of a spent fuel shipping cask into our spent fuel pool. The cask will be used to ship selected fuel rods in support of hot-cell work sponsored by EPRI. The primary purpose of the EPRI program is to provide high burnup fuel performance data that will support operation at extended fuel burnups. Data will be obtained relating to corrosion and hydrogen pick up in Zircaloy-4 fuel rod cladding and assembly structural components, effects of extended operation (including corrosion and fluence) on mechanical properties of Zircaloy-4 cladding and components, and fission gas release and its relationship with UO_2 pellet microstructure and fuel rod internal pressure. In order to meet schedules for availability of the spent fuel shipping cask and for completion of the hot cell examination, the license amendment would need to be issued by July 1, 1990.

Besides permitting a shipment of fuel rods in support of the EPRI program, the proposed change will also allow a reactor vessel weld surveillance capsule to be removed from our spent fuel pool using a shipping cask.

DESCRIPTION OF CHANGE

A change to Technical Specification 3/4.9.13, "Spent Fuel Cask Handling Crane" is proposed to allow movement of a spent fuel shipping cask within a cask length of fuel within the pool. The movement is to be allowed only if the boron concentration of the spent fuel pool is greater than or equal to 1000 ppm AND the following criteria are met by all assemblies within one cask length radius of the pathway: 1) initial

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enrichment less than or equal to 4.1 w/o U-235; 2) burnup greater than or equal to 28,000 MWD/MTU; and 3) greater than 440 days elapsed from the shutdown of the last operating cycle in which the assembly was present in the core. The change is applicable only for a shipment of spent fuel rods supporting EPRI sponsored hot-cell work and for a shipment of a reactor vessel weld surveillance capsule in support of our life cycle management program.

The proposed change (attached) will add the following footnote to Technical Specification 3/4.9.13:

> "These conditions are modified to permit shipping cask travel to and from the cask pit in the presence of fuel within one cask length radius of the pathway provided the boric acid concentration in the spent fuel pool is greater than or equal to 1000 ppm AND the following criteria are met by all assemblies within one cask length radius of the pathway: 1) initial enrichment less than or equal to 4.1 w/o U-235, 2) Burnup greater than or equal to 28,000 MWD/MTU, and 3) greater than 440 days elapsed from the shutdown of the last operating cycle in which the assembly was present in the core. Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly not satisfying the above criteria shall be demonstrated OPERABLE within 24 hours prior to using the crane for moving a cask within one cask length of fuel assemblies meeting the above criteria. These modifications are applicable only for the shipment of fuel rods supporting the EPRI sponsored hot-cell work and for the shipment of a reactor vessel weld material surveillance capsule."

REASON FOR CHANGE

Compliance with Technical Specification 3/4.9.13 would require movement of fuel out of rack locations located within one cask length of the spent fuel shipping cask pit whenever a cask is brought into the pit. Figure 1 (attached) shows the area of the spent fuel pool affected by this requirement. Because of the amount of spent fuel stored in the pool, which has been increased because of the offload of the Unit 2 Cycle 8 core into the pool, there is presently insufficient space to comply with the requirements of Technical Specification 3/4.9.13. Unit 2 core onload and modifications to fuel handling equipment which would provide access to additional storage locations cannot occur until later this year. As described below, cask shipments must occur before those evolutions can take place. Consequently, we cannot comply with Technical Specification 3/4.9.13 and accommodate the cask shipments. Therefore, a change to the Technical Specification is required to allow a one-time shipment of fuel rods supporting EPRI sponsored hot-cell work and a one-time shipment of a reactor vessel weld material surveillance capsule supporting our life cycle management program.

The surveillance capsule contains neutron irradiation dosimetry data and must be removed from the spent fuel pool and analyzed within the next year. Otherwise, important data needed to monitor reactor vessel fracture toughness properties will be lost through radioactive decay. We have considered other means of removing the capsule but all would result in increased radiation exposure to the personnel involved. Thus, the use of a shipping cask to make this shipment is preferred.

With regard to the hot cell program, EPRI has stated that the fuel rod shipment to the hot cell must be scheduled for the summer of 1990. The other schedule constraint is a window for shipping cask availability which is currently from June to mid July 1990. The tight restriction on availability is due to DOE's heavy use of all available casks. In consideration of the schedule constraints, we are scheduling a cask shipment for early July 1990.

DETERMINATION OF NO SIGNIFICANT HAZARDS

This proposed change has been evaluated against the standards 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 a significant increase in the probability or consequences of an accident previously evaluated; or

The referenced letters considered a heavy load drop in response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and while the proposed Technical Specification change will not increase the probability of the drop, it will affect the consequences. Our NUREG responses reported that no fuel would be impacted by the drop due to restrictions placed on locating fuel within the arc shown in Figure 1. The Basis for Technical Specification 3.9.13 relates to the fact that when fuel storage within one cask length of the pathway is prohibited, a dropped cask will not cause fuel damage (the only safety consequences of fuel damage being offsite dose) or result in a critical array. Because the proposed change would allow a dropped cask to impact the fuel, we have evaluated the offsite dose and criticality concerns of a postulated cask drop.

Our evaluation of the dose consequences conservatively assumes that every fuel assembly within a cask length radius of the shipping cask's pathway would experience complete cladding failure of every rod in the assembly. The methods for calculating gas gap activity and resulting off-site dose are the same as those used in performing calculations in Chapter 14 of the FSAR for the Fuel Handling Incident (FHI), where total cladding failure is assumed for one fuel assembly damaged three days after shutdown. For this analysis we decayed the FHI gas gap inventory for 440 days as opposed to the 3 day decay assumed in the FHI analysis. This resulted in the gas gap activity being reduced by a factor of 267 from the FHI activity.

Using acceptance criteria contained in NUREG-0612, which states that the heavy load drop is acceptable if the resulting offsite doses are limited to less than 25% of the 10 CFR 100 limits, over 5000 assemblies would have to be damaged to reach the NUREG limits based on the gas gap activity calculated for a 440-day decay. Because a maximum of approximately 500 assemblies could be contained within a cask length radius of the pathway, we would remain well within the NUREG limits even if all 500 assemblies could be damaged by the cask drop.

A more realistic analysis would predict that every assembly within the arc could not conceivably be damaged. The arc contains fuel assemblies in both the North and South pools, separated by a dam. Because the cask pit is located in the North pool and the placement of mechanical stops, controlled by a cask handling procedure, will prohibit travel of the cask over the dam into the South pool, fuel damage from a cask drop should occur only in the North pool. Considering this approach, the predicted offsite doses for the cask drop can even be shown to be bounded by the doses predicted for the FHI, where one assembly is damaged 3 days after shutdown. Based on the calculation described above for a 440-day decay, up to 267 assemblies could be damaged and the FHI dose would still not be exceeded. Since the number of fuel assemblies in the affected area of the North pool can be limited to 260, the FHI dose would bound the cask drop dose even if all of the fuel in the North pool lying within the arc was to be damaged. Furthermore, damage to every assembly in the North pool within the arc is highly unlikely, since if the cask fell, even if it rotated on its side, it would not be likely to roll continuously in a trajectory which would result in damage to every assembly within the arc.

In summary, the offsite dose consequences associated with the proposed modification remain well within limits specified in Criterion I of NUREG 0612, Section 5.1 (< 25% of 10 CFR 100 limits), and with a more realistic analysis, the results can even be shown to be bounded by doses previously calculated for the FHI. We thus conclude that the probability or offsite dose consequences of a heavy load drop have not increased significantly.

Criticality concerns were considered for the proposed modification because a cask drop would cause a geometrical distortion of the fuel/rack system. Because the distortion is difficult to predict, assumptions were made to bound the most reactive configuration. For the calculations, we assumed that the geometry of individual fuel assemblies was not deformed (this maximized reactivity) and that the storage racks were deformed to remove the inter-storage cell gap (neutron flux trap). Additionally, the poison material contained within the racks was ignored and replaced by pool water. While NUREG-0612 states that criticality analysis for a dropped load may assume that poison material integral to the rack remains in place, we conservatively chose to ignore this benefit. We also assumed an initial fuel assembly enrichment limit of 4.1 w/o U-235, a minimum assembly burnup of 28,000 MWD/MTU, and a minimum boron concentration in the spent fuel pool of 1000 ppm. Consistent with NUREG requirements, these assumptions (enrichment, burnup and boron concentration) are incorporated directly into the proposed Technical Specification change. Our procedures will require that serial numbers of fuel assemblies located within one cask length radius of the pathway be checked to confirm that all fuel within the arc conforms to the above restrictions. Also the boron concentration of the pool will be verified to be greater than or equal to 1000 ppm prior to cask movement into the pool and prior to cask movement out of the pool.

The method used to assess criticality concerns in the same method used to address criticality concerns relative to our spent fuel pool enrichment upgrade, which has received NRC approval. A two-dimensional analysis

using the DOT-IV computer code was performed for an infinite array of fuel assembly storage modules distorted as described above. The results yielded a K-eff of 0.898. Because uncertainties are less than 0.03, the K-eff value with uncertainties will be no greater than 0.928. Consequently, results are well within the criticality limit (i.e., K-eff ≤ 0.95) specified in Criterion II of NUREG 0612, Section 5.1.

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

The proposed change would allow fuel to be stored in an area previously prohibited when a shipping cask is being moved into the cask pit. Because fuel is normally stored around the cask pit when cask movement is not taking place, the conditions within the spent fuel pool have not changed. Since a heavy load drop was previously considered in our NUREG-0612 responses, we have not created a new accident scenario, but rather have increased the consequences of such an accident and as discussed above, these consequences remain well within established limits.

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

The potential adverse effects on safety margins associated with the proposed change involve offsite dose and criticality concerns caused by a cask drop. As previously discussed, by complying with the restrictions contained within the proposed change, offsite dose and subcritical margin are not significantly affected. Consequently, there is no significant decrease in a margin of safety.

In the March 6 1986, Federal Register Notice, the NRC listed examples of changes which are considered not likely to involve significant hazards considerations. Example (vi) from this list states:

"A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan, . . .".

The proposed change is similar to Example (vi) in that the change conforms with NRC guidance for heavy loads and the resulting consequences are well within established limits. Accordingly, the proposed change does not involve a significant hazards consideration.

SAFETY COMMITTEE REVIEW

The proposed change 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 this change will not result in an undue risk to the health and safety of the public.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT** :
County of Calvert :

I hereby certify that on the 30th day of April, 1990, before me, the subscriber, a Notary Public of the State of Maryland in and for St Mary's County, 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 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: Gerry L Adams
Notary Public

My Commission Expires: 1 July 1990
Date

GCC/DBO/bjd

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
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