GPU Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number:

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C321-95-2070

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Response to Request for Additional Information

References:
1) NRC letter, A. W. Dromerick to J. J. Barton, "Oyster Creek Spent Fuel Pool Expansion", dated December 29, 1994.
2) GPUN letter C321-94-2134, J. J. Barton to Document Control Desk, "Technical Specification Change Request No. 222", dated November 11, 1994.

The NRC staff requested additional information (Reference 1) regarding a GPU Nuclear request (Reference 2) to utilize forty five additional spent fuel storage locations which currently exist in the fuel storage pool. This existing capacity had been included in the original spent fuel pool expansion project but not reflected in the Technical Specifications. Attachment 1 is our response to the requested information. Attachment 2 provides the Technical Specification pages (5.3-1 and 5.3-2) affected by our Reference 2 request. Page 5.3-1 is revised in response to NRC staff question 9. Both pages reflect a change in font.

This letter is a resubmittal of our letter C321-95-2041 dated February 8, 1995. The original submittal inadvertently omitted the attached graph.

Sincerely,

Sander Levin

for J. J. Barton
Vice President and Director
Oyster Creek

Attachments
cc: Administrator, NRC Region I
NRC Senior Resident Inspector, Oyster Creek
Oyster Creek NRC Project Manager

Attachment 1

(5 pages)

Response to NRC Request
for Additional Information

References

1. *"Criticality Safety Evaluation of Oyster Creek Spent Fuel Storage Racks with Fuel up to 4.2% Enrichment", Holtec International (Stanley E. Turner), August 1993.*

2. *EPRI TR-101986, "Boraflex Test Results and Evaluation", February 1993.*

Question/Responses

1. **What methods and computer codes were used for the reanalysis of 4 wt.% fuel and Boraflex shrinkage? What organization performed the calculations and benchmarking?**

   CASMO-3 was used for the 4.0 wt.% enriched fuel analysis and was verified using the NITAWL-KENO5a code with the 27 group SCALE cross section library. The analysis method involves determining the k_{∞} of an infinite bundle lattice in the fuel pool geometry. Adjustments for mechanical, manufacturing, fuel burnup and computer code uncertainties are then applied to establish a fuel pool k_{eff} at a 95/95 confidence level. Previous analysis and benchmarking of the codes was performed by Southern Science and the new analysis was performed by Holtec International (Reference 1). Both analyses use the same methodology and were supervised by Dr. Stanley E. Turner. Additional benchmarking and verification was performed by GPU Nuclear on the 27 group (SCALE) NITAWL-KENO5a code package which was used to determine the Δk associated with Boraflex gaps.

2. **Since credit was taken for the Gadolinia burnable poison pins, at what point in core life does the fuel attain its peak reactivity?**

   Peak reactivity for 4.0 wt.% fuel occurs at 10.5 GWD/MT.

3. **What uncertainty was included to account for burnup calculations?**

   Burnup calculations include an uncertainty of 0.01 Δk.

4. **How have the other 95/95 calculational uncertainties (e.g. lattice spacing, channel bulge, etc.) changed as a result of the higher enrichment?**

   The 95/95 calculational uncertainties were not changed. The calculational uncertainties were calculated in the original criticality analysis based on a fuel enrichment of 3.01 wt.% <u>without</u> Gadolinia. The 3.01 wt.% enriched lattice

without Gadolinia is more reactive (k_∞ = 0.9295) than 4.0% fuel enriched with seven 3.0 wt.% Gadolinia pins at peak reactivity (k_∞ = 0.8938). Hence, the 95/95 uncertainties for the 4.0% analysis are bounded by the original analysis.

5. **Describe the Boraflex gap assumptions in more detail. For example, what does the assumption that the gaps are coplanar mean in terms of axial location and what percentage of shrinkage does a 3.9 inch gap correspond to?**

   The coplanar assumption is that a 3.9 inch gap (2.8% shrinkage) occurs in a single gap at the same axial level in all Boraflex panels. This is the maximum shrinkage expected by EPRI studies for the type of racks installed at Oyster Creek. A gap size of 3.9 inches bounds all gaps found to date as a result of blackness testing and is expected to bound all future gap formation due to Boraflex shrinkage reaching a saturation point at exposures greater than 1×10^{10} rads. The coplanar assumption represents the most conservative gap configuration since reactivity is compounded by neutron coupling effects. Actual measurement data has shown that less than half of the Boraflex panels have gaps and have an axial distribution which will reduce neutron coupling.

6. **Recent EPRI data indicates that, in addition to shrinkage, additional Boraflex degradation may occur due to long-term exposure to pool water flow. Has this been considered in the reanalysis?**

   No. While GPUN is aware of the loss of silica due to long term exposure to fuel pool water flow in combination with gamma exposure, it has not been considered in the reanalysis because, currently, the phenomenon is not well understood (see question 12). Due to the rack designs at Oyster Creek the Boraflex panels are not directly exposed to fuel pool water and therefore are resistant to this form of degradation. Although small holes exist on each panel for the purposes of gas release it does not provide sufficient communication with the fuel pool water to cause degradation due to water impingement. A surveillance program is in place to measure any material loss from Boraflex test coupons. These coupons are expected to bound any degradation that would occur within the rack structure due to more direct fuel pool water contact. At this time no significant weight loss in the test coupons is indicated. The next coupon measurement is scheduled for March 1995.

7. **How is assurance provided that no fuel assemblies enriched to 3.8 wt.% or higher have less than 7 Gadolinia pins of 3 wt.% loading?**

   The minimum poison loading criterion of 7 Gadolinia pins of 3 wt.% is based on an analysis for a fuel assembly that was utilized at Oyster Creek that had 3.19

wt.% fuel with 7 Gadolinia pins of 3 wt.%. As fuel enrichments increase, it has necessitated going to a higher number of fuel rods containing Gadolinia and to higher wt.% Gadolinia. A fuel lattice enriched to 3.6% is the highest enriched lattice at Oyster Creek containing 3 wt.% Gadolinia and has 9 Gadolinia rods. Higher enriched fuel lattices have both higher wt.% Gadolinia and more than 7 Gadolinia rods. Criticality analyses at higher fuel enrichments have continued to assume 7 rods of 3 wt.% Gadolinia for conservatism.

From an administrative perspective, new fuel designs are required to undergo design review by GPUN procedures. A New Fuel Design Checklist incorporated in the review process ensures that new fuel complies with enrichment, poison loading and other criteria established by analysis.

8. **Has the dry storage vault been analyzed for storage of 4.0 wt.% fuel under accident conditions of full flooding and optimum moderation?**

No. The dry storage vault design meets the design requirements of 10CFR general design criteria 61 and 62 for the initial fuel loading at Oyster Creek for unflooded and fully flooded conditions. No subsequent analysis has been performed to update the dry storage vault analysis for new fuel and/or for optimum moderation. For this reason, Oyster Creek maintains restrictions on the uncovering of the dry storage vault to prevent a spray type moderation from the refueling floor fire system. The initial fuel loading at Oyster Creek did not contain Gadolinia and is more reactive at beginning of life than any subsequent fuel designs loaded at Oyster Creek. Although enrichment loadings have increased substantially, the Gadolinia loading have kept the beginning of life reactivity at much lower reactivity than bundles initially loaded in the dry storage vault. The initial Oyster Creek bundles at beginning of life have a k_{∞} of 1.227 and the highest reactivity in any current new bundle lattice has a k_{∞} of only 1.120. Therefore, the current fuel designs are bounded by previous analysis on the dry storage vault.

9. **Is the phrase "as performed on the poison racks" in the third line of Basis 5.3.1 misplaced?**

This sentence was intended to remain unchanged; "as performed on the poison racks" should be removed. A revised page 5.3-1 is included in Attachment 2.

10. **How was the maximum gap of 3.9 inches in Boraflex panels determined? What was the corresponding gamma exposure (rads)?**

3.9 inches is the maximum expected gap size that would result from 2.8% Boraflex shrinkage based on EPRI studies (Reference 2) for the rack type

installed at Oyster Creek. The enclosed plot shows Oyster Creek's blackness testing results and coupon data as it compares to EPRI maximum expected shrinkage levels. The 2.8% is from the upper maximum shrinkage curve. This is the shrinkage associated with gap formation. Additional shrinkage will occur at the top and bottom ends of the Boraflex panels which has a very small effect on reactivity due to leakage effects. The largest total gap size found at OysterCreek was 2.42 inches at an exposure of 3.0×10^9 rads (2 gaps of 1.07 and 1.35 inches). The shrinkage is expected to saturate at exposure levels greater than 1.0×10^{10} rads and gap growth is expected to be bounded by the 3.9 inch gap assumed in the coplanar gap analysis.

11. What kind of evaluations were performed on the surveillance coupons?

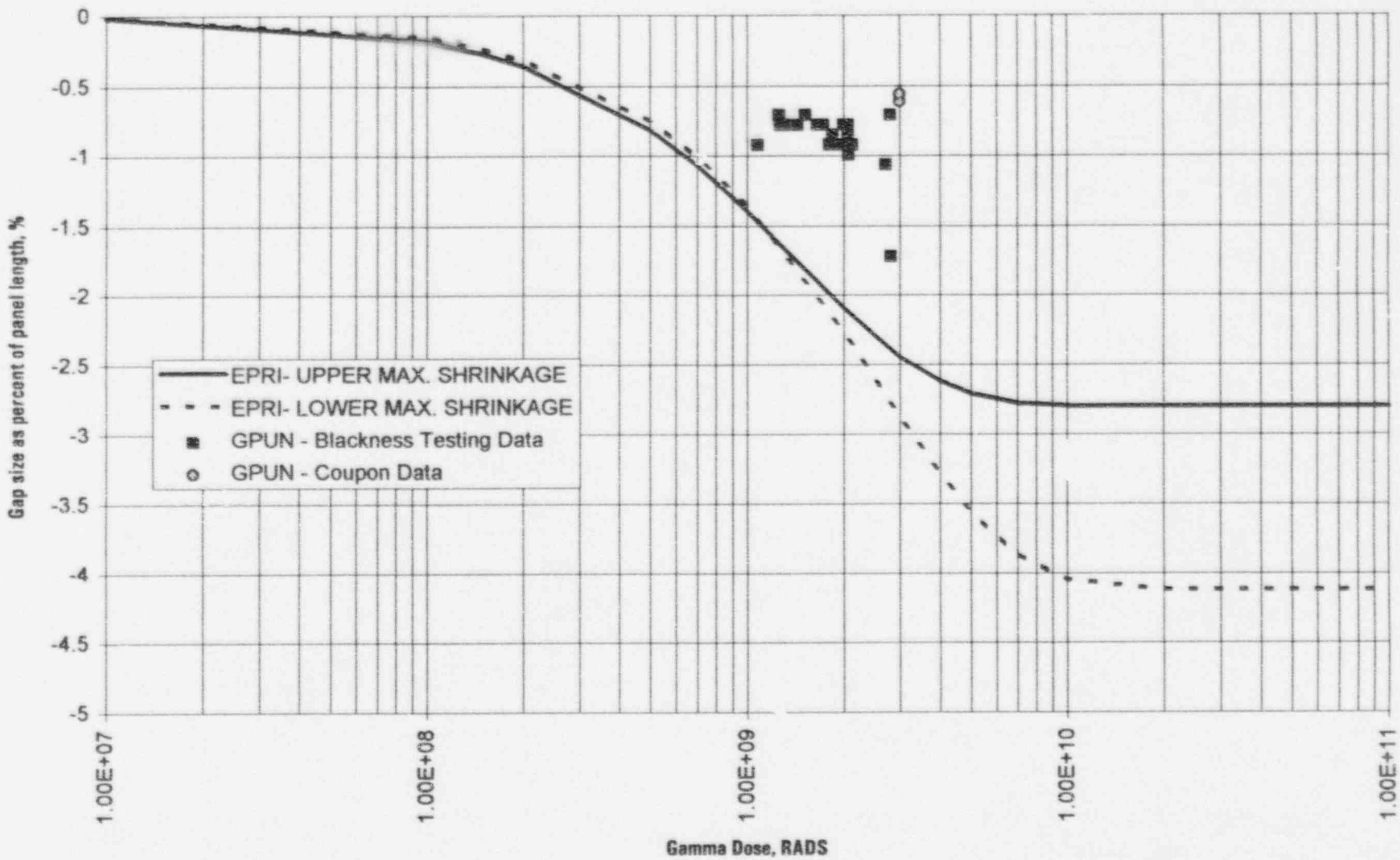
Coupons are visually inspected as well as measurements recorded of thickness, length, width, weight and hardness. Neutron attenuation testing is planned if problems are suspected from the weight and dimensional checks. Coupons are inspected at predesignated intervals depending on if the samples are from the long term test or the accelerated test assembly. High gamma dose bundles are placed near the accelerated coupons to insure that they are receiving a higher than average gamma dose rate than the rack panels.

12. Was the concentration of silica measured in the spent fuel pool? If this measurement was made, was the data used to predict degradation (loss of boron) of Boraflex panels?

Silica concentration measurements are made on a continual basis as part of normal fuel pool chemistry monitoring. The peak silica level was found to be 3.0 ppm. However, fuel pool clean-up system is placed in service periodically to remove the silica. Starting this cycle (Cycle 15), the silica will not be removed until end of cycle. This will allow for a better understanding of silica buildup which will be needed to evaluate Boraflex degradation in the future.

At this time no significant weight loss has been indicated based on the Boraflex sample coupons. The sample coupons are expected to bound degradation within the racks due to more direct contact with fuel pool water. GPUN is participating in an EPRI study currently underway to develop a methodology for predicting the degradation of Boraflex. Results of the study should be available by mid 1995. When a methodology is available, GPUN will evaluate the need to include Boraflex degradation into the fuel pool criticality analysis.

Gap Size versus Integrated Gamma Exposure



Attachment 2

(2 pages)

Replacement Technical
Specification Pages

5.3 AUXILIARY EQUIPMENT

5.3.1 Fuel Storage

A. The fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties.

B. Loads greater than weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility.

C. The spent fuel shipping cask shall not be lifted more than six inches above the top plate of the cask drop protection system. Vertical limit switches shall be operable to assure the six inch vertical limit is met when the cask is above the top plate of the cask drop protection system.

D. The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F.

E. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 2645.

BASIS

The specification of a K-effective less than or equal to 0.95 in fuel storage facilities assures an ample margin from criticality. This limit applies to unirradiated fuel in both the dry storage vault and the spent fuel racks as well as irradiated fuel in the spent fuel racks. Criticality analyses were performed on the poison racks to ensure that a K-effective of 0.95 would not be exceeded. The analyses took credit for burnable poisons in the fuel and included manufacturing tolerances and uncertainties as described in Section 9.1 of the FSAR. Calculational uncertainties described in 5.3.1.A are explicitly defined in FSAR Section 9.1.2.3.9. Any fuel stored in the fuel storage facilities shall be bounded by the analyses in these reference documents.

The effects of a dropped fuel bundle onto stored fuel in the spent fuel storage facility has been analyzed. This analysis shows that the fuel bundle drop would not cause doses resulting from ruptured fuel pins that exceed 10 CFR 100 limits (1,2,3) and that dropped waste cans will not damage the pool liner.

The elevation limitation of the spent fuel shipping cask to no more than 6 inches above the top plate of the cask drop protection system prevents loss of the pool integrity resulting from postulated drop accidents. An analysis of the effects of a 100-ton cask drop from 6 inches has been done (4) which showed that the pool structure is capable of sustaining the loads imposed during such a drop. Limit switches on the crane restrict the elevation of the cask to less than or equal to 6 inches when it is above the top plate.

Detailed structural analysis of the spent fuel pool was performed using loads resulting from the dead weight of the structural elements, the building loads, hydrostatic loads from the pool water, the weight of fuel and racks stored in the pool, seismic loads, loads due to thermal gradients in the pool floor and the walls, and dynamic load from the cask drop accident. Thermal gradients result in two loading conditions; normal operating and the accident conditions with the loss of spent fuel pool cooling. For the normal condition, the containment air temperature was assumed to vary between 65°F and 110°F while the pool water temperature varied between 85°F and 125°F. The most severe loading from the normal operating thermal gradient results with containment air temperatures at 65°F and the water temperature at 125°F. Air temperature measurements made during all phases of plant operation in the shutdown heat exchanger room, which is directly beneath part of the spent fuel pool floor slab, show that 65°F is the appropriate minimum air temperature. The spent fuel pool water temperature will alarm control room before the water temperature reaches 120°F.

Results of the structural analysis show that the pool structure is structurally adequate for the loadings associated with the normal operation and the condition resulting from the postulated cask drop accident (5) (6). The floor framing was also found to be capable of withstanding the steady state thermal gradient conditions with the pool water temperature at 150°F without exceeding ACI Code requirements. The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F (5).

Since the cooled fuel pool water returns at the bottom of the pool and the heated water is removed from the surface, the average of the surface temperature and the fuel pool cooling return water is an appropriate estimate of the average bulk temperature; alternately the pool surface temperature could be conservatively used.

<u>References</u>

1. Amendment No. 78 to FDSAR (Section 7)
2. Supplement No. . to Amendment No. 78 to the FDSAR (Question 12)
3. Supplement No. 1 to Amendment 78 of the FDSAR (Question 40)
4. Supplement No. 1 to Amendment 68 of the FDSAR
5. Revision No. 1 to Addendum 2 to Supplement No. 1 to Amendment No. 78 of FDSAR (Questions 5 and 10)
6. FDSAR Amendment No. 79
7. Deleted