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August 27, 1996

Administrative Judge G. Paul Bollwerk, Chairman
Administrative Judge Dr. Peter S. Lam
Administrative Judge Dr. Charles N. Kelber
Atomic Safety & Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of GPU Nuclear Corporation
(Oyster Creek Nuclear Generating Station)
Docket No. 50-219-OLA

Chairman Bollwerk and Judges Lam and Kelber:

Upon my return yesterday from a trip outside the U.S., I learned for the first time of the Board's August 14 Memorandum, Petitioners' related communications to the Board (telecon on August 15 and letter mailed on August 16 and received by Licensee counsel on August 20), Mr. Lewis' and Ms. Hodgdon's joint conference with Chairman Bollwerk by telephone on August 19, and an August 23 submittal to NRC Staff by GPU Nuclear concerning the NUHOMS Transfer Cask Shield Plug.

The latter document is enclosed for the information of the Board and participants because it relates directly to the subject matter of this proceeding. Additionally, I feel constrained to clarify the substance of settlement discussions between Licensee and Petitioners.

It is only with reluctance that I discuss the substance of settlement discussions at all. It is my experience that for well understood and sensible reasons this topic is regarded as confidential between parties. Mr. Gunter and I have discussed the matter of confidentiality of settlement discussions quite explicitly, and a voice mail I had from him on August 15 confirmed this when he related his intention to call Chairman Bollwerk and also send a short letter saying "simply that no agreement was reached. It doesn't detail anything." To the contrary, however, the Petitioners' letter of August 15 reports on a settlement discussion Mr. Gunter and I had immediately following the bench conference with the Board at the end of the August 8 prehearing conference. Petitioners' summary is "Mr. Blake informed Mr. Gunter that the GPUN safety analysis could not be made available to the Petitioners."



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Administrative Judge G. Paul Bollwerk, Chairman
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What actually occurred was that I told Mr. Gunter that I believed I had no obligation to provide him the dose assessment analysis that I had referred to during the prehearing conference, but if he really wanted it to use in subsequent broader hearings we both anticipated may occur on actual transfer cask movements of spent fuel (as opposed to this very particularized DSC shield plug amendment proceeding), I would make it available to him in return for Petitioners' withdrawal from the instant case. He said he would consider the offer and several days later called me to say no. No other settlement discussions have occurred.

Respectfully,

Ernest L. Blake
Counsel for Licensee

Enclosure

cc: Service List

347199-01 / DOCSDC1

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6730-96-2265

August 23, 1996

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
NUHOMS® Transfer Cask Shield Plug

Reference: Technical Specification Change Request No. 244 dated 4/15/96 from Michael B. Roche (GPU Nuclear) to USNRC

As requested by the NRC staff, the attachments to this letter provide the results of the analyses discussed before the Atomic Safety and Licensing Board on August 7, 1996 regarding the reference license amendment request.

The license amendment request proposes to revise Technical Specification 5.3.1.B to allow the top shield plug for the dry shielded canister to be moved over spent fuel assemblies in the canister while it is in the plant's cask drop protection system. The basis for the request is that a drop is not a credible event. Attachment 1 addresses criticality potential and attachment 2 provides an evaluation of radiological consequences for a hypothetical drop of the shield plug.

Sincerely,

Michael B. Roche

Michael B. Roche
Vice President and Director
Oyster Creek

MBR/PFC/plp

Attachments

c: Administrator, USNRC Region I
USNRC Senior Resident Inspector
Oyster Creek USNRC Project Manager

**Attachment 1**

**Potential for Criticality**

The potential for criticality as a result of dropping the shield plug lid onto fuel assemblies in the dry shielded canister (DSC) was determined by GPU Nuclear based on guidelines provided in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". All the fuel in the cask (52 assemblies) is assumed to crush together such that $k_{eff}$ is maximized. The impact of the shield plug drop is not considered severe enough to significantly damage the rigid structural material of the cask containing the DSC and, therefore, the borated stainless steel plates are expected to remain intact. An ENC3e-3f fuel assembly, 7x7 lattice with 2.63 weight % enriched $U_{235}$ was used for this analysis as it bounds the reactivity of all fuel available for dry storage.

Criticality analysis was performed using the Monte-Carlo code KENO-Va using the 27 group cross section library collapsed from ENDF-IV. Mixture cross sections were developed using the material information processor sequence CSAS25 of SCALE 4.2 which uses the BONAMI, NITAWL and ICE modules. In order to insure confidence in the cross section library and the KENO model of the DSC a comparison was made to results found in the cask safety analysis report (CSAR)(Section 3.3) for a GE 7x7 4.0 weight % enriched bundle at beginning of core life. Results compared well, with a 0.880 $k_{eff}$ in the CSAR compared with 0.882 $k_{eff}$ for the GPU Nuclear analysis including all biases and uncertainties.

Using the above assumptions, the maximum DSC $k_{eff}$, with a 95/95 confidence level, as a result of the dropped shield plug was determined to be 0.957. This includes all biases and uncertainties associated with KENO and mechanical uncertainty in the DSC design. The result is conservative because the analysis does not include fuel burnup, which will significantly lower the $k_{eff}$ since burnup averages above 23 GWD/MT for this fuel. In addition, this analysis assumes all bundles in the DSC are affected by the dropped shield plug whereas geometric considerations show that only 16 bundles would be directly impacted.

References:

1) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

2) Safety Analysis Report for the Standardized NUHOMS Horizontal Modular Storage System, Revision 3

**Attachment 2**

**Radiological Consequences**

The radiological consequences of dropping the shield plug onto the NUHOMS®-52B dry shielded canister (DSC) is bounded by the radiological analysis summarized in Table 2.1-1 of NUREG-0612. The NUREG-0612 analysis for fuel that has been subcritical for 120 days indicates that 16,000 fuel assemblies must be damaged to reach ¼ of Part 100 exposure limits, or 75 rem thyroid and 6.25 rem whole body. The DSC contains 52 fuel assemblies and is in the transfer cask located in the Oyster Creek Cask Drop Protection System (CDPS) twenty feet below the surface of the spent fuel pool. A comparison of assumptions used in the NUREG-0612 analysis to Oyster Creek data is provided in Table 1. Clearly, the NUREG-0612 analysis is bounding for Oyster Creek.

Assuming that all 52 fuel assembles are damaged by the load drop, radiological releases from the fuel (minimum ten year decay) will have minimal consequences. The fission gases (primarily Kr-85 with a half-life of 10 years) are released inside secondary containment. Due to the length of time for decay, there is no iodine to release. Even if conservative NUREG-0612 assumptions are applied and fuel only 120 days subcritical assumed, radiological consequences as a result of damage to all 52 fuel assemblies in the DSC could be no more than 20 millirem whole body dose at the site boundary. For the case of 16 directly impacted fuel assemblies (maximum possible due to shield plug drop), the whole body dose could be no more than 6.25 millirem. In reality, given the conservatism of NUREG-0612 assumptions for power level, X/Q, and cooling time relative to actual Oyster Creek data, radiological consequences for a shield plug drop would be expected to be essentially zero.

**Table 1**

**Comparison of NUREG-0612 Heavy Load Drop Assumptions To Oyster Creek Data**

| Parameters | NUREG-0612 | Oyster Creek |
|---|---|---|
| Power Level ($MW_{th}$) | 3000 | 1930 |
| 0-2 hour X/Q, $sec/M^3$ (exclusion area boundary) | $1.0x10^{-3}$ [1] | $1.1x10^{-5}$ [5] |
| 0-2 hour X/Q LPZ, $sec/M^3$ | $1.0x10^{-4}$ [1] | $1.1x10^{-6}$ [5] |
| Peaking Factor | 1.2 [2] | <1.0 [6] |
| No. of Assemblies in Core | 760 | 560 |
| Pool Water Decontamination Factor | 100 [3] | N/A |
| Filter Efficiency Elemental Iodine | 95% [4] | N/A |
| Filter Efficiency Organic Iodine | 95% | N/A |
| Cooling Time (hours) | 100 or greater | $4.38x10^4$ |

Notes

1. Based on 5% worst meteorological conditions.

2. Value is 1.2 for greater than one damaged fuel assembly. For a single assemble the values are 1.65 and 1.5 for PWRs and BWRs, respectively.

3. See Regulatory Guide 1.25.

4. See Regulatory Guide 1.52.

5. Oyster Creek UFSAR Table 2.3-30 for an elevated level release. Values based on Regulatory Guide 1.145 rev 1. LPZ (Low Population Zone) value is for 8 hours.

6. Fuel loaded in cask is fuel discharged at end of life.