November 7, 1996

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: ISSUANCE OF AMENDMENT RE: HANDLING HEAVY LOADS OVER IRRADIATED FUEL (TAC NO. M95233)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated April 15, 1996.

The amendment revises Specification 5.3.1.B to allow the shield plug and the associated lifting hardware to be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by:

Ronald B. Eaton, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 187 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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cc w/encls: See next page

M. Roche
GPU Nuclear Corporation

Oyster Creek Nuclear
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated April 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 7, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
5.3-1	5.3-1
5.3-2	5.3-2

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. 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.

2. The shield plug and the associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

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.

Administrative controls over crane movements, which include mechanical rail stops, serve to prevent travel of the crane outside the analyzed load path over the cask drop protection system. A safety factor greater than 10 with respect to ultimate strength, and redundant shield plug lift cables provide adequate margin for the shield plug lift. These features, combined with operator training and required inspections, contribute to the determination that dropping the shield plug onto a loaded dry shielded canister in the spent fuel pool is not a credible event.

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.

References

1. Amendment No. 78 to FDSAR (Section 7)
2. Supplement No. 1 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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 187

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated April 15, 1996, GPU Nuclear Corporation (GPU, the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TS). The requested changes would revise TS pages 5.3-1 and 5.3-2 to permit loads in excess of the current TS limits to be moved over a cask loaded with fuel assemblies in the spent fuel storage facility. By letter of August 23, 1996, the licensee supplemented its request with an analysis of criticality potential and of the radiological consequences of a hypothetical drop of the shield plug. The supplement did not change the staff's conclusions in its proposed no significant hazards consideration determination (May 8, 1996, 61 FR 20849).

2.0 BACKGROUND

At the Oyster Creek plant site, the process of transferring spent fuel assemblies from the spent fuel storage facility to the Independent Spent Fuel Storage Installation (ISFSI) includes placing a dry shielded canister (DSC) within a transfer cask into the cask drop protection system (CDPS) located inside the spent fuel storage facility. The CDPS protects the spent fuel pool and the irradiated fuel stored in racks in the spent fuel pool in the event the cask is dropped. This movement does not involve the handling of a heavy load over irradiated fuel. The DSC is then loaded with spent fuel assemblies. Before the DSC and the transfer cask in which it is contained can be removed from the spent fuel storage facility, the DSC shield plug must be lowered into the CDPS and placed in position on top of the DSC to serve as a radiological shield. The current TS is ambiguous regarding this movement because the DSC, at that point, contains irradiated fuel, and the weight of the shield plug and lifting yoke is greater than the weight of one fuel assembly. However, the fuel in the DSC is not "stored" in the pool and the prohibition against movement of a load heavier than an assembly plus its lifting gear refers to "stored" fuel. GPU has sought to resolve the ambiguity by modifying the TS to clarify that the shield plug may be moved onto the DSC after the DSC has been loaded with irradiated fuel. The proposed TS change would facilitate the

off-load of spent fuel to Oyster Creek's ISFSI by permitting the licensee to lower the DSC shield plug into the CDPS and place it in position on top of the DSC after the DSC has been loaded with irradiated fuel. This movement will not involve the handling of a heavy load over irradiated fuel in the storage racks.

3.0 EVALUATION

Section 5.3.1, Fuel Storage, reads as follows:

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

In order to implement the changes described in Section 2.0 above, the licensee proposes to change the TS as follows:

> B. 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.
>
> 2. The shield plug and the associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

As indicated above, this section would enable the licensee to lift the DSC shield plug and associated lifting hardware over irradiated fuel assemblies in the DSC within the transfer cask in the CDPS.

In addition to the proposed change to the TS, the licensee has updated the TS Basis to state that

> "Administrative controls over crane movements, which include mechanical rail stops, serve to prevent travel of the crane outside the analyzed load path over the cask drop protection system. A safety factor greater than 10 with respect to ultimate strength, and redundant shield plug lift cables provide adequate margin for the shield plug lift. These features, combined with operator training and required inspections, contribute to the determination that dropping the shield plug onto a loaded dry shielded canister in the spent fuel pool is not a credible event."

The NRC staff has completed its review of the proposed change, the reason for the change, and the safety analysis provided by the licensee. This NRC staff review and evaluation is limited to the specific issue of placing the DSC shield plug (a heavy load) in position on top of the DSC after the DSC has been loaded with irradiated fuel. This review does not address the movement of other heavy loads. The staff has considered the guidance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and NUREG-0554, "Single-

Failure-Proof Cranes for Nuclear Power Plants," and other guidance such as ANSI B30.9, "Slings," and ANSI B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder Top Running Hoist)."

According to information provided by the licensee, the reactor building (RB) crane has a main hoist capacity of 100 tons. The actual safety factors of the main crane for its 100-ton rated load are: cables 6.5:1; main hoist gearing 5.2:1; and main hoist brakes 120% capacity. These safety factors are within the guidelines established in NUREG 0612. These safety factors are with respect to ultimate strength. As a result, when moving the shield plug and the lifting yoke with a combined weight of approximately 7 tons, a safety factor greater than 14 with respect to the 100-ton rated capacity of the RB crane will be provided, and greater than 70 with respect to the ultimate strength. For the lifting yoke, a safety factor greater than 26 will be provided, based on the lifting yoke's 105-ton rated capacity. The least conservative safety factor is that for the shield plug lift cables. The safety factor is 11:1, based on the ultimate load of 22,800 pounds. The shield plug lift cables are redundant and each of the four has sufficient capacity to support the total weight of the 8000 pound shield plug.

The licensee has modified the RB crane to enhance its performance and reliability by improving the instrumentation and controls and has developed an error-free plan that includes a dedicated management team and a dedicated crew who will be trained and on-shift. The plan also includes detailed operating instructions and procedures. In its April 15, 1996, application the licensee committed to a special crane inspection that will be performed prior to each dry fuel storage campaign; the main hoist coupling, shaft, and hook will be examined by NDE [nondestructive examination] prior to each campaign. The licensee has also stated that personnel training, crane inspections, testing, and maintenance will be in accordance with ANSI B30.2.

Based on the considerations discussed above, the NRC staff concludes that the design features and modifications of the crane, the licensee's error-free plan and commitments, and the significant factors of safety described in the licensee's request for changes to the TS makes a drop of the shield plug extremely unlikely to the point of not being credible.

This proposed TS amendment specifically addresses the issue of placing the shield plug (a radiological shield for the dry shielded canister) on the DSC. Even though the event is not credible, the staff evaluated the potential radiological consequences that could result from a hypothetical drop of a shield plug that lands in a random position on top of the DSC resulting in damage to the spent fuel within the DSC.

By letter dated August 23, 1996, GPU Nuclear provided an analysis of the radiological consequences of dropping the shield plug after fully loading the dry storage canister with spent fuel. Sixteen fuel assemblies are damaged such that all of the gaseous radioactive materials in the fuel pin gaps is released into the secondary containment. This radioactivity is assumed to immediately mix with the air volume of the reactor building and be exhausted to the environment through the plant stack by the standby gas treatment system (SBGT). The staff used the TACT5 computer code to calculate the resulting

radiation doses at the exclusion area boundary (EAB) and low population zone (LPZ) as defined in 10 CFR Part 100. The following assumptions and input parameters were used:

(a) All 16 fuel assemblies (1/35 of the core) were exposed to the maximum neutron flux for three operational cycles. Therefore, a peaking factor of 1.5 was applied (consistent with the guidance in Regulatory Guide 1.25) to the 1930 MWt full power level for each assembly.

(b) The free volume of the secondary containment of 1,800,000 ft^3 was taken from Table 6.2-11 of the Oyster Creek UFSAR.

(c) No credit was taken for scrubbing of activity by the fuel pool water.

(d) Charcoal filter in the SBGT system credited with removing 90% of the radioactive iodine species.

(e) Consistent with the guidance in Regulatory Guide 1.25, the fraction of the fuel's radioactivity in the fuel pin gap (i.e., available for release from the damaged fuel) was assumed to be 10% of the radioactive iodines and 30% of the noble gases.

(f) The affected fuel had 10 years of decay in the fuel pool before loading into the cask. (For comparison, a second calculation assuming only 1 year of decay was performed.)

For the case where the fuel had decayed for 10 years, virtually the only gaseous radioisotope remaining in the fuel gap is the noble gas Kr-85. Therefore, as would be expected, the TACT5 code calculated zero thyroid dose at the EAB and LPZ. The 2-hour whole-body dose at the EAB and the 30-day whole-body dose at the LPZ were 4.12×10^{-6} rem and 1.62×10^{-6} rem, respectively. As noted above, a case was run with only 1-year of radioactive decay for the spent fuel. Although the TACT5 code calculated some residual Iodine-131 in the source term, Kr-85 still dominated the resulting dose such that zero thyroid dose was calculated at the EAB and LPZ. The whole-body doses were 7.36×10^{-6} rem and 2.90×10^{-6} rem, for the EAB and LPZ respectively. The siting criteria in 10 CFR Part 100 specify that the doses resulting from a spectrum of accidents not exceed 300 rem to the thyroid or 25 rem to the whole body for individuals at the EAB and LPZ boundaries, respectively. As implemented in NRC staff policy for the acceptable consequences of a fuel handling accident in Section 15.7.4 "Radiological Consequences of Fuel Handling Accidents" in NUREG-0800 "Standard Review Plan," resulting doses do not exceed 25% of the Part 100 criteria. The doses calculated by the staff for the postulated accident are <u>well within</u> (6 orders of magnitude below) the acceptance criteria in Section 15.7.4 of NUREG-0800.

Accidental criticality caused by the dropping of the shield plug onto the DSC is not a credible event not only because of the multiple protections against dropping the plug but also because of the design specifications for the DSC. On the basis of the analysis presented in the NUHOMS SAR and independent confirmatory calculations performed by the staff, the staff concluded in the NUHOMS SER that the standardized NUHOMS-52B design and proposed operating

procedures are adequate to maintain the system in a subcritical configuration and to prevent a nuclear criticality accident and therefore satisfy 10 CFR 72.124 and 10 CFR 72.236(c), subject to the key factors assumed by the vendor in the analysis, specifically: 1) criticality safety calculations presented in the SAR and independent confirmatory calculations performed by the staff showing that criticality safety is ensured for a maximum initial U-235 fuel enrichment of 4.0 wt%, which was determined for the design basis GE-2 7x7 fuel assembly; and 2) the criticality safety analysis assuming a minimum boron density of 0.75 wt% boron in the borated stainless steel absorber plates. The key factors and assumptions used by the vendor in the criticality safety analysis are as follows: 1) maximum fuel enrichment of fuel assemblies stored in the standardized NUHOMS-52B system of 4.0 wt% U-235; 2) minimum of 0.75 wt% boron loading in the neutron absorber plates; and 3) altered mechanical configuration of the array of fuel assemblies resulting from an accident not credible.

In addition to the analyses provided by the NUHOMS vendor for the NUHOMS SAR and the NRC staff confirmatory calculations, GPU has provided an analysis for a configuration specifically applicable for Oyster Creek. The analysis used the widely used industry standard Monte-Carlo code KENO-Va (developed by ORNL [Oak Ridge National Laboratory]), and standard auxiliary codes and data to provide cross section information. These provide an acceptable methodology to examine criticality aspects of relevant configurations. GPU validated its use of this methodology by comparison calculations from the cask safety analysis report calculations.

With a full load of 52 fuel assemblies in the cask the hypothetical drop of the shield plug, based on the geometry of the system, would not be expected to affect more than 16 assemblies. Expected damage would be some crushing of the upper part of the fuel assemblies, in the area of the upper end reflector region of the fuel, and result in little change in reactivity. GPU, however, has analyzed a configuration in which all 52 assemblies are moved together to form a tight, cylindrical bundle to maximize the reactivity increase. The boron/stainless steel blocks are assumed to remain between the assemblies, but the compression lowers their effectiveness by removing the flux trap water gaps initially present. The normal fuel assembly configuration is maintained since it is near maximum reactivity for the materials involved. The fuel enrichment used was 2.63 wt% U-235 with no burnable poison and no burnup assumed. The 2.63 value bounds the fuel enrichments to be used for dry storage. The burnup provides a considerable conservatism since the actual burnup would average over 23 GWD/MT, which would offer little if any potential for forming a critical configuration. The result of this calculation was a k(eff) value of 0.957 at a 95/95 probability/confidence level, considering the uncertainties associated with KENO-Va and the canister design. This provides a reasonable demonstration that there is little probability of a criticality event from rearrangement caused by a shield plug drop.

4.0 SIGNIFICANT HAZARDS CONSIDERATIONS COMMENTS

The licensee's request for amendment was noticed in the FEDERAL REGISTER on May 8, 1996 (61 FR 20849). In the notice, the staff made a proposed determination of no significant hazards consideration and offered an

opportunity for hearing. On June 6, 1996, Nuclear Information and Resource Service (NIRS), Oyster Creek Nuclear Watch (OCNW), and Citizens Awareness Network (CAN) jointly filed a request for hearing and petition to intervene. Included in the hearing request were comments on the proposed no significant hazards consideration determination. Petitioners allege that the proposed amendment (1) represents a significant increase in the probability of an accident, (2) creates the possibility of an accident not previously identified in the Safety Analysis Report and, (3) constitutes a significant reduction in the margin of safety. The staff's response to these comments follows.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 include three standards used by the NRC staff to arrive at a determination regarding whether a request for amendment involves no significant hazards considerations. The regulation states that the Commission may make such a final determination if operation of a 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 following staff evaluation in relation to the three standards demonstrates that the proposed TS amendment to place the DSC shield plug in position on top of the DSC to serve as a radiological shield does not involve a significant hazards consideration.

First Standard

"Involve a significant increase in the probability or consequences of an accident previously evaluated."

In accordance with the information provided by the licensee, the reactor building (RB) crane has a main hoist capacity of 100 tons. The actual safety factors of the main crane for its 100-ton rated load are: cables 6.5:1, main hoist gearing 5.2:1, and main hoist brakes 120% capacity. These safety factors are with respect to ultimate strength. As a result, when moving the shield plug and the lifting yoke with a combined weight of approximately 7 tons, a safety factor greater than 14 with respect to the 100-ton rated capacity of the RB crane will be provided, and greater than 70 with respect to the ultimate strength. For the lifting yoke, a safety factor greater than 26 will be provided, based on the lifting yoke's 105-ton rated capacity. The least conservative safety factor is that for the shield plug lift cables. The safety factor is 11:1, based on the ultimate load of 22,800 lbs. The shield plug lift cables are redundant and each of the four has sufficient capacity to support the total weight of the 8000-pound shield plug.

The licensee has modified the RB crane to enhance its performance and reliability by improving the instrumentation and controls, and has developed an error-free plan that includes a dedicated management team and a dedicated crew, who will be trained and on shift along with detailed operating instructions and procedures. The licensee has committed to a special crane

inspection that will be performed prior to each dry fuel storage campaign; the main hoist coupling, shaft, and hook will be examined by NDE prior to each campaign. The licensee has also stated that personnel training, and crane inspections, testing, and maintenance will be in accordance with ANSI B30.2.

Based on the above discussion, the staff concludes that when considering the qualitative analysis of the safety factors and RB crane enhancements, the event is so unlikely as to be non-credible.

Second Standard

"Create the possibility of a new or different kind of accident from any accident previously evaluated."

The accident to consider with respect to the proposed TS amendment is dropping a shield plug (a shield plug is a heavy load for Oyster Creek) that lands in a random position on top of the DSC, damaging the fuel within the DSC.

As discussed above, under the first standard, an accident resulting from a plug drop is not a credible event and, therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

"Involve a significant reduction in a margin of safety."

The staff agrees with the licensee's conclusion that dropping the DSC shield plug onto a loaded DSC and damaging the spent fuel assemblies therein is not a credible event.

The staff finds that the proposed amendment does not involve a significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (61 FR 20849).

In Section 5.0 of this safety evaluation the Commission has made a final no significant hazards consideration determination with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulation and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Harold Walker, SPLB
Howard J. Richings, SRXB
Roger L. Pedersen, PERB

Date: November 7, 1996