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June 26, 1996

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION '96 JUN 27 A10:42

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITY NUCLEAR CORPORATION	)	Docket No. 50-219-OLA
	)	
(Oyster Creek Nuclear Generating Station)	)	

NRC STAFF RESPONSE IN OPPOSITION TO REQUEST
FOR HEARING AND PETITION TO INTERVENE
OF NUCLEAR INFORMATION AND RESOURCE SERVICE,
OYSTER CREEK NUCLEAR WATCH AND CITIZENS AWARENESS NETWORK

INTRODUCTION

Pursuant to the Commission's regulations at 10 C.F.R. § 2.714(c), the staff of the Nuclear Regulatory Commission (Staff) hereby submits its response to the June 6, 1996 request for a hearing and petition to intervene[1] jointly filed by Nuclear Information and Resource Service (NIRS), Oyster Creek Nuclear Watch (OCNW) and Citizens Awareness Network (CAN) (collectively Petitioners). As set forth below, none of the Petitioners has shown that it has standing to intervene in a hearing on the proposed amendment and, therefore, the Atomic Safety and Licensing Board (Board) should deny the Petition.

[1] "Nuclear Information and Resource Service, Oyster Creek Nuclear Watch, and Citizens Awareness Network Request for a Hearing and Petition to Intervene on General Public Utility Nuclear License Amendment Request for Oyster Creek Nuclear Generating Station" (Petition).

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BACKGROUND

On April 15, 1996, General Public Utility Nuclear Corporation (Licensee, GPUN) submitted an amendment request to the Staff in which it sought to revise Specification 5.3.1.B of the Oyster Creek Technical Specifications relating to refueling and spent fuel handling. The current specification prohibits handling a load greater in weight than one fuel assembly over irradiated fuel stored in the spent fuel storage facility. The proposed change will permit the shield plug for the dry shield canister (DSC) and the associated lifting hardware to be moved over irradiated fuel that is contained in the DSC within the transfer cask located in the Cask Drop Protection System. On May 8, 1996, the Staff published a notice of consideration of issuance of an amendment to the Oyster Creek Nuclear Generating Station operating license. 61 Fed. Reg. 20842, 20848 (May 8, 1996). The Staff, in its notice, stated its proposal to determine that the amendment request involves no significant hazards consideration. On June 6, 1996, the Petitioners filed their Petition.

DISCUSSION

I. Intervention as of Right

A. Standards for Standing to Intervene

A request for hearing and petition for leave to intervene must be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 C.F.R. Part 2. Pursuant to 10 C.F.R. § 2.714(a):

> (1) Any person whose interest may be affected by a proceeding and who desires to participate as a party shall file a written petition for leave to intervene.
>
> * * * *

> (2) The petition shall set forth with particularity the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, including the reasons why petitioner should be permitted to intervene, with particular reference to the factors in paragraph (d)(1) of this section, and the specific aspect or aspects of the subject of the proceeding as to which petitioner wishes to intervene.

Pursuant to 10 C.F.R. § 2.714(d)(1), a petition for leave to intervene must address 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.
>
> (3) The possible effect of any order that may be entered in the proceeding on the petitioner's interest.

The Commission has long held that contemporaneous judicial concepts of standing will be applied in determining whether a petitioner has sufficient interest in a proceeding to be entitled to intervene as a matter of right under Section 189 of the Atomic Energy Act (AEA). *Georgia Institute of Technology* (Georgia Tech Research Reactor), CLI-95-12, 42 NRC 111 (1995), *citing Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant) CLI-93-21, 38 NRC 87, 92 (1993). Accordingly, a petitioner must allege a concrete and particularized injury that is fairly traceable to the challenged action and likely to be redressed by a favorable decision. *Id., citing generally Lujan v. Defenders of Wildlife*, 112 S.Ct. 2130, 2136 (1992) and *Perry*, CLI-93-21 at 32.

Regarding claims to standing based on residence in the vicinity of a power plant, there is a presumption that a person whose base of normal activities is within 50 miles of the site has an interest that might be affected by reactor construction or operation. *Florida Power and Light Co.* (St. Lucie Nuclear Power Plant, Units 1 and 2), CLI-89-21, 30 NRC 325, 329 (1989). However, the Commission has held that standing based on living within a specific distance of a plant is limited to cases involving the construction or operation of the reactor itself "with clear implications for the offsite environment, or major alterations to the facility with a clear potential for offsite consequences." *Id.*

In order for an organization to establish standing, it must either demonstrate standing in its own right or claim standing through one or more individual members who have standing. *Georgia Institute of Technology* (Georgia Tech Research Reactor), CLI-95-12, 42 NRC 111 at 115. To establish standing through its members, an organization must show that at least one of its members suffers "immediate or threatened injury as a result of the challenged action of the sort that would make out a justiciable case had the members themselves brought suit . . ." *Warth v. Seldin*, 422 U.S. 490, 511 (1975). An organization normally must identify at least one member by name and address and must demonstrate that the named member has authorized the organization to represent that member in the proceeding. *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-535, 9 NRC 377, 393-96 (1979). An organization must normally submit affidavits of its members to demonstrate that it has authorization to represent them. *Sacramento Municipal Utility Dist.* (Rancho Seco Nuclear Generating Station), LBP-92-23, 36 NRC 120, 126 (1992), *rev'd on other grounds*, CLI-93-3, 37 NRC 135 (1993). An organization may not represent those individuals who are not members of the organization and may not base its claim for standing on those individuals. *Florida Power*

& Light Co. (Turkey Point Nuclear Generating Plant, Units 3 and 4), ALAB-952, 33 NRC 521, 530-31 (1991).

B. NIRS, OCNW and CAN have not established that they have standing through their members.

1. NIRS and OCNW appear to satisfy the residency requirements; CAN does not.

NIRS has provided the affidavit of one of its members, Mr. William deCamp, Jr., who has authorized NIRS to represent him in this proceeding. It appears from his affidavit that Mr. deCamp may live or do business near enough to Oyster Creek to satisfy the residency test. OCNW has provided an affidavit from one of its members, Ms. Jean Burnett, who has authorized OCNW and/or NIRS to represent her in this proceeding.[2] In her affidavit, Ms. Burnett states that she lives at Forked River Beach, whose residents must go toward the plant in order to get away from the plant, which is within a half mile. Thus it appears from Ms. Burdett's affidavit that she meets the residency test. Also, on page 4 of the Petition, the Petitioners assert that the affiants "have authorized their organizations to be represented through NIRS in this proceeding." As noted above, organizations may represent their own members only and not the members of other organizations. Thus, NIRS, if it were found to have standing through Mr. deCamp, could not represent Ms. Burnett, and it could not base its standing on Ms. Burnett.

CAN's Petition fails in that CAN does not identify at least one member by name and address who has standing in his or her own right and who has authorized CAN to represent him or her in this proceeding. CAN does not allege that any of its members reside in the vicinity of the

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[2] The affidavits of Shirley R. Schmidt and Maria Szczech do not state that they are members of OCNW, and thus cannot be used to support OCNW's claim for standing in this proceeding.

Licensee's facility or have any association with it. CAN claims an interest in the proceeding based on its member who is within seventeen miles of the Vermont Yankee Nuclear Power Station.

CAN's interest is based on a speculative injury that is too conjectural to support a showing of standing. CAN's alleged injury is not imminent or actual. Rather, the basis of the concern that an offsite consequence will occur at the Vermont Yankee Power Station is too remote from the amendment at issue in this proceeding to satisfy standing. Further, CAN has failed to show that "but for the particular action it challenges, its injury would abate." *Public Serv. Co. of New Hampshire*, (Seabrook Station, Unit 1), CLI-91-14, 34 NRC 261, 267 (1991). As the Commission has recognized, "[w]here the injury alleged does not stem directly from the challenged governmental action, but instead involves predicting the actions of third parties not before the court, the difficulty of showing redressability is particularly great." *Westinghouse Elec. Corp.* (Nuclear Fuel Export License for Czech Republic -- Temelin Nuclear Power Plants), CLI-94-07, 39 NRC 322, 332 (1994). CAN has failed to demonstrate that its alleged injury is redressable. Indeed, a decision in this proceeding to reject Oyster Creek's present amendment request would leave untouched any rights the operators of Vermont Yankee may have to request amendments or take other action regarding its spent fuel handling. Therefore, even if it had successfully identified an aspect on which it wished to intervene, CAN should not be granted intervention in this proceeding.

2. Petitioners have not shown that the injury they might suffer arises from the proposed amendment.

Petitioners state that NIRS and OCNW are entitled to intervene in this proceeding on behalf of their members living near the plant who would be injured by an accident caused by the inadequate or unsafe movement of heavy loads over or near irradiated fuel at the Oyster Creek Nuclear Generating Station. Petition at 2. As stated above, the amendment would allow the shield plug for the dry shield canister (DSC) and the associated lifting hardware to be moved over irradiated fuel in the DSC within the transfer cask located in the Cask Drop Protection System. It concerns a specific movement in the spent fuel pool. The scope of the "movement of heavy loads over or near irradiated fuel at the Oyster Creek Nuclear Generating Station" far exceeds the scope of the proposed amendment. Petitioners have failed to show that their injury would arise from the proposed amendment.

Petitioners state that the members of their organizations would suffer injury-in-fact by the proposed license amendment which would 1) increase the probability of an accident; 2) create the possibility of an accident not previously identified in the Safety Analysis Report and (3) constitute a significant reduction in the margin of safety. *Id.* Petitioners' comments on the Staff's proposed no significant hazards consideration determination do not constitute an injury redressable by the Board, as the Commission's regulation in 10 C.F.R. § 50.58(b)(6) leaves that determination to the Staff and the Commission. *See Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1, LBP-91-26, 33 NRC 537, 545 (1991) and LBP-91-23, 33 NRC 430, 442 (1991); *Sacramento Municipal Utility Dist.* (Rancho Seco Nuclear Generating Station), LBP-91-17, 33 NRC 379, 381 (1991); *Vermont Yankee Nuclear Power Corp.* (Vermont

Yankee Nuclear Power Station), LBP-90-6, 31 NRC 85, 90-91 (1990); *Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 and 4), LBP-89-15, 29 NRC 493, 499-500 (1989).

3. Petitioners' identification of aspects on which they seek intervention does not relate to the subject matter of the proposed amendment.

As set forth above, the Commission's regulations in 10 C.F.R. § 2.714(a)(2) require petitioners for intervention to set forth the specific aspect of the subject matter of the proceeding on which they wish to intervene. Under the heading "Aspects...on which petitioners seek to intervene," Petitioners state that they intend to contest the Staff's proposed no significant hazards consideration determination. Petition at 6. As discussed above, the Commission's regulations in 10 C.F.R. § 50.58(b)(6) do not allow contentions concerning that determination to be admitted in hearings but rather leave that matter to the Staff and the Commission. Under "Aspects," Petitioners also state that they base their concerns on four documents. NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety - Related Equipment" (April 11, 1996) is one of the documents Petitioners rely on as a basis for their concern and as identifying an aspect of the proceeding on which they wish to intervene. Petition at 6. However, Petitioners misread Bulletin 96-02. It is GPUN's § 50.59 analysis for moving the 100-ton cask that Bulletin 96-02 concerns, not a § 50.59 analysis for movement of the dry cask shield plug. Bulletin 96-02 at 3. Thus, Petitioners' identification of aspect rests on a mistaken reading of the Staff document on which they rely.

Petitioners also place reliance on a recent Information Notice, IN 96-26: Recent Problems with Overhead Cranes, as a basis for their concern. Petitioners do not indicate how or why they believe this IN is applicable to Oyster Creek and it is not apparent on its face that it has any applicability to Oyster Creek. The IN concerns overhead cranes at two PWR's, Trojan and Prairie Island. The problem at Trojan was a construction problem and the problem at Prairie Island was a calibration problem, which does not appear to be applicable to the proposed amendment.

The Daily Report concerning Indian Point on which Petitioners rely reports a container drop onto the fuel handling floor. Again the only apparent relationship between the incident at Indian Point and the proposed action at Oyster Creek is that both involve lifting. This generalized concern about lifting does not constitute identification of an aspect of the proposed amendment on which Petitioners seek intervention. The Preliminary Notice concerning an event at Hatch on which Petitioners rely also concerns lifting but that event did not involve a heavy load. It does not relate to any aspect of the proceeding in which Petitioners seek intervention.

Thus, Petitioners have failed to identify an aspect of the subject matter of the proceeding on which they seek to intervene.

II. Discretionary Intervention

A. Standards for Discretionary Intervention

Although a petitioner may lack standing to intervene as of right under judicial standing concepts, he may nevertheless be admitted to the proceeding in the Licensing Board's discretion. In determining whether discretionary intervention should be permitted, the Commission has indicated that the Licensing Board should be guided by the following factors, among others:

(a) Weighing in favor of allowing intervention --

(1) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.

(2) The nature and extent of the petitioner's property, financial, or other interest in the proceeding.

(3) The possible effect of any order which may be entered in the proceeding on the petitioner's interest.

(b) Weighing against allowing intervention --

(4) The availability of other means whereby petitioner's interest will be protected.

(5) The extent to which the petitioner's interest will be represented by existing parties.

(6) The extent to which petitioner's participation will inappropriately broaden or delay the proceeding.

Portland General Electric Co. (Pebble Springs Nuclear Plant, Units 1 & 2), CLI-76-27, 4 NRC 610, 616 (1976). The discretionary intervention doctrine comes into play only in circumstances where standing to intervene as a matter of right has not been established. *Duke Power Company* (Oconee Nuclear Station and McGuire Nuclear Station), ALAB-528, 9 NRC 146, 148 n.3 (1979).

The primary factor to be considered is the significance of the contribution that a petitioner might make. *Pebble Springs, supra.* Thus, foremost among the factors listed above is whether the intervention would likely produce a valuable contribution to the NRC's decisionmaking process on a significant safety or environmental issue appropriately addressed in the proceeding in question. *Tennessee Valley Authority* (Watts Bar Nuclear Plant, Units 1 & 2), ALAB-413, 5 NRC 1418 (1977). *See also Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-470, 7 NRC 473, 475 n.2 (1978); *Sacramento Municipal Utility District* (Rancho Seco Nuclear Generating Station), LBP-92-23, 36 NRC 120, 131-32 (1992).

The need for a strong showing as to potential contribution is especially pressing in an operating license proceeding where no petitioners have established standing as of right and where, absent such a showing, no hearing would be held. *Watts Bar, supra*, 5 NRC at 1422. Where there are no intervenors as of right, a Licensing Board will determine whether a discernible public interest would be served by ordering a hearing based on a grant of discretionary intervention. *Envirocare of Utah, Inc.*, LBP-92-8, 35 NRC 167, 183-84 (1992).

For discretionary intervention, the burden of convincing a Licensing Board that a petitioner could make a valuable contribution lies with the petitioner. *Nuclear Engineering Co., Inc.* (Sheffield, Ill. Low-Level Radioactive Waste Disposal Site), ALAB-473, 7 NRC 737, 745 (1978). Considerations in determining the petitioner's ability to contribute to development of a sound record include:

(1) a petitioner's showing of significant ability to contribute on substantial issues of law or fact which will not be otherwise properly raised or presented;

(2) the specificity of such ability to contribute on those substantial issues of law or fact;

(3) justification of time spent on considering the substantial issues or law or fact;

(4) provision of additional testimony, particular expertise, or expert assistance;

(5) specialized education or pertinent experience.

Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), LBP-81-1, 13 NRC 27, 33 (1981) (and cases cited therein). *See Florida Power and Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 and 4), LBP-90-24, 32 NRC 12, 16-17 (1990), *aff'd*, ALAB-952, 33 NRC 521, 532 (1991).

As to the second and third factors to be considered with regard to discretionary intervention (the nature and extent of property, financial or other interests in the proceeding and the possible effect any order might have on the petitioner's interest), interests which do not establish a right to intervention because they are not within the "zone of interests" to be protected by the Commission should not be considered as positive factors for the purposes of granting discretionary intervention. *Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), LBP-78-11, 7 NRC 381, 388, aff'd, ALAB-470, 7 NRC 473 (1978).

B. Petitioners have not Satisfied the Standards for Discretionary Intervention.

Petitioners assert that if not found to have standing as of right they should be granted discretionary intervention, citing *Pebble Springs* in support of this assertion. Petition at 4. Petitioners state that NIRS and OCNW have filed petitions under 10 C.F.R. § 2.206 concerning reactor components, fire protection, and design deficiencies in the spent fuel pool cooling system at Oyster Creek. Petition at 4-5. They also mention a public forum that they held on the Mark I pressure suppression system, cracking of the drywell concrete shield wall and corrosion of the drywell steel liner. Id. With regard to CAN, Petitioners state that granting the proposed amendment will establish a precedent for similar plants, namely Vermont Yankee. Petition at 5. Neither NIRS/OCNW nor CAN makes a convincing argument that their participation would constitute a valuable contribution or would contribute to a sound record. Neither cites hearing experience or indicates that it has expert knowledge of issues related to the proposed action. Rather, as discussed above, Petitioners seem to have confused the proposed action with movement of the 100-ton cask, which is the subject matter of the NRC Bulletin on which Petitioners rely but which is not the subject of the amendment request. Thus, Petitioners have failed to satisfy the first of the *Pebble Springs* factors for discretionary intervention. The other two factors, which are of less importance, are not directly addressed in Petitioners' argument supporting discretionary intervention.

Accordingly, if the Board finds that Petitioners have not established standing as of right, it should not grant the discretionary intervention that Petitioners seek in the alternative.

CONCLUSION

As discussed above, none of the Petitioners has established standing to intervene in a hearing on the proposed amendment. Therefore, the Licensing Board should deny the Petition.

Respectfully submitted,

Ann P. Hodgdon

Ann P. Hodgdon
Counsel for NRC Staff

Richard G. Bachmann
Counsel for NRC Staff

Dated at Rockville, Maryland
this 26th day of June 1996

DOCKETED
JUN 27 1996
DOCKETING & SERVICE BRANCH
SECY-NRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITY NUCLEAR CORPORATION	)	Docket No. 50-219-OLA
	)	
(Oyster Creek Nuclear Generating Station)	)	

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney enters an appearance in the above-captioned matter. In accordance with § 2.713(b), 10 C.F.R., Part 2, the following information is provided:

Name:	Ann P. Hodgdon
Address:	Office of the General Counsel U.S. Nuclear Regulatory Commission Washington, D.C. 20555
Telephone Number:	(301) 415-1587
Admissions:	U.S. Court of Appeals, District of Columbia
Name of Party:	NRC Staff

Respectfully submitted,

Ann P Hodgdon

Ann P. Hodgdon
Counsel for NRC Staff

Dated at Rockville, Maryland
this 26th day of June, 1996.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITY NUCLEAR CORPORATION	)	Docket No. 50-219-OLA
	)	
	)	
(Oyster Creek Nuclear Generating Station)	)	

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney enters an appearance in the above-captioned matter. In accordance with § 2.713(b), 10 C.F.R., Part 2, the following information is provided:

Name:	Richard G. Bachmann
Address:	Office of the General Counsel U.S. Nuclear Regulatory Commission Washington, D.C. 20555
Telephone:	(301) 415-1574
Admissions:	Supreme Court for the State of California
Name of Party:	NRC Staff

Respectfully submitted,

Richard G. Bachmann
Counsel for NRC Staff

Dated at Rockville, Maryland
this 26th day of June, 1996.

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:

April 15, 1996
6730-96-2087

U.S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington D.C. 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

In accordance with 10 CFR 50.4(b)(1), enclosed is a Technical Specification Change Request (TSCR) No. 244. Also enclosed is a Certificate of Service for this request certifying service to the chief executive of the township in which the facility is located, as well as the designated official of the State of New Jersey Bureau of Nuclear Engineering.

The purpose of this TSCR is to revise specification 5.3.1.B of the Oyster Creek Technical Specifications. The current specification prohibits handling a load greater in weight than one fuel assembly over irradiated fuel in the spent fuel storage facility. The proposed change will facilitate the off load of spent fuel to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI). Specifically, the shield plug for the dry shielded canister (DSC) and the associated lifting hardware will be moved over irradiated fuel which is contained in the DSC within the transfer cask located in the Cask Drop Protection System (CDPS).

Pursuant to 10 CFR 50.91(a)(1), enclosed is an analysis applying the standards of 10 CFR 50.92 to make a determination of no significant hazards consideration.

Very truly yours,

Michael B. Roche

Michael B. Roche
Vice President & Director
Oyster Creek

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

A001 1/1

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:

April 15, 1996
C321-96-2087

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, NJ 08625

Dear Mr. Tosch:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

Pursuant to 10 CFR 50.91(b)(1), please find enclosed a copy of the subject document which was filed with the United States Nuclear Regulatory Commission on April 15, 1996.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachment
DPK/plp

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

April 15, 1996
C321-96-2087

The Honorable John C. Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Dear Mayor Parker:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

Enclosed herewith is one copy of Technical Specification Change Request No. 244, for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the United States Nuclear Regulatory Commission on April 15, 1996.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachment

DPK/gl

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
GPU Nuclear Corporation)
Docket No. 50-219)

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 244, for Oyster Creek Nuclear Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on April 15, 1996, has this day of April 15, 1996, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable John Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

BY Michael B. Roche

Michael B. Roche
Vice President and Director
Oyster Creek

GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Facility Operating
License No. DPR-16

Technical Specification Change Request No. 244
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 244, to the Oyster Creek Nuclear Generating Station Operating License, a change to pages 5.3-1 and 5.3-2.

BY Michael B. Roche
Michael B. Roche
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 15th day of April 1996.

Geraldine E. Levin
A Notary Public of NJ

GERALDINE E. LEVIN
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 06/28/2000

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 65°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

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) No. 244

GPU Nuclear requests the following replacement pages be inserted into existing Technical Specifications:

Replace existing pages 5.3-1 and 5.3-2 with the attached revised replacement pages 5.3-1 and 5.3-2.

II. REASON FOR CHANGE

The current specification 5.3.1.B requires that "Loads greater than weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility". This restriction is based upon the structural strength of the fuel racks in which the spent fuel is stored and the damage that would occur if the load were dropped. The process of transferring spent fuel assemblies to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI) includes placing a dry shielded canister (DSC) and a transfer cask in the cask drop protection system (CDPS). That movement does not handle 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 pool, the DSC shield plug must be lowered into the CDPS and placed atop the DSC. The current specification prohibits this movement since the shield plug and the lifting yoke weigh more than one fuel assembly and the DSC contains irradiated fuel.

III. SAFETY EVALUATION JUSTIFYING CHANGE

GPU Nuclear has evaluated the process of transferring spent fuel assemblies from the spent fuel pool to the ISFSI. That evaluation considers the safe load paths, the design features of the reactor building crane and the requirements of NUREG 0612.

The CDPS has been designed to mitigate a cask drop into the spent fuel pool. The transfer path for the cask centerline is on a controlled path width of six inches in the north-south and east-west directions. Visual aids are used to control the motion of the cask centerline to the prescribed transfer path. Mechanical rail stops are installed to prevent travel of the crane outside the analyzed load path over the cask drop protection system. Stops are installed for limiting bridge movements in the north-south direction and for limiting trolley movements in the west direction. The movement of the shield plug would be in accordance with these same constraints. The weight of the load, however, would be considerably different. The shield plug weighs approximately 8,000 pounds and the lifting yoke weighs about 6,200 pounds.

A series of modifications have been made to enhance the crane's performance and reliability by improving the instrumentation and controls. These modifications include:

- Various crane monitoring systems have been installed. These include drum over-speed detection, mechanical drive train continuity detection, wire rope spooling monitor, fault display and reset panel and hoist speed indication.

- Phase loss/phase reversal protection has been installed. Phase loss results in substantial loss of drive motor torque and possible load drop.

- A power circuit upper limit switch to directly interrupt power to the hoist motor was installed. This reduces the possibility of two-blocking as a result of failure of existing control circuit limit switches.

- A load cell weight display was installed in the cab to provide an indication for load hang-ups and over-capacity lifts.

- The magnetic drive controllers were replaced. The new variable frequency drive (VFD) controllers provide smooth and precise speed control along with torque limitation, reducing the possibility of a load snatch.

- New controls were installed in the cab that provide spring control to normal function. These controls considered human factors in their design.

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. 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 will be provided, based on the RB crane 100 ton rated capacity. For the lifting yoke, a safety factor greater than 26 will be provided, based on the lifting yoke 105 ton rated capacity. The least conservative safety factor is that for the wire rope assemblies. That safety factor is 11:1, based on the ultimate load of 22,800 lbs. Furthermore, the wire rope assemblies are redundant and each of the four has sufficient capacity to support the total weight of the shield plug.

In addition, GPU Nuclear has developed an error free plan for the movement of spent fuel assemblies to the ISFSI. That plan includes a dedicated management team and a dedicated crew who will be trained and on shift. Detailed operating instructions/procedures will be developed and mock-up training and a dry run will be conducted. A special crane inspection will be performed prior to each dry fuel storage campaign. The main hoist coupling, shafts, and hook will be examined by NDE prior to each campaign. Plant procedures for the reactor building crane satisfy the inspection, testing and maintenance criteria of ANSI B30.2.

The design features and modifications to the reactor building crane increase its reliability and enhance its performance. The safety factors of the reactor building crane relative to this load exceed 10 to 1. Personnel training, and crane inspections, testing, and maintenance will be in accordance with ANSI B30.2. Therefore, dropping the DSC shield plug onto a loaded DSC in the spent fuel pool is not considered a credible event.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this TSCR poses no significant hazard as defined by 10 CFR 50.92.

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

The design features and capacity of the reactor building crane provide a significant safety factor. In addition, personnel training and other administrative controls further reduce risk. Thus, the dropping of the DSC shield plug onto a loaded DSC and causing damage to the spent fuel assemblies is not a credible event. Therefore, it does not increase the probability of or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR.

This activity will not create the possibility of a new or different type of accident than previously evaluated in the SAR because the proposed heavy load handling exception does not create a new credible accident scenario. Dropping the shield plug on a loaded DSC and damaging spent fuel assemblies is not considered a credible event.

3. State the basis for the determination that the margin of safety is not reduced.

This activity will not involve a significant reduction in the margin of safety because the proposed heavy load handling evolution does not create a credible accident scenario.

V. IMPLEMENTATION

GPUN requests that the amendment authorizing this change be effective upon issuance.

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.

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'96 JUN 27 A10:43

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITY NUCLEAR CORPORATION	)	Docket No. 50-219-OLA
	)	
(Oyster Creek Nuclear Generating Station)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE IN OPPOSITION TO REQUEST FOR HEARING AND PETITION TO INTERVENE OF NUCLEAR INFORMATION AND RESOURCE SERVICE, OYSTER CREEK NUCLEAR WATCH AND CITIZENS AWARENESS NETWORK " "NOTICE OF APPEARANCE" for Ann P. Hodgdon and Richard G. Bachmann, and the following documents: April 15, 1996 GPU Amendment Request (TSCR No. 244) and Oyster Creek Technical Specification 5.3.1.B. in the above-captioned proceeding have been served on the following through deposit in the Nuclear Regulatory Commission's internal mail system, or by deposit in the United States mail, first class, as indicated by an asterisk this 26th day of June, 1996:

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