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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-219-OLA
GPU NUCLEAR CORPORATION	)	(Tech. Spec. 5.3.1.B)
	)	
(Oyster Creek Nuclear Generating Station)	)	ASLBP No. 96-717-02-OLA

LICENSEE'S MOTION FOR SUMMARY DISPOSITION

I. INTRODUCTION

GPU Nuclear Corporation ("GPUN" or "Licensee") submits this motion for summary disposition pursuant to the Atomic Safety And Licensing Board's ("Board") October 25, 1996 Memorandum and Order Ruling on Intervention Petition of Nuclear Information and Resource Service, Oyster Creek Nuclear Watch, and Citizens Awareness Network ("Petitioners").[1] Consistent with that Memorandum and Order, the motion addresses the sole legal issue remaining in this proceeding. A Statement of Material Facts as to Which There is no Genuine Dispute, an Affidavit of John C. Fornicola ("Fornicola Aff."), and exhibits are attached in support.

The gravamen of Petitioners' position is that Technical Specification 5.3.1.B of the Oyster Creek license is a vital control prohibiting the movement of heavy loads in the Cask Drop

[1] ASLB Memorandum and Order, Ruling on Intervention Petition, LBP-96-23, October 25, 1996 (hereinafter LBP-96-23).

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Protection System ("CDPS") and, as such, cannot be changed as a matter of law. For the reasons discussed in this motion, this position is legally untenable because Technical Specification 5.3.1.B only applies to heavy loads moved over stored fuel in the spent fuel storage racks and is no legal impediment to the movement of heavy loads over spent fuel in the CDPS.

II. BACKGROUND

The underlying action of the proceeding is the Licensee's request to modify a technical specification for the Oyster Creek Nuclear Generating Station ("Oyster Creek"). On April 15, 1996, GPUN requested that Technical Specification 5.3.1.B be modified to add a second sub-part[2] to read as follows:

> 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 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.

Fornicola Aff., ¶ 3 and Exh. 2 (GPUN Corporation, Oyster Creek Nuclear Generating Station, Technical Specification Change Request No. 244 (Apr. 15, 1996)). Pursuant to a final no significant hazards consideration determination, the NRC issued this amendment on November 7, 1996. U.S. Nuclear Regulatory Commission, Issuance of Amendment Re: Handling Heavy Loads Over Irradiated Fuel (Nov. 7, 1996), attached as Exhibit A hereto. Petitioners challenge this change.

[2] Prior to the amendment, Technical Specification 5.3.1.B read "Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility."

The legal issue in the proceeding established by the Board is from Petitioner's contention. The Board has summarized Basis C as follows:

> The NRC's fundamental regulatory defense-in-depth principle is implemented in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," which is the equivalent of a regulatory guide. Because OCNGS does not employ a single failure proof crane for shield plug movement, consistent with NUREG-0612 guidelines as described in enclosure 1 to NRC Generic Letter 85-11 (June 28, 1985), GPUN must rely on analyzed safe load paths and restricted load limits for movement of heavy loads to "assure, to the extent practical" that heavy loads are not carried over or near irradiated fuel. Although GPUN claims in its safety evaluation regarding the proposed technical specification change that a shield plug drop accident is not credible because of GPUN administrative controls (e.g., rail stops), operator training, and inspections concerning dry-storage related spent fuel movements, this does not adequately address human error or mechanical/electrical failure issues. Rather, the most effective way to avoid such failures is to restrict both human-directed activity and prohibit the movement of heavy loads as is done with current [prior to the November 7, 1996 amendment] Technical Specification 5.3.1.B. As such, consistent with the agency's NUREG-0612 defense-in-depth guidance, the [pre-] existing provision cannot be revised as the licensee has requested.

LBP-96-23 at 12.

Regarding Basis C, the Board states that "petitioners seek to establish the 'single fuel assembly' weight limitation in [pre-]existing Technical Specification 5.3.1.B reflects an agency judgment about the particular measures that are necessary for compliance with the purported regulatory guidance in NUREG-0612 as it is asserted to implement the 'defense-in-depth' principle." Id. at 40. Petitioners assert that "this weight limitation [in Technical Specification 5.3.1.B] is a vital control meant to remove the potential that human error or any mechanical/electrical

failure could cause damage to irradiated fuel." Id. at 40 (citing Petitioners' statements in the Pre-Hearing Conference, Tr. at 68). In summary, Petitioners assert that "[b]ecause of the importance of this limitation . . . , this technical specification cannot be changed." Id. at 40-41 (emphasis added).

The Board identifies two factors that provide "sufficient reason to conclude Basis C establish[es] a material disputed issue of law that should be considered further." Id. at 41. The first factor is that, based on the fact that Licensee's CDPS has been in place for some time,[3] the Licensee and the NRC staff "had some notion that GPUN at some point could be in a position to place an object heavier that a fuel assembly over fuel assemblies being packaged for removal and storage," but "[n]onetheless, the [pre-]existing technical specification with its specific 'fuel assembly' weight limitation seemingly was adopted for OCNGS after NUREG-0612 was issued with its 'to the extent practicable' language."[4] Id. at 41, citing U.S. Nuclear Regulatory Commission, NUREG-0612, Control of Heavy Loads at Nuclear Power Plants (July 1980) (hereinafter NUREG-0612), at 3-9 (Table 3.2-1), 5-2 . The implication is that both the Licensee and the NRC staff realized that the original Technical Specification 5.3.1.B was adopted to prohibit the Licensee from ever packaging spent fuel into casks for removal from Oyster Creek.

[3] Fornicola Aff., ¶ 8. The Cask Drop Protection System ("CDPS") was approved by the Atomic Energy Commission in 1973 and subsequently installed at Oyster Creek.

[4] Although the assumption is easy to make, Technical Specification 5.3.1.B was actually adopted in 1977, more than three years prior to NUREG-0612, in response to an NRC staff request related to changing out the spent fuel storage racks at Oyster Creek, as is discussed infra. See Fornicola Aff., ¶ 6.

The second factor identified by the Board as providing sufficient reason to establish a material disputed issue of law is that the licensee and NRC staff "have asserted that NUREG-0612 is simply 'guidance' that contains no regulatory mandate," while at the same time "there are any number of references to NUREG-0612 'requirements' in the Licensee and agency documents provided to [the Board]." LBP-96-23 at 42 (citing the Pre-Hearing Conference, Tr. at 99-101, as well as the Certificate of Compliance for the NUHOMS system). This, the Board states, raises a "legitimate question about the regulatory significance of [NUREG-0612] and its 'to the extent practical' language." Id. at 42. Although it does not raise any question about the adequacy of GPUN's load handling training or procedures (id. at 41 n.19), Petitioners' Basis C, as summarized by the Board, taken together with the two above factors, poses a "matter of legal interpretation that merits further scrutiny." Id.

The legal issue in Basis C, as summarized by the Board, is the only remaining issue in this proceeding as all of the Petitioners' other contentions and bases were dismissed as inadequate to establish a material dispute warranting further inquiry. Id. at 36, 39-40.

III. STATEMENT OF ISSUE

The disputed issue of law established by the Board within this proceeding can be summarized as follows:

> A. What is the Regulatory Significance of NUREG-0612 and is Technical Specification 5.3.1.B Required Pursuant to NUREG-0612.

B. May Technical Specification 5.3.1.B be Changed to Allow the Movement of Heavy Loads over Spent Fuel in the Cask Drop Protection System.

Regarding Part A, GPUN maintains that the true regulatory significance of NUREG-0612 is only through the generic letters issued by the NRC requesting licensee implementation of selected parts of NUREG-0612. Further, the adoption of a technical specification comparable to 5.3.1.B was not one of the selected parts of NUREG-0612 which licensees were requested to implement by these generic letters.

Regarding Part B, GPUN maintains that the change to Technical Specification 5.3.1.B was permissible for several reasons. First, the technical specification, which was originally issued before NUREG-0612, has always applied only to stored spent fuel in the fuel storage area and not to spent fuel in the CDPS being packaged for movement out of the reactor building. The recent amendment is merely a clarification of this intended scope and meaning. Further, even if the technical specification were interpreted in light of the subsequent NUREG-0612 recommendations (including the interim recommendation that was not adopted in the generic letters), its scope and meaning would be unaltered. The interim technical specification recommended by NUREG-0612 also applies only to spent fuel stored in racks.

Accordingly, there is no legal restriction to the amendment to Technical Specification 5.3.1.B at issue in this proceeding. The Board should therefore decide the legal issue consistent with the Licensee's position and terminate this proceeding.

IV. ARGUMENT

A. NUREG-0612 REFLECTS RECOMMENDATIONS OF AN NRC STAFF TASK GROUP BUT IS NOT A REGULATION. WHILE CERTAIN OF THOSE RECOMMENDATIONS WERE ADOPTED BY GENERIC LETTERS, THE RECOMMENDATION FOR A TECHNICAL SPECIFICATION WAS NOT.

1. NUREG-0612 Reflects Only Recommendations, and the Recommendations which are of Regulatory Significance are those Adopted in NRC Generic Letters Issued to Licensees

Pursuant to section 103 the Atomic Energy Act ("AEA"), the NRC is authorized to grant and modify commercial licenses for nuclear power plants "subject to such conditions as the Commission may by rule or regulation establish to effectuate the purposes of this Act." AEA § 103, 42 U.S.C. § 2133 (1994). The stated purposes of licensing under the AEA are to maintain activities "in accord with the common defense and security and (to) provide adequate protection to the health and safety of the public." AEA § 182, 42 U.S.C. § 2232 (1994).

Pursuant to the AEA, the NRC has established requirements specific to the movement of heavy loads at nuclear power plants in General Design Criteria established in 10 C.F.R. § 50.34 and 10 C.F.R. Part 50, Appendix A. Compliance with the General Design Criteria "provide[s] reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." 10 C.F.R. Part 50, App. A (introduction). General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," establishes the regulatory requirements for heavy load movements as follows:

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit

> appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Id. at Criterion 61.

The NRC issues NUREGs and Regulatory Guides as advisory guidance providing approaches licensees can follow to meet legal requirements. Curators of the University of Missouri, CLI-95-8, 41 N.R.C. 386, 397 (1995). "[I]t is well established . . . that NUREGs and Regulatory Guides, by their very nature, serve merely as guidance and cannot prescribe requirements." Curators of the University of Missouri, CLI-95-1, 41 N.R.C. 71, 98 (1995). Although conformance with the guidance will likely result in compliance with underlying regulations, nonconformance with the guidance does not equate to noncompliance with the regulations. Id.

NUREG-0612 was initiated to "systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to assure the safe handling of heavy loads. . . ." NUREG-0612 at 1-1. NUREG-0612 develops "recommendations on actions that should be taken" based on an NRC task group's evaluation of NRC "licensing criteria and the adequacy of measures in effect at operating plants." Id. At no time does the document maintain or assert that NUREG-0612 itself establishes requirements that licensees must comply with. NUREG-0612 characterizes itself as "a summary of those recommended actions that should be taken to resolve the concern over the handling of heavy loads near irradiated fuel, or safety related equipment." Id. at 6-1 (emphasis added).

The NRC adopted certain recommendations from NUREG-0612 through generic letters to all licensees. The first generic letter, dated December 22, 1980, requested all licensees to implement selected interim actions, listed in Enclosure 2 of the letter, provide information showing how their facility satisfied the guidelines of NUREG-0612, and demonstrate implementation of the criteria of NUREG-0612 § 5.1.1 over six months and NUREG-0612 §§ 5.1.2-5.1.5 over 9 months. U.S. Nuclear Regulatory Commission, Unnumbered Generic Letter on Control of Heavy Loads (Dec. 22, 1980), attached as Exhibit B hereto, at 2.[5] The second generic letter, 85-11, stated that only the criteria of NUREG-0612 § 5.1.1 must be implemented (referred to as "Phase 1"), and rescinded the prior request to implement NUREG-0612 §§ 5.1.2-5.1.5 (referred to as "Phase II"). U.S. Nuclear Regulatory Commission, Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612" (June 28, 1985), attached as Exhibit D hereto. Thus, it was only through these generic letters that licensees were requested to comply with any of the guidelines of NUREG-0612, and the generic letters only request implementation of selected recommendations from NUREG-0612, not all of the "Recommended Guidelines" in the report.

2. The Generic Letters Do Not Require the Implementation of a Technical Specification Comparable to 5.3.1.B

One of the recommendations made in NUREG-0612 was an "Interim Protection" measure that:

[5] This letter does not have a Generic Letter identifier, but subsequent NRC documentation on the subject refers to it as "the December 22, 1980 generic letter on 'Control of Heavy Loads'." See U.S. Nuclear Regulatory Commission, Control of Heavy Loads (Phase 1) - NUREG-0612 - Oyster Creek Nuclear Generating Station (June 21, 1983), attached as Exhibit C hereto, at 1.

(1) Licenses for all operating reactors not having a single-failure-proof overhead crane in the fuel storage pool area should be revised to include a specification comparable to Standard Technical Specification 3.9.7, "Crane Travel - Spent Fuel Storage Pool Building" for PWR's and Standard Technical Specification 3.9.6.2, "Crane Travel," for BWR's, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1 (see Table 3.2-1).

NUREG-0612 at 5-18. This is the recommendation that is comparable to Technical Specification 5.3.1.B, which was already in place at Oyster Creek. Five other "Interim Protection" recommendations were also developed in NUREG-0612. Id. at 5-18 to 5-19.

The generic letters only requested licensee implementation of the last five "Interim Protection" recommendations, not "Interim Protection" recommendation (1) -- the Technical Specification guideline. The NRC requested licensees to "implement the interim actions described in Enclosure 2 [to the Dec. 22, 1980 Generic Letter] . . . ." Exh. B at 2. The "interim actions" in Enclosure 2 to this generic letter are the five "Interim Protection" measures numbered (2) through (6). Id. at Encl. 2. Compare NUREG-0612 at 5-18. Interim Protection recommendation (1) of NUREG-0612, calling for a Technical Specification comparable to Oyster Creek Technical Specification 5.3.1.B., is not included in the December 22, 1980 generic letter and is not made a requirement in any other NRC document.

B. TECHNICAL SPECIFICATION 5.3.1.B MAY BE CHANGED TO ALLOW THE MOVEMENT OF HEAVY LOADS OVER SPENT FUEL IN THE CASK DROP PROTECTION SYSTEM WHERE THE MEANING AND HISTORY OF THE TECHNICAL SPECIFICATION SHOW IT HAS ALWAYS APPLIED ONLY TO STORED SPENT FUEL IN THE FUEL STORAGE AREA

As discussed below, the recent change to Technical Specification 5.3.1.B explicitly allowing the movement of heavy loads over spent fuel in the CDPS merely clarifies the original intent and meaning of the technical specification. Technical Specification 5.3.1.B has always applied, by both wording and intent, to "stored" spent fuel. GPUN applied for the recent technical specification change, at the suggestion of the NRC staff and out of an abundance of caution, only to make this meaning more explicit. Fornicola Aff., ¶ 23. Thus, Technical Specification 5.3.1.B is not, and has never been a vital control intended to prevent movement of heavy loads over the CDPS, and the change is in fact nothing more than a non-substantive clarification.

In the same vein, the original Technical Specification 5.3.1.B was not issued after or in response to NUREG-0612. Rather, it was issued three years earlier (i.e., in 1977) to address the potential for load drops over fuel storage racks that were being modified. In any event, even if this technical specification were interpreted in a manner consistent with the recommendations of NUREG-0612, its intended scope would not change, because the technical specification recommended by NUREG-0612 applies only to fuel in storage racks. Thus, none of the recommendations of NUREG-0612 would prevent the wording change proposed by GPUN.

1. The Plain Meaning of Technical Specification 5.3.1.B Indicates it Applies Only to Stored Irradiated Fuel in the Fuel Storage Facility

Under its literal terms,[6] Technical Specification 5.3.1.B is and has always been a prohibition against moving certain loads over fuel that is "stored" in a "storage" facility. As originally adopted and prior to the November 7, 1996 amendment, Technical Specification 5.3.1.B stated:

> 5.3.1.B. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility. (emphasis added).

In ordinary usage, "stored" means placed or left in a location, (as a warehouse) for preservation or later use.[7] A storage facility would thus be a location where things are placed and left for later use. This meaning is fully consistent with spent fuel that is placed in the storage racks of that part of the spent fuel pool set aside to leave spent fuel in for future use. On the other hand, this meaning is inconsistent with spent fuel transferred to the Cask Drop Protection System for packaging into a cask for immediate movement out of the building. If "stored" means anything, it means something that is not in the midst of a packaging operation where it is being prepared for transport out of the facility.

The requirement uses the specific terms "stored irradiated fuel" and "spent fuel storage facility" rather than the broader terms "fuel" and "spent fuel pool." The term "irradiated" is used to differentiate it from new or "fresh" fuel which is stored in a different location. The use of the

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[6] Textual provisions of requirements should be interpreted using the plain meaning of the terms. See Estate of Cowart v. Nicklos Drilling Co., 505 U.S. 469, 112 S. Ct. 2589, 2594 (1992) (applying the canon to interpretation of a statute).

[7] Webster's Ninth New Collegiate Dictionary 1162 (1987).

term "stored" irradiated fuel instead of simply "irradiated fuel" indicates a specific intent to limit the requirement to "stored" fuel, and not to any other fuel, including that being "packaged" for transport.[8] The use of the term "storage" facility indicates that the "packaging" area -- the CDPS -- is not included. This careful wording shows that the textual meaning of Technical Specification 5.3.1.B has always been limited in scope to fuel stored in the storage racks area of the spent fuel pool, and that the scope has never included fuel being packaged in the CDPS.

This plain meaning is consistent with prior NRC Staff interpretation. Prior to the adoption of the original Technical Specification 5.3.1.B, 224 spent fuel assemblies were packaged in transportation casks and shipped to the Nuclear Fuel Services' reprocessing plant in West Valley, New York. Fornicola Aff., ¶¶ 6 and Exh. 3 (U.S. Nuclear Regulatory Commission, Issuance of Amendment No. 22 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Mar. 30, 1977)) at page 3 of environmental impact appraisal. In loading the transportation cask, a heavy shielded lid was lowered over the spent fuel assemblies in the cask within the CDPS. Fornicola Aff., ¶ 20. In 1984 and 1985, some seven years after the adoption of Technical Specification 5.3.1.B and four years after the publication of NUREG-0612, the same 224 spent fuel assemblies were returned to Oyster Creek from West Valley.[9] Unloading the spent fuel from the transportation cask in the CDPS involved lifting the shielded transportation cask lid, which is heavier than one fuel assembly, over spent fuel assemblies in the

[8] The expression of one thing in a requirement indicates the exclusion of others: *expressio unius est exclusio alterius*. See O'Melveny & Myers v. FDIC, 512 U.S. 579, 114 S. Ct. 2048, 2054 (1994) (applying this canon to interpretation of a statute).

[9] Fornicola Aff., ¶ 21.

transportation cask.[10] This history reflects the NRC staff's and Licensee's understanding that Technical Specification 5.3.1.B did not apply to spent fuel packaging operations in the CDPS.

This plain meaning is also consistent with prior NRC Staff interpretation for other facilities. The same situation occurred in 1992 at the Consumers Power Company ("Consumers") Palisades Nuclear Plant ("Palisades") in preparation to move spent fuel out of the facility and into dry storage. Palisades Technical Specification 3.21.2 prohibits the movement of heavy loads "over fuel stored in the main pool zone." Fornicola Aff., ¶ 22 and Exh. 9 (Consumers Power Company, Response to NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment (May 16, 1996)) at page 2 of attachment (emphasis added). This technical specification is comparable to Oyster Creek Technical Specification 5.3.1.B (as it read prior to the November 7, 1996 amendment) because it specifically addresses heavy load movements over stored fuel. Palisades' dry storage activities required shield covers and other heavy loads to be moved over spent fuel in the transfer cask in the packaging area of the pool. Id. at pages 2-3 of attachment. Based on several conversations between Consumers and the NRC staff, it was determined that spent fuel in the transfer cask was not subject to the requirements of Technical Specification 3.21.2 because the technical specification applied only to stored fuel and fuel in the transfer cask is not being stored but is rather in transit. Id. at page 3 of attachment. The NRC staff's conclusion on the non-applicability of Palisades Technical Specification 3.21.2 to spent fuel in a transfer cask in the cask packaging area is

[10] Fornicola Aff., ¶ 21.

consistent with the non-applicability of Oyster Creek Technical Specification 5.3.1.B to spent fuel in the transport cask in the CDPS.

This understanding is also reflected in the NRC's Safety Evaluation supporting the November 7, 1996 amendment to Technical Specification 5.3.1.B. The NRC staff stated:

> [t]he current [technical specification] 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 stored 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 [technical specification] to clarify that the shield plug may be moved onto the DSC after the DSC has been loaded with irradiated fuel.

Exh. A at page 1 of Safety Evaluation. This again indicates that Technical Specification 5.3.1.B refers only to "stored" fuel and spent fuel in the transfer cask within the CDPS is not "stored" fuel.

The current change to Technical Specification 5.3.1.B was not sought because of any change in meaning or intent. Rather, in light of the current regulatory climate in which both the NRC and licensees are particularly sensitive to the need for a well-defined and understood licensing basis, it was decided that an amendment clarifying the technical specification would be desirable. The purpose of the technical specification change is merely to make it more explicit.

2. NRC Staff and Licensee Documents Related to Technical Specification 5.3.1.B Establish that its Intent and Scope Is Spent Fuel Stored in Storage Racks and Not Spent Fuel Being Packaged for Transport in the CDPS

While the meaning of the original Technical Specification 5.3.1.B is clear on its face, the regulatory history of the provision confirms that it was always intended to apply only to spent fuel stored in the storage racks.[11] Technical Specification 5.3.1.B was originally adopted for Oyster Creek as part of an amendment which increased the spent fuel pool storage capacity from 840 to 1800 fuel assemblies by replacing existing fuel storage racks with new closer-spaced storage racks. Fornicola Aff., ¶ 6 and Exh. 3 at cover letter and page 2 of enclosure 3 (safety evaluation).

The initial amendment request for the spent fuel pool expansion clearly differentiated the area of the spent fuel storage racks as an area separate and distinct from the CDPS. Fornicola Aff., ¶ 7 and Exh. 4 (Jersey Central Power & Light Company, Request for Amendment to Provisional Operating License No. DPR-16 -- Technical Specification Change Request No. 44 and Facility Description and Safety Analysis Report Amendment No. 78 (Mar. 18, 1976)) at pages 10.0-16, 10.0-17 of FDSAR amendment. This clear differentiation between the spent fuel storage area and the CDPS is continued throughout subsequent correspondence between the NRC staff and the Licensee on the amendment request. Although Technical Specification 5.3.1.B was not part of the initial application, (see id. at pages 6-7 of the introductory material), in the process

[11] Where the text of a requirement is considered ambiguous, the documents related to it should be considered to determine its meaning. See Wisconsin Pub. Intervenor v. Mortier, 501 U.S. 597, 610 n.4 (1991) (applying the canon to use legislative history in interpretation of a statute).

of reviewing the amendment request the NRC staff asked questions about the structural integrity of the new spent fuel storage racks, which eventually resulted in Technical Specification 5.3.1.B. Fornicola Aff., ¶ 10 and Exh. 5 (U.S. Nuclear Regulatory Commission, Request for Additional Information, Oyster Creek Nuclear Generating Station Spent Fuel Pool - Increased Storage Capacity (June 24, 1976)) at pages 1-6.

Thus, question 39 from the NRC staff asked the Licensee to:

> Provide (1) the number of bundles that could be struck by a cask fall or tip, including effects of any superstructure on the cask; (2) a conservative analysis of fission product release from fuel bundles potentially subject to impact assuming that the most recently off-loaded fuel is in the impact area; (3) a realistic (best estimate) radiological analysis of a cask fall or tip; and (4) any technical specifications proposed on the decay time required prior to loading storage positions within the zone which could be struck by a cask fall or tip.

Id. at 5. Licensee's response to Question 39 was that "[a] cask drop accident on or near stored fuel assemblies is not anticipated since the Oyster Creek spent fuel pool is equipped with a cask drop protection system (CDPS)," and the cask "will not be moved over the fuel storage area at any time." Fornicola Aff., ¶ 11 and Exh. 6 (Jersey Central Power & Light Company, Supplement No. 1 to Facility Description and Safety Analysis Report Amendment No. 78 (Aug. 11, 1976)) at page 39-1 of the FDSAR amendment. This again clearly differentiates the CDPS from the location of "stored fuel assemblies" and the "fuel storage area."

Question 40 from the NRC staff asked the Licensee to:

> Discuss the overhead cask handling system from the points of view of (1) yoke and/or cable failure, and (2) braking devices, their capacity and effect on the ability of the handling system to withstand possible sudden decelerations induced by rapid braking following a loss of power to the system. Discuss all typical loads that may be carried near or over the spent fuel pool.

Fornicola Aff., ¶ 12 and Exh. 5 at pages 5-6. Licensee responded to Question 40 that "[s]ince the cask will not be moved over the fuel storage area, a yoke and/or cable failure is not expected to have any effect on stored assemblies." Fornicola Aff., ¶ 12 and Exh. 6 at page 40-1 of the FDSAR amendment. This too differentiates the CDPS and the safe load path followed by the cask from "fuel storage area" and "stored assemblies." It is important to note, as Licensee has previously identified, that the shield plug will follow the same safe load path as the cask. Fornicola Aff., ¶ 3 and Exh. 2 at unnumbered page 4. Licensee's response to Question 40 also added that "[d]uring normal operation loads over the spent fuel pool will be limited to spent fuel assemblies, weighing approximately 700 lbs." Fornicola Aff., ¶ 13 and Exh. 6 at page 40-1 of the FDSAR amendment. As a follow-up, the NRC staff requested the Licensee to "[p]ropose Technical Specifications to limit the weight of loads moved over stored fuel in the spent fuel pool." Fornicola Aff., ¶ 14 and Exh. 7 (Jersey Central Power & Light Company, Revision No. 1 to Technical Specification Change Request No. 44 and Addendum No. 1 to Supplement No. 1 Amendment No. 78 of the Facility Description and Safety Analysis Report (Nov. 30, 1976)) at page 40-2 of the FDSAR amendment. Licensee responded with "a proposed Technical Specification Change to limit the maximum weight of loads moved over the stored fuel in the spent fuel

pool. . . ." Id. The proposed technical specification was labeled 5.3.1.D,[12] but is otherwise identical to what became Technical Specification 5.3.1.B and read:

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

Fornicola Aff., ¶ 14 and Exh. 7 at 5.3.1.D. The NRC staff request, Licensee response, and new technical specification all include storage-related terms including "stored fuel," "stored irradiated fuel," and "fuel storage facility." This usage is consistent with the terms discussed earlier, including "fuel storage area" and "stored assemblies," that are used throughout the documentation of the amendment request.

The NRC issued the amendment to replace the existing fuel storage racks with higher capacity spent fuel storage racks on March 30, 1977, with the technical specification on loads over stored fuel adopted as proposed. Fornicola Aff., ¶ 15 and Exh. 3 at unnumbered page 1. This technical specification, now labeled 5.3.1.B, was adopted over three years prior to the initial publication of NUREG-0612.[13]

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[12] Four new technical specifications were added as a result of this amendment, 5.3.1.B - 5.3.1.E. Fornicola Aff., ¶ 14 and Exh. 7 at unnumbered page 7. Three of the four were subsequently moved to another section of the technical specifications, leaving the subject technical specification to be renumbered 5.3.1.B.

[13] Fornicola Aff., ¶ 6 and Exh. 3 at unnumbered page 6 of introductory materials. The letter granted Amendment 22 and established the contents of Technical Specification 5.3.1.B. The technical specification was adopted as part of Amendment 22 in March 1977; NUREG-0612 was not published until July 1980. Table 3.2-1 of NUREG-0612 mistakenly indicated that Oyster Creek did not have a such a technical specification (one comparable to standard technical specification 3.9.6.2). This oversight regarding the Oyster Creek technical specification may have resulted from the fact that the Technical Specification 5.3.1.B is in Section 5, and not in Section 3, of the Oyster Creek technical specifications.

In issuing the amendment to change the spent fuel storage racks, the NRC specifically stated that the amendment will:

> continue to accommodate one fuel assembly shipping cask for off-site shipping of spent fuel assemblies from the Oyster Creek spent fuel pool when offsite spent fuel shipment is resumed at some indefinite future date. . . .

Fornicola Aff., ¶ 17 and Exh. 3 at unnumbered page 1 of introductory materials. This NRC statement shows that both the staff and the Licensee understood that adopting Technical Specification 5.3.1.B, which is part of the subject amendment, does not prohibit future "shipping of spent fuel assemblies from the Oyster Creek spent fuel pool." Id. This again indicates that the packaging area for loading shipping casks is separate and distinct from the spent fuel storage area. This is consistent with the questions and responses related to Amendment 22, discussed supra, that differentiate the CDPS from the location of "stored fuel assemblies" and the "fuel storage area."

Technical Specification 5.3.1.B thus has always allowed spent fuel to be packaged in the CDPS for transport out of the facility. The NRC staff and Licensee documentation related to adopting Technical Specification 5.3.1.B show it applies specifically to "stored fuel" in the "fuel storage area" and it does not apply to the CDPS, where the shield plug would be handled.

3. The Standard Technical Specification Recommended by NUREG-0612 is Specifically for Stored Spent Fuel in Storage Racks and Does Not Mean Spent Fuel Being Packaged for Transport

While Technical Specification 5.3.1.B was not issued in response to NUREG-0612, even if the technical specification were interpreted in a manner consistent with the recommendations

of NUREG-0612, its meaning would not change. The technical specification recommended by NUREG-0612 is also limited to spent fuel stored in racks.

The interim protection measure specifically recommended by NUREG-0612 was implementation of "a specification comparable to . . . Standard Technical Specification 3.9.6.2, 'Crane Travel,' for BWR's." NUREG-0612 at 5-18.[14] This Standard Technical Specification states that:

> Loads in excess of (2500) pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks. (emphasis added)

Fornicola Aff., ¶ 19 and Exh. 8 (U.S. Nuclear Regulatory Commission, NUREG-0123, Standard Technical Specifications for General Electric Boiling Water Reactors (Rev. 2 1979)) at page 3/4 9-9 (emphasis added). The "Applicability" of this Standard Technical Specification is specifically limited to areas of the spent fuel pool "with fuel assemblies in the spent fuel storage pool racks." Id. (emphasis added). Spent fuel storage pool racks are a set of specific structural elements arranged in the spent fuel pool to hold spent fuel until such time as it is transferred to the packaging area for transport out of the facility. The storage racks are different from, and independent of the CDPS that is used for packaging spent fuel in a transfer cask. Fornicola Aff., ¶ 8. The Standard Technical Specification used as the model for interim protection recommendation (1) of NUREG-0612 thus applies only to heavy loads over spent fuel in storage racks, and not to spent fuel being packaged for transport in the CDPS.

[14] This technical specification is now Standard Technical Specification 3.9.7. See Fornicola Aff., ¶ 19 and Exh. 8 at page 3/4 9-9.

While this interim recommendation was not adopted in the initial December 22, 1980 Generic Letter, subsequent reviews considered by the NRC staff considered the prior recommendation and determined that Oyster Creek's Technical Specification 5.3.1.B (as it existed prior to the recent amendment) is comparable to the standard technical specification.[15] The comparability of Technical Specification 5.3.1.B again shows that the provision allows spent fuel to be packaged in the CDPS for transport out of the facility. The Standard Technical Specification that forms the basis for the recommendation of NUREG-0612 applies only to heavy loads over spent fuel pool storage racks and not to spent fuel in the CDPS.

4. Interpreting Technical Specification 5.3.1.B to Prohibit Loading Casks in the CDPS Leads to the Absurd Result that Spent Fuel Can Never Be Removed from the Facility

The text of a requirement should not be interpreted to reach an absurd result. See United States v. Wilson, 503 U.S. 329, 112 S. Ct. 1351, 1354 (1992) (applying the canon to interpretation of a statute). Petitioners' Basis C, as summarized by the Board, asserts that the Technical Specification is a prohibition on the movement of heavy loads that "cannot be revised as the licensee has requested." LBP-96-23, supra note 1, at 12. If the technical specification could not have been revised as requested, and has the meaning petitioners attribute to it, the Licensee would be precluded from using a shield plug over irradiated fuel in the transfer cask.

[15] Franklin Research Center, Technical Evaluation Report - Control of Heavy Loads (June 10, 1983), attached to Exh. C, at 22-23. See also U.S. Nuclear Regulatory Commission, NUREG-1382, Safety Evaluation Report related to the Full-Term Operating License for Oyster Creek Nuclear Generating Station, attached as Exhibit E hereto, at 9-4 (1991).

Without shielding, the spent fuel could never be removed from the facility.[16] Interpreting the text to require spent fuel to remain in the reactor building indefinitely is an absurd result that should not be attributed to Technical Specification 5.3.1.B.

The "to the extent practical" language of NUREG-0612 avoids this absurd result. NUREG-0612 included this proviso in its summarized defense-in-depth approach as follows:

> (2) Define safe load travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment.

NUREG-0612 at 5-2. The NRC Generic Letter 85-11 also includes this proviso and illustrates its application. Exh. D, Encl. 1 at 1, 4. Closing out "Phase II" of the heavy loads program, this Generic Letter states:

> [M]ost of the risk appears to be associated with carrying heavy loads over or in a location where spent fuel could be damaged. The single most important example of this concerns loads handled over the open reactor vessel during refueling (such as the reactor vessel head). However, as previously mentioned, this is limited to the extent practical and where necessary, is performed with a specifically implemented program in conformance with the Phase I guidelines.

[16] Many regulations directly require the use of shielding for spent fuel moved out of the spent fuel pool, including 10 C.F.R. §§ 71.47 (external radiation standards for transport packages); 72.104 (criteria for direct radiation from Independent Spent Fuel Storage Installation ("ISFSI") operations); 72.106 (dose limits for individuals at boundary of the ISFSI controlled area); 72.126 (design requirements for ISFSI radiological protection); 72.128 (shielding requirements for spent fuel storage systems); and 72.236 (shielding requirements for spent fuel storage cask approval). 10 C.F.R. §§ 71.47, 72.104, 72.106, 72.126, 72.128, 72.236 (1996). Without the shield plug in place, the transfer cask would violate these requirements.

If the requirements for shielding were not applied, Petitioners' interpretation would lead to an equally absurd position that the shield plug could be raised over spent fuel in the dry shielded canister (DSC) once it has been removed from the spent fuel pool, but not raised over the DSC while it is in the spent fuel pool. Thus, the only effect of Petitioners' interpretation would be to prohibit a licensee from taking advantage of the shielding effect of the water in the spent fuel pool.

Id. at 4. Handling of the reactor vessel head over spent fuel in the open reactor vessel must be done to remove the head and to reinstall the head. There is no alternative. Similarly, handling of the shield plug over spent fuel in the transfer cask must be done to install shielding in the cask. Again, there is no alternative. Just as the reactor vessel head movement is enabled under the "to the extent practical" proviso, the shield plug movement must also be enabled.

The meaning of Technical Specification 5.3.1.B should not be interpreted to reach the absurd result that the spent fuel can never be removed from the facility.

V. CONCLUSION

For all of the above stated reasons, the Board should find that there is no legal restriction to changing Technical Specification 5.3.1.B as the Licensee has requested, and thus the Petitioners' contention should be dismissed.

Respectfully submitted,

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Counsel for Licensee

Dated: November 15, 1996

November 15, 1996

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-219-OLA
GPU NUCLEAR CORPORATION	)	(Tech. Spec. 5.3.1.B)
	)	
(Oyster Creek Nuclear Generating Station)	)	ASLBP No. 96-717-02-OLA

LICENSEE'S MOTION FOR SUMMARY DISPOSITION

EXHIBITS A-E

377958-01 / DOCSDC1

Exhibit A

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

November 7, 1996

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporaticn
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,

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

M. Roche
GPU Nuclear Corporation

Oyster Creek Nuclear
Generating Station

cc:

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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

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

Exhibit B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

December 22, 1980

TO ALL LICENSEES OF OPERATING PLANTS AND
APPLICANTS FOR OPERATING LICENSES AND
HOLDERS OF CONSTRUCTION PERMITS*

Gentlemen:

Subject: Control of Heavy Loads

In January 1978, the NRC published NUREG-0410 entitled, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants - Report to Congress." As part of this program, the Task Action Plan for Unresolved Safety Issue Task No. A-36, "Control of Heavy Loads Near Spent Fuel," was issued.

We have completed our review of load handling operations at nuclear power plants. A report describing the results of this review has been issued as NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of TAP A-36." This report contains several recommendations to be implemented by all licensees and applicants to ensure the safe handling of heavy loads.

The purpose of this letter is to request that you review your controls for the handling of heavy loads to determine the extent to which the guidelines of Enclosure 1 are presently satisfied at your facility, and to identify the changes and modifications that would be required in order to fully satisfy these guidelines.

To expedite your compliance with this request, we have enclosed the following:

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

Staff Position - Interim Actions for Control of Heavy Loads (Enclosure 2).

Request for Additional Information on Control of Heavy Loads (Enclosure 3).

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*With the exception of licensees for Indian Point 2 and 3, Zion 1 and 2 and Three Mile Island 1 (These were previously sent a letter)

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You are requested to implement the interim actions described in Enclosure 2 as soon as possible but no later than 90 days from the date of this letter.

In order to enable the NRC to determine whether operating licenses should be modified (10 CFR 50.54(f)), operating reactor licensees are requested to provide the following:

1. Submit a report documenting the results of your review and the required changes and modifications. This report should include the information identified in Sections 2.1 through 2.4 of Enclosure 3, on how the guidelines of NUREG-0612 will be satisfied. This report should be submitted in two parts according to the following schedule:

   - Submit the Section 2.1 information within six months from the date of this letter.

   - Submit the Sections 2.2, 2.3 and 2.4 information within nine months.

2. Furnish confirmation within six months that implementation of those changes and modifications you find are necessary will commence as soon as possible without waiting for staff review, so that all such changes, beyond the above interim actions, will be completed within two years of submittal of Section 2.4 for the above report.

3. Furnish justification within six months for any changes or modifications that would be required to fully satisfy the guidelines of Enclosure 1 which you believe are not necessary.

The criteria in NUREG-0612 are also applicable to applicants for operating licenses. Such applicants are expected to provide the information requested by item 1 above and to meet the same schedule of implementation as indicated in 2 above. Any item for which the implementation date is prior to the expected date of issuance of an operating license will be considered to be a prerequisite to obtaining that license.

For any date that cannot be met, furnish a proposed revised date, justification for the delay, and any planned compensating safety actions during the interim.

This request for information was approved by GAO under a blanket clearance number R0072 which expires November 30, 1983. Comments on burden and duplication may be directed to the U.S. General Accounting Office, Regulatory Reports Review, Room 5106, 441 G Street, N.W., Washington, D.C. 20548.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. NUREG-0612
2. Staff Position
3. Request for Additional Information

cc: w/o Enclosure (1)
Service List

STAFF POSITION -
INTERIM ACTIONS FOR
CONTROL OF HEAVY LOADS

(1) Safe load paths should be defined per the guidelines of Section 5.1.1(1) (See Enclosure 1);

(2) Procedures should be developed and implemented per the guidelines of Section 5.1.1(2) (See Enclosure 1);

(3) Crane operators should be trained, qualified and conduct themselves per the guidelines of Section 5.1.1(3) (See Enclosure 1);

(4) Cranes should be inspected, tested, and maintained in accordance with the guidelines of Section 5.1.1(6) (See Enclosure 1); and

(5) In addition to the above, special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures.

REQUEST FOR ADDITIONAL INFORMATION ON CONTROL OF HEAVY LOADS

1. INTRODUCTION

Verification by the licensee that the risk associated with load-handling failures at nuclear power plants is extremely low will require a systematic evaluation of all load-handling systems at each site. The following specific information requests have been organized to support such a systematic approach, and provide a basis for the staff's review of the licensee's evaluation. Additionally, they have been organized to address separately the two hazards requiring investigation (i.e., radiological consequences of damage to fuel and unavailability consequences of damage to certain systems). The following general information is provided to assist in this evaluation and reduce the need for clarification as to the intent and expected results of this inquiry.

1. Risk reduction can be demonstrated by either of two approaches:

   a. The likelihood of failure is made extremely low through enhanced handling-system design features (NUREG 0612, Section 5.1.6).

   b. The consequences of a failure can be shown to be acceptable (NUREG 0612, Section 5.1, Criteria I-IV).

   Regardless of the approach selected, the general guidelines of NUREG 0612, Section 5.1.1, should be satisfied to provide maximum practical defense-in-depth.

2. Evaluations concerning radiological consequences or criticality safety, where used, can rely on either the adoption of generic analyses reported in NUREG 0612, requiring only verfication that these generic assumptions are valid for a specific site, or employ a site-specific analysis.

3. Systems required for safe shutdown and continued decay heat removal are site-specific and are not, therefore, identified in this request. Individual plants should consider systems and components identified in Regulatory Guide 1.29, Position C.1 (except those systems or portions of systems that are required solely for (a) emergency core cooling (b) post-accident containment heat removal, or (c) post-accident containment atmosphere cleanup), for evaluation and recognize that the approach taken in this respect is similar to that identified in Regulatory Guide 1.29, Position C.2. The fact that a load-handling system may be prevented from operating during plant conditions requiring the actual or potential use of some of these systems, is rec-

ognized in this request for information.

4. The scope of this systematic review should include all heavy loads carried in areas where the potential for non-compliance with the acceptance criteria (NUREG 0612, Section 5.1) exists. A summary of typical loads to be considered has been provided in NUREG 0612, Table 3.1-1. It is recognized that some cranes will carry additional miscellaneous loads, some of which are not identifiable in detail in advance. In such cases an evaluation or analysis demonstrating the acceptability of the handling of a range of loads should be provided.

5. At some sites loads which must be evaluated will include licensed shipping casks provided for the transportation of irradiated fuel, solidified radioactive waste, spent resins, or other byproduct material. Licensing under 10CFR71 is not evidence that lifting devices for these shipping casks meet the criteria specified in NUREG 0612, Sections 5.1.1(4), 5.1 1(5), 5.1.6(1), or 5.1.6(3), as appropriate, and thus does not eliminate the need to provide appropriate information concerning these devices. A tabulation (Attachment 5) is provided to indicate multiple-site use of these shipping casks.

The results of the licensee's evaluation, as reported in response to this request, should provide information sufficient for the staff to conduct an independent review to determine that the intent of this effort (i.e., the uniform reduction of the potential hazard from load-handling-system failures) has been satisfied.

2. INFORMATION REQUESTED FROM THE LICENSEE

2.1 GENERAL REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS

NUREG 0612, Section 5.1.1, identifies several general guidelines related to the design and operation of overhead load-handling systems in the areas where spent fuel is stored, in the vicinity of the reactor core, and in other areas of the plant where a load drop could result in damage to equipment required for safe shutdown or decay heat removal. Information provided in response to this section should identify the extent of potentially hazardous load-handling operations at a site and the extent of conformance to appropriate load-handling guidance.

1. Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking n credit for any

interlocks, technical specifications, operating procedures, or detailed structural analysis).

2. Justify the exclusion of any overhead handling system from the above category by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal.

3. With respect to the design and operation of heavy-load-handling systems in the reactor building and those load-handling systems identified in 2.1-1, above, provide your evaluation concerning compliance with the guidelines of NUREG 0612, Section 5.1.1. The following specific information should be included in your reply:

   a. Drawings or sketches sufficient to clearly identify the location of safe load paths, spent fuel, and safety-related equipment.

   b. A discussion of measures taken to ensure that load-handling operations remain within safe load paths, including procedures, if any, for deviation from these paths.

   c. A tabulation of heavy loads to be handled by each crane which includes the load identification, load weight, its designated lifting device, and verification that the handling of such load is governed by a written procedure containing, as a minimum, the information identified in NUREG 0612, Section 5.1.1(2).

   d. Verification that lifting devices identified in 2.1.3-c, above, comply with the requirements of ANSI N14.6-1978, or ANSI B30.9-1971 as appropriate. For lifting devices where these standards, as supplemented by NUREG 0612, Section 5.1.1(4) or 5.1.1(5), are not met, describe any proposed alternatives and demonstrate their equivalency in terms of load-handling reliability.

   e. Verification that ANSI B30.2-1976, Chapter 2-2, has been invoked with respect to crane inspection, testing, and maintenance. Where any exception is taken to this standard, sufficient information should be provided to demonstrate the equivalency of proposed alternatives.

   f. Verification that crane design complies with the guidelines of CMAA Specification 70 and Chapter 2-1 of ANSI B30.2-1976, including the demonstration of equivalency of actual design requirements for instances where specific compliance with these standards is not provided.

8. Exceptions, if any, taken to ANSI B30.2-1976 with respect to operator training, qualification, and conduct.

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE REACTOR BUILDING

NUREG 0612, Section 5.1.4, provides guidelines concerning the design and operation of load-handling systems in the vicinity of spent fuel in the reactor vessel or in storage. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that, in this area, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG 0612, Section 5.1, Criteria I through III.

1. Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads over spent fuel in the storage pool or in the reactor vessel.

2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of heavy loads over stored fuel or into any location where, following any failure, such load may drop into the reactor vessel or spent fuel storage pool.

3. Identify any cranes listed in 2.2-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6 or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

4. For cranes identified in 2.2-1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG 0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the Reactor Building and your determination of compliance. This response should include the following information for each crane:

   a. Where reliance is placed on the installation and use

electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the administrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specifications concerning the bypass of such interlocks.

b. Where reliance is placed on the operation of the Standby Gas Treatment System, discuss present and/or proposed technical specifications and administrative or physical controls provided to ensure that these assumptions remain valid.

c. Where reliance is placed on other site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications, and discuss administrative or physical controls provided to ensure the validity of such considerations.

d. Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of NUREG 0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

2.3 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG 0612, Section 5.1.5, provides guidelines concerning the design and operation of load-handling systems in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in these areas, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment from load drops will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

1. Identify any cranes listed in 2.1-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

2. For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.3-1, a comprehensive hazard evaluation should be provided which includes the following information:

   a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1-3-c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

   b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned consideration should be supplemented by the following specific information:

      (1) For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

      (2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

      (3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance seque .ing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the validity of such considerations.

c. For interactions not eliminated by the analysis of 2.3-2-b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.3-2-b or 2.3-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

(1) An indication of whether or not, for the specific load being investigated, the overhead crane-handling system is designed and constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake (SSE).

(2) The basis for any exceptions taken to the analytical guidelines of NUREG 0612, Appendix A.

(3) The information requested in Attachment 4.

NOTES TO FIGURE 1

Note 1: Indicate by symbols the safety-related equipment. The licensee should provide a list consistent with the clarification provided in 1.2-3.

Note 2: <u>Hazard Elimination Categories</u>

a. Crane travel for this area/load combination prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations eliminate the need to consider load/equipment combination.

d. Likelihood of handling system failure for this load is extremely small (i.e. section 5.1.6 NUREG 0612 satisfied).

e. Analysis demonstrates that crane failure and load drop will not damage safety-related equipment.

FIGURE 1

Typical Load/Impact Area Matrix

CRANE: (IDENTIFY THE CRANE BY NAME AND EQUIPMENT NUMBER)

LOCATION	INDICATE THE BUILDING(S) CORRESPONDING TO THE IMPACT AREA(S) EXAMPLE: REACTOR BUILDING, AUXILIARY BUILDING					
IMPACT AREA / LOADS	(IDENTIFY AREA BY CONSTRUCTION ZONES) Example: Column Line F-S, Column Line 89-812					
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
(Heavy Load Identification should include designation and weight) Example Spent Fuel Cask NLI 10/24 (100 tons)	(Indicate the various elevations) Example: Elev. 435'	Note 1	Note 2			

SINGLE-FAILURE-PROOF HANDLING SYSTEMS

1. Provide the name of the manufacturer and the design-rated load (DRL). If the maximum critical load (MCL), as defined in NUREG 0554, is not the same as the DRL, provide this capacity.

2. Provide a detailed evaluation of the overhead handling system with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG 0554 and supplemented by the identified alternatives specified in NUREG 0612, Appendix C. This evaluation must include a point-by-point comparison for each section of NUREG 0554. If the alternatives of NUREG 0612, Appendix C, are used for certain applications in lieu of complying with the recommendation of NUREG 0554, this should be explicitly stated. If an alternative to any of those contained in NUREG 0554 or NUREG 0612, Appendix C, is proposed, details must be provided on the proposed alternative to demonstrate its equivalency.[1/]

3. With respect to the seismic analysis employed to demonstrate that the overhead handling system can retain the load during a seismic event equal to a safe shutdown earthquake, provide a description of the method of analysis, the assumptions used, and the mathematical model evaluated in the analysis. The description of assumptions should include the basis for selection of trolley and load position.

4. Provide an evaluation of the lifting devices for each single-failure-proof handling system with respect to the guidelines of NUREG 0612, Section 5.1.6.

5. Provide an evaluation of the interfacing lift points with respect to the guidelines of NUREG 0612, Section 5.1.6.

1/ If the crane in question has previously been approved by the staff as satisfying NUREG 0554, Reg. Guide 1.104, or Part B to BTP-ASB9-1, please reference the date of the staff's safety evaluation report or approval letter in lieu of providing the information requested by item 2.

ANALYSIS OF RADIOLOGICAL RELEASES

The following information should be provided for an analysis conducted to demonstrate compliance with Criterion I of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

   a. Identify the time after shutdown, the number of fuel assemblies damaged, and the assumed duration of radiological release associated with each accident analyzed.

   b. NUREG 0612, Table 2.1-2, provides the assumptions used to arrive at generic conclusions concerning radiological dose consequences. To rely on the radiological dose analysis of NUREG 0612, the licensee should verify that these assumptions are conservative with regard to the plant/site evaluated. If the assumptions are not conservative for the specific plant, or if a more site-specific analysis is required, the licensee should identify plant-specific assumptions used in place of those tabulated.

   c. Identify and provide the basis (e.g., USNRC Regulatory Guide 1.25) for any assumptions employed in site-specific analyses not identified in NUREG 0612, Table 2.1-2.

   d. Dose calculations based on the termination or mitigation of radiological releases should be supported by information sufficient to demonstrate both that the time delay assumed is conservative and that the system provided to accomplish such termination or mitigation will perform its safety function upon demand (i.e., the system meets the criteria for an Engineered Safety Feature). Specific information so provided should include the following:

      (1) Details concerning the location of accident sensors, parameters monitored and the values of these parameters at which a safety signal will be initiated, system response time (including valve-operation time), and the total time required to automatically shift from normal operation to isolation or filtration following an accident.

      (2) A description of the instrumentation and controls associated with the Engineered Safety Feature which includes information sufficient to demonstrate that the requirements (Section 4) of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," are satisfied.

(3) A description of any Engineered Safety Feature filter system which includes information sufficient to demonstrate compliance with the guidelines of USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

(4) A discussion of any initial conditions (e.g., manual valves locked shut, containment airlocks or equipment hatches shut) necessary to ensure that releases will be terminated or mitigated upon Engineered Safety Feature actuation and the measures employed (i.e., Technical Specification and administrative controls) to ensure that these initial conditions are satisfied and that Engineered Safety Feature systems are operable prior to the load lift.

2. METHOD OF ANALYSIS

Discuss the method of analysis used to demonstrate that post-accident dose will be well within 10CFR100 limits. In presenting methodology used in determining the radiological consequences, the following information should be provided.

a. A description of the mathematical or physical model employed.

b. An identification and summary of any computer program used in this analysis.

c. The consideration of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

3. CONCLUSION

Provide an evaluation comparing the results of the analysis to Criterion I of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident analyzed bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

CRITICALITY ANALYSIS

The following information should be provided for analysis conducted to demonstrate compliance with Criterion II of NUREG 0612, Section 5.1

1. INITIAL CONDITIONS/ASSUMPTIONS

   The conclusions of NUREG 0612, Section 2.2, are based on a particular model fuel assembly. If a licensee uses the results of Section 2.2 rather than performing an independent neutronics analysis, the assumptions should be verified to be compatible with plant-specific design. For any analysis conducted, the following assumptions should be provided as a minimum:

   a. Water/UO_2 volume ratio
   b. The boron concentration for the refueling water and spent-fuel pool
   c. The amount of neutron poison in the fuel
   d. Fuel enrichment
   e. The reactivity insertion value due to crushing of the core
   f. The k_{eff} value allowed by technical specifications for the core during refueling

2. METHOD OF ANALYSIS

   Provide the method of analysis used to demonstrate that accidental dropping of a heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95. The discussion of the method of analysis should include the following information:

   a. Identification of the computer codes employed
   b. A discussion of allowances or compensation for calculation and physical uncertainties

3. CONCLUSION

   Provide an evaluation comparing the results of the analysis to Criterion II of NUREG 0612, Section 5.1. If the postulated heavy-load-drop accident

bounds other postulated heavy-load drops, a list of these bounded heavy loads should be provided.

ANALYSIS OF PLANT STRUCTURES

The following information should be provided for analyses conducted to demonstrate compliance with Criteria III and IV of NUREG 0612, Section 5.1.

1. INITIAL CONDITIONS/ASSUMPTIONS

   Discuss the assumptions used in the analysis, including:

   a. Weight of heavy load
   b. Impact area of load
   c. Drop height
   d. Drop location
   e. Assumptions regarding credit taken in the analysis for the action of impact limiters
   f. Thickness of walls or floor slabs impacted
   g. Assumptions regarding drag forces caused by the environment
   h. Load combinations considered
   i. Material properties of steel and concrete

2. METHOD OF ANALYSIS

   Provide the method of analysis used to demonstrate that sufficient load-carrying capability exists within the wall(s) or floor slab(s). Identify any computer codes employed, and provide a description of their capabilities. If test data was employed, provide it and describe its applicability.

3. CONCLUSION

   Provide an evaluation comparing the results of this analysis with Criteria III and IV of NUREG 0612, Section 5.1. Where safe-shutdown equipment has a ceiling or wall separating it from an overhead handling system, provide an evaluation to demonstrate that postulated load drops do not penetrate the ceiling or cause secondary missiles that could prevent a safe-shutdown system from performing its safety function.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

1 - Fuel (New and Spent)

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
4986	RA-1, 2, 3, J	General Electric Co.		TVA
5450	RCC, 1, 2, 3	Westinghouse Electric		VEP, DLC
5805	Vandenburgh	Chem-Nuclear Systems, Inc.	70,000	APC, CPL, DLP, DPC, FPL, FPC, JCP, NPP, VEP
5901	NFS Model 100	Nuclear Fuel Services	126,200	CPC, PGE
5938	BN F		48,000	PEC
6078	927A1 927C1	Combustion Engineering, Co.	6200 7000	APL
6206	B	Babcock & Wilcox Co.	6940	DPC, FPC
6273	48 (Series)		4500	VEP
6375	PB-1	Chem-Nuclear Systems, Inc.	67,050	APC, BEC, CPL, DPC FPL, FPC, GPC, JCP, MYA, MEC, NNE, NSP, PNY, TVA, VEP
6400	Super Tiger	Westinghouse Electric Co.	45,000	APL, CPC, DLP, DLC, MEC, NPP, SMU, VEP
6698	NFS-4	Nuclear Fuel Services, Inc.	50,000	BGE, BEC, CWE, DLP, DPC, FPL, FPC, JCP, MYA, RGE, SCE, WMP,
9001	IF 300	General Electic Co.	140,000	CPL, CWE
9010	NLI-1/2	NL Industries, Inc.	47,500	BEC, FPL, VYC
9044	GE-1600	General Electric Co.	23,000	APC, BGE, BEC, CPL, CPC, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NNE, NSP, VEP, VYC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5026	BC-48-220	Chem-Nuclear Systems, Inc.	71,000	APC, BEC, CPL, CWE, CYA, DPC, DLC, FPL, FPC, JCP, NPP, VEP, WPS
6058	B3-1	Nuclear Engineering Co.	30,000	APL, CPC, DLP, IEL, MEC, NPP, NSP, PGE, SMU, TEC, VEP
6144	6144	Nuclear Engineering Co.	42,000	APC, APL, CPL, CEC, CPC, DLP, DPC, FPL, FPC, GPC, IEL, JCP, MEC, NPP, NSP, PGE, PNY, RGE, SMU, VEP
6244	6244	Chem-Nuclear Systems, Inc.	46,000	APC, CPL, CWE, DPC, FPL, FPC, GPC, JCP, MEC, NPP, NSP, PEC, VEP, WMP
6272	Poly Panther	Nuclear Engineering Co.	6100	APL, CPC, DLP, MEC NPP, SMU, VEP
6568	LL-60-150	Tennessee Valley Auth.	73,000	
6574	HN 200	Hittman Nuclear and Development Corp.	47,000	APL, BGE, CWE, CEC, DLP, DLC, IME, JCP, MYA, MEC, NPP, PEC, PNY, VYC, YAC
6601	LL-50-100	Chem-Nuclear Systems, Inc.	70,000	APC, BEC, CPL, CYA, CEC, CPC, DLP, DPC, FPL, FPC, JCP, NPP, NNE, PEG, PGE, TVA, VEP
6679	1/2 Super Tiger	Nuclear Engineering Co.	45,000	APL, CPC, DLP, MEC, NPP, SMU, VEP
6722	BS-33-150	Tennessee Valley Auth.	51,000	

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
6744	Poly Tiger	Nuclear Engineering Co.	35,000	APL, BEC, CPC, DLP, MEC, NPP, SMU, VEP
6771	SN-1	Nuclear Engineering Co.	60,000	APL, CPC, DLP, NPP, SMU, VEP
9074	AP-100		28,000	DLC
9079	HN-100 Ser. 2	Hittman Nuclear and Development Corp.	98,000	APL, BGE, CEC, CWE, DLP, IME, JCP, MYA, MEC, NPP, PEC
9080	HN-600	Hittman Nuclear and Development Corp.	42,000	BGE, CWE, CEC, DLP, IME, IEL, JCP, MYA, MEC, NPP, PEC, YAC
9086	HN-100 Ser. 1	Hittman Nuclear and Development Corp.	46,000	APL, BGE, CWE, DLP, IME, JCP, MYA, MEC, NPP, NNE, PEC, RGE, VYC
9089	HN-100S	Hittman Nuclear and Development Corp.	36,300	BGE, CWE, CEC, IME, JCP, MYA, NPP, PEC
9092	HN-300	Hittman Nuclear and Development Corp.	43,000	MYA
9093	HN-400	Hittman Nuclear and Development Corp.	43,000	MYA
9094	CNSI-14-195-H	Chem-Nuclear Systems, Inc.	56,500	APC, APL, BEC, CPL, CWE, CYA, CEC, CPC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, OPP, PGE, PEC, PGC, PNY, PEG, TVA, VEP
9096	CNSI-21-300	Chem-Nuclear Systems, Inc.	57,450	APC, APL, CPL, CEC, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, PNY, PEG, VEP

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

II - Waste

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
9105	RAD-Waste CR.I	Chem-Nuclear Systems, Inc.	58,400	APC, CPL, DPC, FPL, FPC, GPC, JCP, MEC, NMP, VEP
9108	AL-33-90	Chem-Nuclear Systems, Inc.	41,300	APC, CPL, CWE, CEC, DPC, FPL, FPC, JCP, NPP, NMP, NNE, PGC, VEP, WEP
9111	CN6-80A	Chem-Nuclear Systems, Inc.	51,500	APC, CPL, CWE, CEC, DPC, FPL, FPC, GPC, MEC, NNE, PGC, SMC, VEP
9113	7-100	Chem-Nuclear Systems, Inc.	7000	APC, BEC, CPL, CWE, CYA, DPC, FPL, FPC, GPC, JCP, MEC, NMP, NNE, NSP, VEP
9122	18-450	Chem-Nuclear Systems, Inc.	61,000	BEC

*See attached list of abbreviations.

SHIELDED SHIPPING CASKS CERTIFICATED FOR NUCLEAR POWER PLANTS

III - Byproducts

CERT.	MODEL	PRIMARY LICENSEE	GROSS LOT IN LBS. (APPROX.)	SECONDARY LICENSEE*
5971	GE-200		10,000	PEC
5980	GE-600		18,500	NNE, NSP
6275	LL-28-4	Chem-Nuclear Systems, Inc.	30,000	APC, CPL, DPC, FPL, FPC, NPP, VEP
9081	CNS-1600	Chem-Nuclear Systems, Inc.	26,000	APC, BGE, CPL, DPC, FPL, FPC, GPC, NSP, TVA, VEP

*See attached list of abbreviations.

LICENSEE ABBREVIATIONS

APC	Alabama Power Company
APL	Arkansas Power and Light Company
BEC	Boston Edison Company
BGE	Baltimore Gas and Electric Company
CEC	Consolidated Edison Company
CPC	Consumers Power Company
CPL	Carolina Power and Light Company
CWE	Commonwealth Edison Company
CYA	Connecticut Yankee Atomic Power Company
DLC	Duquesne Light Company
DLP	Dairyland Power Cooperative
DPC	Duke Power Company
FPC	Florida Power Corporation
FPL	Florida Power and Light Company
GPC	Georgia Power Company
IEL	Iowa Electric Light and Power Company
IME	Indiana and Michigan Electric Company
JCP	Jersey Central Power and Light Company
MEC	Metropolitan Edison Company
MYA	Maine Yankee Atomic Power Company
NMP	Niagara Mohawk Power Corporation
NNE	Northeast Nuclear Energy Company
NPP	Nebraska Public Power Corporation
NSP	Northern States Power Company
OPP	Omaha Public Power District
PEC	Philadelphia Electric Company
PEG	Public Service Electric and Gas Company
PGC	Portland General Electric Company
PNY	Power Authority of the State of New York
RGE	Rochester Gas and Electric Corporation
SMU	Sacramento Municipal Utilities Corporation
TEC	Toledo Edison Company
TVA	Tennessee Valley Authority
VEP	Virginia Electric and Power Company
VYC	Vermont Yankee Nuclear Power Corporation
YAC	Yankee Atomic Electric Company
WMP	Wisconsin-Michigan Power Company
WPS	Wisconsin Public Service Corporation

BOILING WATER REACTOR LICENSEES

Docket No. 50-293
Pilgrim Unit 1

Docket No. 50-325
Brunswick Unit 1

Docket No. 50-324
Brunswick Unit 2

Docket No. 50-10
Dresden 1

Docket No. 50-237
Dresden 2

Docket No. 50-249
Dresden 3

Docket No. 50-254
Quad-Cities Unit 1

Docket No. 50-265
Quad-Cities Unit 2

Docket No. 50-155
Big Rock Point

Docket No. 50-409
Lacrosse

Docket No. 50-321
Edwin I. Hatch Unit 1

Docket No. 50-366
Edwin I. Hatch Unit 2

Docket No. 50-331
Duane Arnold

Docket No. 50-219
Oyster Creek

Docket No. 50-220
Nine Mile Point Unit 1

Docket No. 50-298
Cooper Station

Docket No. 50-245
Millstone Unit 1

Docket No. 50-263
Monticello

Docket No. 50-133
Humboldt Bay

Docket No. 50-277
Peach Bottom Unit 2

Docket No. 50-278
Peach Bottom Unit 3

Docket No. 50-333
FitzPatrick

Docket No. 50-259
Browns Ferry Unit 1

Docket No. 50-260
Browns Ferry Unit 2

Docket No. 50-296
Browns Ferry Unit 3

Docket No. 50-271
Vermont Yankee

BWRs

LB-1

- La Salle 1+2 50-373+50-374
- Nine Mile 2 - 50-410
- Allens Creek 1 - 50-466
- Susquehanna 1+2 - 50-387+50-388
- Shoreham - 50-322
- Bailly - 50-367
- WNP-2

LB-2

- Perry
- Hartsville 1,2,3+4 50-518, 50-520, 52
- Fermi - 2 - 50-341
- Zimmer - 1 - 50-358
- Limerick 1+2 50-352+50-353

LB-3

- Clinton 1+2 50-461+50-462
- River Bend 1+2 50-458+50-459
- Grand Gulf 1+2 50-416+50-417
- Hope Creek 1+2 50-354+50-355
- Pilgrim 2 50-471

POOR ORIGINAL

PRESSURIZED WATER REACTOR LICENSEES

Docket No. 50-348
Farley Unit 1

Docket No. 50-313
Arkansas Unit 1

Docket No. 50-368
Arkansas Unit 2

Docket No. 50-317
Calvert Cliffs Unit 1

Docket No. 50-318
Calvert Cliffs Unit 2

Docket No. 50-261
H. B. Robinson Unit 2

Docket No. 50-295
Zion Unit 1

Docket No. 50-304
Zion Unit 2

Docket No. 50-213
Connecticut Yankee (Haddam Neck)

Docket No. 50-3
Indian Point Unit 1

Docket No. 50-247
Indian Point Unit 2

Docket No. 50-~~296~~ 286
Indian Point Unit 3

Docket No. 50-255
Palisades

Docket No. 50-269
Oconee Unit 1

Docket No. 50-270
Oconee Unit 2

Docket No. 50-287
Oconee Unit 3

Docket No. 50-334
Beaver Valley Unit 1

Docket No. 50-302
Crystal River 3

Docket No. 50-335
St. Lucie 1

Docket No. 50-250
Turkey Point Unit 3

Docket No. 50-251
Turkey Point Unit 4

Docket No. 50-315
D. C. Cook Unit 1

Docket No. 50-316
D. C. Cook Unit 2

Docket No. 50-309
Maine Yankee

Docket No. 50-289
Three Mile Island Unit 1

Docket No. 50-320
Three Mile Island Unit 2

Docket No. 50-336
Millstone Unit 2

Docket No. 50-282
Prairie Island Unit 1

Docket No. 50-306
Prairie Island Unit 2

Docket No. 50-285
Ft. Calhoun

Docket No. 50-344
Trojan

Docket No. 50-272
Salem Unit 1

Docket No. 50-244
R. E. Ginna 1

Docket No. 50-312
Rancho Seco

Docket No. 50-206
San Onofre Unit 1

Docket No. 50-346
Davis-Besse 1

Docket No. 50-338
North Anna 1

Docket No. 50-280
Surry Unit 1

Docket No. 50-281
Surry Unit 2

Docket No. 50-266
Point Beach Unit 1

Docket No. 50-301
Point Beach Unit 2

Docket No. 50-305
Kewaunee

Docket No. 50-29
Yankee-Rowe

Docket No. 50-339
North Anna 2

Docket No. 50-311
Salem 2

~~PLANTS UNDER CONSTRUCTION~~

1.	Palo Verde 1/2/3	50-528, 50-529, 50-530
~~2.~~	~~Perry 1/2~~	~~50-440, 50-441~~
3.	Cherokee 1/2/3	50-491, 50-492, 50-493
4.	Beaver Valley 2	50-412
5.	St. Lucie 2	50-389
6.	Vogtle 1/2	50-424, 50-425
~~7.~~	~~River Bend 1/2~~	~~50-458, 50-459~~
~~8.~~	~~Clinton 1/2~~	~~50-461, 50-462~~
9.	Forked River	50-363
10.	Wolf Creek 1	50-482
11.	~~[illegible]~~	~~[illegible]~~
~~12.~~	~~Nine Mile Point 2~~	~~50-410~~
13.	Millstone 3	50-423
~~14.~~	~~Bailly 1~~	~~50-367~~
~~15.~~	~~Limerick 1/2~~	~~50-352, 50-353~~
~~16.~~	~~Hope Creek 1/2~~	~~50-354, 50-355~~
17.	Marble Hill 1/2	50-546, 50-547
18.	Seabrook 1/2	50-443, 50-444
19.	Sterling 1	50-485
~~20.~~	~~Hartsville 1/2/3/4~~	~~50-518, 50-519, 50-520, 50-521~~
21.	Phipps Bend 1/2	50-553, 50-554
22.	Yellow Creek 1/2	50-566, 50-567
23.	~~[illegible]~~	~~[illegible]~~
24.	WPPSS 1/3/4/5	50-460, 50-508, 50-513, 50-509
25.	Callaway 1/2	50-483, 50-486
26.	Harris 1/2/3/4	50-400, 50-401, 50-402, 50-403
27.	Catawba 1/2	50-413, 50-414

ccr

~~PLANTS UNDER OL REVIEW~~

1. ~~[illegible]~~	50~~-[illegible]~~
2. Byron 1/2	50-454, 455
3. Braidwood 1/2	50-456/457
~~4. LaSalle 1/2~~	~~50-373, 374~~
5. Midland 1/2	50-329,330
6. McGuire 1/2	50-369, 370
7. So. Texas 1/2	50-498, 499
8. Shoreham	50-322
9. Waterford	50-382
~~10. Grand Gulf 1/2~~	~~50-416/417~~
11. Diablo Canyon 1/2	50-275, 323
~~12. Susquehana 1/2~~	~~50-387, 388~~
13. ~~[illegible]~~	5~~[illegible]~~
14. Summer 1	50-395
15. San Onofre 2/3	50-361, 362
16. Bellefonte 1/2	50-438, 439
17. Watts Bar 1/2	50-390, 391
18. ~~Sequoyah 1/2~~ Sequoyah	~~50-[illegible], 328~~ 50-328
19. Comanche Peak 1/2	50-445, 446
20. ~~[illegible]~~ 2	5~~[illegible]~~
21. WPPSS-2	50-397
~~22. Fermi 2~~	~~50-341~~
~~23. Zimmer 1~~	~~50-358~~

Exhibit C

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

June 21, 1983

Docket No. 50-219
LS05-83-06-045

Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: CONTROL OF HEAVY LOADS (PHASE 1) - NUREG-0612 - OYSTER CREEK NUCLEAR GENERATING STATION

Enclosed is a copy of our Safety Evaluation (SE) for Oyster Creek, which was developed based on your response to the December 22, 1980 generic letter on "Control of Heavy Loads." This SE was prepared after receiving the Technical Evaluation Report (TER) prepared by the Franklin Research Center (FRC). We concur with the findings contained in the TER and conclude that the guidelines in NUREG-0612, Sections 5.1.1 and 5.3 have been satisfied and, therefore, conclude that Phase 1 for Oyster Creek is acceptable. The TER is attached to our SE. Phase II of NUREG-0612 will be the subject of future correspondence.

The issuance of this letter and the enclosed SE completes our action on Phase 1 of this ~~item~~.

Sincerely,

Walter A. Paulson
for Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
Safety Evaluation and the
Technical Evaluation Report
attached thereto

cc w/enclosures:
See next page

RECEIVED
JUN 24 1983

cc
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Mayor
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Region II Office
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New York, New York 10007

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

CONTROL OF HEAVY LOADS - PHASE 1

1.0 INTRODUCTION

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Plants" was developed. Following the issuance of NUREG-0612 a generic letter, dated December 22, 1980, was sent to all operating plants, applicants for operating licenses and holder of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses were to be made in two stages. The first response (Phase 1) was to identify the load handling equipment within the scope of NUREG-0612 and to describe the associated general load handling operations such as safe load paths, procedures, operator training, special and general purpose lifting devices, the maintenance, testing and repair of equipment and the handling equipment specifications. The second response (Phase 2) was intended to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment had been provided. This safety evaluation contains the staff's evaluation of Phase 1. An evaluation of Phase 2 will be the subject of future correspondence.

2.0 EVALUATION AND CONCLUSION

By letter dated December 22, 1980 the General Public Utilities Nuclear Corporation, the licensee for Oyster Creek was requested to review their provisions for handling and control of heavy loads at Oyster Creek to determine the extent to which the guidelines of NUREG-0612 are presently satisfied and to discuss and commit to mutually agreeable changes and modifications that would be required in order to fully satisfy these guidelines. The staff and its consultant, the Franklin Research Center (FRC), have reviewed the General Public Utilities Nuclear Corporation submittals for Oyster Creek. As a result of its review FRC has issued a Technical Evaluation Report (TER). The staff has reviewed the TER and concurs with its findings that the guidelines in NUREG-0612 Sections 5.1.1 and 5.3 have been satisfied. The staff, therefore, concludes that Phase 1 for Oyster Creek is acceptable.

3.0 ACKNOWLEDGEMENT

The following NRC employee was the principal contributor to this SE:

A. Singh.

Attached: Technical Evaluation Report, dated June 10, 1983.

Date: June 21, 1983

._CHNICAL EVALUATION R_PORT

CONTROL OF HEAVY LOADS (C-10)

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT

NRC DOCKET NO. 50-219

NRC TAC NO. 47128

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 13

FRC TASK 377

Prepared by

Franklin Research Center
20th and Race Streets
Philadelphia, PA 19103

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Author: C. Bomberger, N. Ahmed

FRC Group Leader: I. H. Sargent

Lead NRC Engineer: A. Singh
T. Chan

June 10, 1983

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

TECHNICAL EVALUATION REPORT

CONTROL OF HEAVY LOADS (C-10)

JERSEY CENTRAL POWER AND LIGHT COMPANY
OYSTER CREEK NUCLEAR POWER PLANT

NRC DOCKET NO. 50-219
NRC TAC NO. 47128
NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506
FRC ASSIGNMENT 13
FRC TASK 377

Prepared by
Franklin Research Center
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Author: C. Bomberger, N. Ahmed
FRC Group Leader: I. H. Sargent

Prepared for
Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: A. Singh
T. Chan

June 10, 1983

Prepared by:	Reviewed by:	Approved by:
C. Bomberger	[illegible]	A. P. Carfagno
Principal Author	Group Leader	Department Director
Date: 6/10/83	Date: 6/9/83	Date: 6-10-83

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A Division of The Franklin Institute
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CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. C. R. Bomberger and Mr. I. H. Sargent contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This technical evaluation report documents an independent review of general load handling policy and procedures at the Jersey Central Power & Light Company (JCP&L)/General Public Utilities' (GPU) Oyster Creek Nuclear Power Plant. This evaluation was performed with the following objectives:

- o to assess conformance to the general load handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1
- o to assess conformance to the interim protection measures of NUREG-0612, Section 5.3.

1.2 GENERIC BACKGROUND

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to ensure the safe handling of heavy loads and to recommend necessary changes in these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2] to all power reactor licensees, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-part objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Section 5.1.1, is to ensure that all load handling systems at

nuclear power plants are designed and operated so that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Sections 5.1.2 through 5.1.5, is to ensure that, for load handling systems in areas where their failure might result in significant consequences, either (1) features are provided, in addition to those required for all load handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (2) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

A defense-in-depth approach was used to develop the staff guidelines to ensure that all load handling systems are designed and operated so that their probability of failure is appropriately small. The intent of this guideline is to ensure that licensees of all nuclear power plants perform the following:

- o define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment
- o provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to ensure reliable operation of the handling system.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612. Section 6 of NUREG-0612 recommended that a program be initiated to ensure that these guidelines are implemented at operating plants.

1.3 PLANT-SPECIFIC BACKGROUND

On December 22, 1980, the NRC issued a letter [3] to JCP&L, the Licensee for the Oyster Creek plant, requesting that the Licensee review provisions for handling and control of heavy loads at the Oyster Creek plant, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On September 22, 1981, JCP&L provided the

nitial response [4] to this request. A draft technical evaluation report (TER) was prepared based on this information and was informally transmitted to the Licensee for review and comment. On July 9, 1982, a telephone conference call was conducted with the representatives of NRC, FRC, and JCP&L to discuss unresolved issues. As a result of this call, additional information was provided by the Licensee on February 18, 1983 [5] and on May 27, 1983 [6] which has been incorporated into this final technical evaluation.

2. EVALUATION

This section presents a point-by-point evaluation of load handling provisions at Oyster Creek Nuclear Power Plant with respect to NRC staff guidelines provided in NUREG-0612. Separate subsections are provided for both the general guidelines of NUREG-0612, Section 5.1.1 and the interim measures of NUREG-0612, Section 5.3. In each case, the guideline or interim measure is presented, Licensee-provided information is summarized and evaluated, and a conclusion as to the extent of compliance, including recommended additional action where appropriate, is presented. These conclusions are summarized in Table 2.1.

2.1 GENERAL GUIDELINES

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- o Guideline 1 - Safe Load Paths
- o Guideline 2 - Load Handling Procedures
- o Guideline 3 - Crane Operator Training
- o Guideline 4 - Special Lifting Devices
- o Guideline 5 - Lifting Devices (Not Specially Designed)
- o Guideline 6 - Cranes (Inspection, Testing, and Maintenance)
- o Guideline 7 - Crane Design.

These seven guidelines should be satisfied for all overhead handling systems that handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent fuel pool, or in other areas where a load drop may damage safe shutdown systems. The Licensee's verification of the extent to which these guidelines have been satisfied and the evaluation of this verification are contained in Sections 2.1.1 through 2.1.8 of this report.

Table 2.1. Oyster Creek Nuclear Station/NUREG-0612 Compliance Matrix

Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane - Test and Inspection	Guideline 7 Crane Design	Interim Measure 1 Technical Specifications	Interim Measure 6 Special Attention
1. Reactor Building Crane	100/5	--	--	C	--	--	R	C	--	--
a. Drywell Head	62	C	C	--	R	--	--	--	--	--
b. Reactor Vessel Head	92	C	C	--	R	--	--	--	--	C
c. Cavity Shield Plugs (8)	85 ea.	C	C	--	R	--	--	--	C	--
d. Reactor Vessel Head Insulation	5	C	C	--	--	C	--	--	C	--
e. Steam Dryer	26	C	C	--	C	--	--	--	--	C
f. Steam Separator	44	C	C	--	C	--	--	--	--	C
g. Fuel Pool Gates (2)	Approx. 1	C	C	--	--	C	--	--	C	--
h. Spent Fuel Cask	30/5	C	C	--	R	--	--	--	C	--

C = Licensee action complies with NUREG-0612 Guideline.
P = Licensee information indicates partial compliance.
R = Licensee has proposed revisions/modifications designed to comply with NUREG-0612 Guideline.
-- = Not applicable.

Table 2.1 (Cont.)

Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane - Test and Inspection	Guideline 7 Crane Design	Interim Measure 1 Technical Specifications	Interim Measure 6 Special Attention
i. Fuel Transfer Shield	16.5	C	C	--	R	C	--	--	C	--
j. Equipment Storage Pool Shield Plugs (4)	37.5-39	C	C	--	R	--	--	--	--	--
k. Dryer/Separator Sling Assembly	1.5	C	C	--	--	C	--	--	--	C
l. Fuel Storage Pool Shield Plugs (4)	4.5 ea.	C	C	--	--	C	--	--	C	--
m. Plant Equipment	Less than 20	C	C	--	--	C	--	--	C	--
n. New Fuel and Shipping Containers	1	C	C	--	--	C	--	--	--	--
o. Head Strongback	3.2	C	C	--	--	C	--	--	--	--
p. Stud Tensioner Assembly	10	C	C	--	--	C	--	--	--	--
2. Recirculation Pump Monorail	1	--	--	C	--	--	R	C	--	--
3. Spent Fuel Pool Jib Cranes	0.5	--	--	C	--	--	R	C	C	--

2.1.1 NUREG-0612, Heavy Loads Overhead Handling Systems

a. Summary of Licensee Statements and Conclusions

The Licensee has evaluated the load handling systems at the Oyster Creek plant and concluded that the following load handling systems are subject to NUREG-0612:

- o Reactor building crane
- o Recirculation pump monorail
- o Spent fuel pool jib cranes.

The Licensee has also identified other load handling devices that have been excluded from satisfying the criteria of the general guidelines of NUREG-0612 due to physical separation from safe shutdown equipment or irradiated fuel; these devices include:

- o Machine shop monorail
- o Turbine building crane
- o Equipment handling monorail (outside CRD rebuild room at 75-ft elevation)
- o Filter and demineralizer monorail
- o Equipment handling monorail (adjacent to reactor building equipment hatch at 95-ft elevation)
- o Hatch bay crane
- o CRD rebuild room monorail
- o Railroad bay monorail
- o Jib crane (located 23 ft from reactor building equipment hatch)
- o Maintenance building crane
- o Radwaste building crane.

A second 1-ton jib crane is located adjacent to the reactor building equipment hatchway and has been excluded from NUREG-0612 guidelines due to separation from the torus by the railroad bay floor. The crane is used to lift small equipment, crates, and tools to various elevations in the reactor building. A conservative analysis shows that a heavy load drop by this crane will not result in perforation or scabbing of this floor to damage the equipment located below it.

The intake gantry crane has been excluded from NUREG-0612 applicability due to removal from service. If at some time in the future this crane is placed back into service, an evaluation will be performed to ensure that NUREG-0612 criteria are satisfied.

The three refueling platform auxiliary hoists have been derated from their current rating of 1000 lb to 750 lb so that heavy loads cannot be handled by these load handling systems. This derating would not affect the lifts that they were originally intended to service.

The drywell air lock monorail has been excluded from NUREG-0612 due to the fact that it handles the air lock a few inches off the floor and there is no safe shutdown equipment in close proximity to the airlock. A load drop will not affect safe shutdown capability based on the evaluation of this handling system.

b. Evaluation

The Licensee's conclusions regarding the applicability of general guidelines are is consistent with the intent of NUREG-0612.

c. Conclusion ahd Recommendations

The Oyster Creek plant complies with NUREG-0612 regarding applicability of heavy load overhead handling systems.

2.1.2 Safe Load Paths [Guideline 1, NUREG-0612, Section 5.1.1(1)]

> "Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

a. Summary of Licensee Statements and Conclusions

The Licensee has addressed the handling of heavy loads by defining four safety class designations. Each heavy load is assigned one or more safety classes. The safe load path/procedural requirements corresponding to the assigned safety class have been added to the appropriate plant operating or

maintenance procedures. When more than one safety class assignment is made for a particular load, the safe load path/procedural requirements of all safety class assignments are included in the procedures. Safety class definitions and their respective handling requirements are listed in Table 1, and loads contained in each safety class are listed in Table 2. These safety classes, by procedure, limit lift height and time over areas of concern for the most critical loads (Safety Class 1), define areas over which loads shall not be carried (Safety Class 2), or define safe load paths that follow, to the extent practical, structural floor members, using the minimum practical lift height (Safety Class 3).

All loads designated as Safety Class 3 shall have specified load paths shown on drawings and attached to load handling procedures. In addition, a signalman will be used to ensure that the load is carried along its designated load path. The signalman with the job supervisor will walk down the designated load path prior to load movement to ensure that there are no obstructions that could affect the ability of the crane operator to follow the designated path.

For the reactor building crane load block, shipping casks, fuel channel crates, and new fuel containers, the Licensee stated that the primary concern is the potential for dropping these loads the full length of the equipment hatch located in the southeast quadrant. For these lifts, the crane will be oriented so that the crane hoist is directly over the main structural members for the track bay floor when moving these loads up or down the equipment hatch, in order to assure maximum available resistance to impact in the event of a load drop. In addition, the Licensee added that safe load paths will be defined for movement of shipping casks on the refueling floor prior to their use, including definition of load paths in specific procedures covering movement to and from the equipment hatch, spent fuel pool, and cask washdown area. These load paths will be defined by establishing boundaries around the floor area over which the cask may travel, will be shown on a drawing included in the procedure, and will be marked temporarily using tape on the refueling floor. Within these boundaries, move height will not exceed 6 inches above

Table 1. Load Safety Classes and Safe Load Path Actions

Heavy Handling Situations	Safe Load Path/Procedural Actions Required
Safety Class 1: Load must be carried directly over spent fuel, the reactor vessel, or safe shutdown equipment (i.e., there are no intervening structures such as floors).	Procedurally limit time and height load is carried over the area of concern; define laydown area, show on drawings included in the procedure the prescribed laydown area. Procedures will be reviewed with crane operators and signalmen prior to lifts over an open reactor vessel.
Safety Class 2: Load could be carried directly over spent fuel, the reactor vessel, or safe shutdown equipment, i.e., load can be handled during the time when spent fuel or the reactor vessel is exposed or safe shutdown equipment is required to be operable and there are no physical means (such as interlocks or mechanical stops) available to restrict load movement over these objects.	Procedurally limit time and height that load is carried over area of concern; define laydown area, show on drawings attached to procedure the prescribed safe load path and laydown area.
Safety Class 3: Load could be carried over spent fuel or safe shutdown equipment, but the fuel or equipment is not directly exposed to the load drop, i.e., intervening structures such as floors provide some protection.	Define safe load paths that follow, to the extent practical, structural floor members. Define laydown areas. Limit load travel height to the minimum height practical. Load paths and laydown areas shown on drawings attached to procedures.
Safety Class 4: Load cannot be carried over spent fuel or over safe shutdown equipment when such equipment is required to be operable, i.e., design or operational limitations prohibit movement.	No safe load path or special procedural actions required.

Table 2. Heavy Load Safety Classification

Safety Classification	Heavy Load	Additional Safety Classes
1	Drywell head	3
	Reactor vessel head	3
	Steam dryer	3
	Steam separator	3
2	Fuel pool gates	
	Spent fuel casks	3
	Fuel transfer shield	
	Equipment storage pool shield plugs	3
	Dryer/separator sling assembly	
	Fuel storage pool shield plugs	3
	Head strongback	
	Stud tensioner assembly	
3	Reactor vessel head insulation	
	Plant equipment	
	New fuel and shipping containers	
	Cavity shield plugs	

floor (or small obstructions) and movement will follow structural members to the extent practical.

With regard to the recirculation pump monorail and the spent fuel pool jib cranes, the Licensee stated that safe load paths are limited by the physical capabilities of the equipment. Operating procedures shall be developed, however, that will caution operators not to carry loads over or in the vicinity of spent fuel or safety-related equipment unless absolutely necessary and, if so, to limit the height and duration of the lifts.

Each heavy load lift will be controlled by a designated individual who will be responsible for enforcing procedural requirements. Deviations from these procedures and load paths require a revision to procedures or a Temporary Procedure Change, either of which must be reviewed and approved by the Plant Operations Review Committee and the resident manager.

b. Evaluation

The Licensee's method of identifying safety classes and differentiating the relative safety significance of the identified loads is consistent with NUREG-0612 guidelines.

As noted by the Licensee for Class 1 and 2 loads, the most direct route to the laydown area is most likely to be an acceptable load path. Other precautions taken by the Licensee (defining laydown areas and incorporating drawings into plant procedures) are adequate to meet the intent of Guideline 1.

Identification of specific loads paths for Class 2 and 3 designated loads and incorporation of these paths in the controlling load handling procedures meets the requirements of this guideline. The use of a knowledgeable signalman is a reasonable alternative which provides the crane operator with visual aids to ensure that load movement adheres to the established load paths. In addition, the handling of load path and procedure deviations meets the intent of Guideline 1 because the authority to approve deviations is vested in the plant operations and review committee and the resident manager.

b. Conclusion

The Oyster Creek plant complies with Guideline 1 based on the implementation of actions proposed by the Licensee.

2.1.3 Load Handling Procedures, [Guideline 2, NUREG-0612, Section 5.1.1(2)]

"Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

a. Summary of Licensee Statements and Conclusions

The Licensee has indicated that the following lifting procedures are used at the Oyster Creek plant:

205.0 - Reactor refueling

701.1.001 - Reactor vessel head removal and replacement

701.1.002 - Reactor vessel steam dryer and steam separator removal and replacement

701.1.003 - Reactor vessel insulation removal and replacement

704.1.002 - Drywell head removal and replacement

756.1.002 - Fuel transfer shield installation and removal

756.1.003 - Shield plugs removal and replacement

756.1.004 - Fuel pool gates removal and installation.

The Licensee has stated that all lifting procedures have been revised to satisfy the requirements of Section 5.1.1(2) of NUREG-0612 including:

1. description of the safety concern in the handling of heavy loads with the reactor building bridge crane
2. defined safe load paths
3. precautions
4. prerequisites
5. identification of proper handling equipment
6. training and qualification requirements for crane operators

7. verification that required detailed inspections have been performed
8. sling selection criteria
9. required crane inspection by operator prior to load handling
10. supervision of work involving a heavy load lift by a designated job supervisor
11. critical steps in order to perform the lift.

In addition, the Licensee has indicated that new procedures are being developed for the following load handling devices:

- o reactor building bridge crane
- o recirculation pump monorail and hoist
- o spent fuel pool jib cranes
- o spent fuel cask operation will be governed by a new procedure each time with special lifting requirements applicable to that particular cask.

b. Evaluation

The implementation of procedural controls on load handling at the Oyster Creek plant meets the intent of Guideline 2 of NUREG-0612 based on the Licensee's description of Oyster Creek plant lifting procedures.

c. Conclusion

The Oyster Creek plant complies with Guideline 2 of NUREG-0612.

2.1.4 Crane Operator Training [Guideline 3, NUREG-0612, Section 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [7]."

a. Summary of Licensee Statements and Conclusions

The Licensee has stated that the current practices for qualification and training of crane operators essentially cover the provisions of ANSI B30.2-1976, Chapter 2-3. However, these practices are not currently in the form of an approved procedure. Portions of the training are performed by the

maintenance supervisor and other portions are performed by the plant training staff. A new procedure with qualification records has been developed and implemented in order to formalize the program for crane operator qualification for the reactor building and spent fuel pool jib cranes. The new procedure requires that operators be familiar with appropriate handling system operating procedures and pass a practical operating examination with the handling system.

The Licensee has taken exception to ANSI B30.2-1976 with respect to Section 2-3.1.7, "Conduct of Operators, Part F." The standard requires that "before leaving the crane unattended, the operator shall land any attached load, place the controllers in the 'off' position, and open the main line device of the specific crane." However, during reactor disassembly at the Oyster Creek plant, it is necessary to keep the steam separator covered with water during handling to maintain exposure levels as low as practicable. Consequently, the separator is raised incrementally, and then left suspended until the water level rises sufficiently to allow additional raising of the separator. The separator may stay suspended at one level as long as 1.5 hours while flooding is proceeding. During these periods when the separator is left suspended, the crane operator may leave the cab until recalled. However, prior to leaving the crane, the operator places the controller in the "off" position and opens the main line device.

b. Evaluation

Crane operator training at the Oyster Creek plant is considered acceptable based on the Licensee's verification that the program meets the provisions of ANSI B30.2-1976 and that a new procedure has been developed to formalize the program. The Licensee's exception to Chapter 2-3, Section 2.-3.1.7 concerning leaving the crane unattended while loaded is reasonable based upon the specified manner in which the crane is secured. However, it should be noted that this practice appears to be in violation of Title 29 CFR 1910.179.(N).(3).(X) (OSHA) and thus should be evaluated by the Licensee unless such deviation has been previously approved.

c. Conclusion and Recommendation

The Oyster Creek plant complies with Guideline 3 of NUREG-0612 concerning crane operator training.

2.1.5 Special Lifting Devices [Guideline 4, NUREG-0612, Section 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [8]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device [NUREG-0612, Guideline 5.1.1(4)]."

a. Summary of Licensee Statements and Conclusions

The Licensee has indicated that there are six handling devices made up for special applications and currently used in handling heavy loads:

1. dryer/separator sling
2. head strongback
3. cask yokes and slings
4. fuel transfer shield slings
5. cavity shield plug lifting beam
6. equipment storage pool plug lifting beam.

The comparison of these special lifting devices to ANSI N14.6-1978 was limited to Sections 3.2 and 5 of the standard. The Licensee's review indicated the following exceptions to ANSI N14.6-1978:

1. Sections 3.1 (Designer's Responsibilities), 3.3 (Design Considerations), 4.1 (Fabricator's Responsibilities), 4.2 (Inspector's Responsibilities), and 4.3 (Fabricator's Considerations) are difficult to apply in retrospect. However, information on drawings indicates that sound engineering practices were placed on the fabricator and the inspector for the purpose of ensuring that the designer's intent was accomplished.

2. Sections 1.0 (Scope), 2.0 (Definitions), 3.4 (Design Considerations to Minimize Decontamination Effects in Special Lifting Device Use), 3.5 (Castings), and 3.6 (Lubricants) are not pertinent to load handling reliability.

3. Section 6, Special Lifting Devices for Critical Loads, is not applicable at the Oyster Creek plant because none of the loads lifted by these devices has been identified to be a critical load.

4. Plant procedures do not specify a visual inspection by maintenance or other non-operating personnel at intervals of 3 months or less as required by Section 5.3.7 of ANSI N14.6-1978. Procedures have been revised so that these devices are inspected by a qualified personnel prior to each usage and so that a thorough testing and nondestructive examination is performed prior to each refueling. Based on the controlled storage between periods of usage, dedicated single usage, and complete inspection schedule, the equivalency of Section 5.3.7 is demonstrated.

5. Section 5.3.3 of ANSI N14.6-1978 requires that special lifting devices be load tested according to Section 5.2.1 to 150% of maximum load following any incident in which any load-bearing component may have been subjected to stresses substantially in excess of those for which it was qualified by previous testing, or following an incident that may have caused permanent distortion of load-bearing parts. Since distortion may already have occurred or since defects may have already developed due to the overstressed condition, it seems more prudent and practical to perform the dimensional examinations for deformation and the NDE for defects to determine whether the device is still acceptable for use rather than subject the device to 150% load testing. If defects or deformation are detected, then the device shall be repaired or modified and tested to 150% load followed by examination for defects or deformation.

During the Licensee's review of special lifting devices against Sections 3.2 and 5 of ANSI N14.6-1978, the following results were obtained:

1. The dryer/separator sling design exceeds the criteria in ANSI B30.9 and ANSI N14.6. The lifting device has been load tested to a weight well in excess of 150% of the rated load. In addition, a preventive maintenance procedure has been developed for inspection of this lifting device in accordance with ANSI B30.9 and ANSI N14.6.

2. The head strongback drawings are available showing dimensional and material requirements and types of welds to be used for each weldment. However, information on stress analyses that may have been performed, design safety factors used, load tests performed, or processes and standards used in fabrication were not available. Accordingly, the Licensee performed a stress analysis and design

evaluation to demonstrate the adequacy of the design. As a result of this evaluation, the head strongback was found not in full compliance with ANSI specified factors of safety against bending in the lifting arms although stresses were within AISC allowables. Modifications are being made to the lifting arms to bring the head strongback into compliance with ANSI N14.6. Following these modifications, the device will be load tested in accordance with Section 5.3.2 of ANSI N14.6-1978. In addition, a preventive maintenance procedure including visual and NDE examination and inspections prior to each refueling has been developed to comply with ANSI N14.6 criteria.

3. For casks (including NAC-1) having unique special lifing devices or yokes, the lifting devices are the property of the cask owner. Accordingly, procedures have been revised to require that a certification be obtained from the cask owner, prior to handling the cask on-site, that verifies the cask lifting device or yoke design satisfies the criteria of ANSI N14.6, Section 3.2, and that the device has been inspected and maintained in accordance with ANSI N14.6, Section 5.0.

4. The fuel transfer shield sling is used for the shield and the GE200 cask. The design of the sling assembly was compared to ANSI B30.9 and found to exceed the criteria in this standard. In addition, a new preventive maintenance procedure that complies with ANSI B30.9 criteria requires inspections of the slings prior to each refueling.

5. The cavity shield plug and equipment storage pool plug lifting beams have insufficient documentation to evaluate the beams against the criteria of ANSI N14.6. Therefore, the Licensee performed a stress analysis and design evaluation of these lifting beams. As a result of this evaulation, these beams were found not to comply with ANSI N14.6 for factors of safety against bending. These beams are being modified to bring them in compliance with ANSI N14.6. Following these modifications, the devices will be load tested in accordance with Section 5.3.2 of ANSI N14.6-1978. A preventive maintenance program that includes examination and inspection to satisfy ANSI N14.6 has been developed.

A new lifting device for the core spray sparger will be evaluated against the design criteria of ANSI N14.6 when the design of the sparger and strongback are finalized.

b. Evaluation

The Oyster Creek plant satisfies the criteria of ANSI N14.6-1978 Section 3.2 (Design Criteria) for the dryer/separator sling and the fuel transfer shield sling based upon verification by the Licensee that the design meets or exceeds

the criteria in ANSI N14.6 and/or ANSI B30.9. The head strongback, cavity shield plug lifting beam, and the equipment storage pool plug lifting beam will comply after the proposed modifications and load tests have been completed.

The Licensee's response that Subsections 3.4, 3.5, and 3.6, Section 4, and Section 6 of ANSI N14.6-1978 are not applicable or pertinent is consistent with the desired intent of this guideline. The Licensee's response that design evaluations have been performed for all lifting devices and, where not in compliance, will be modified to satisfy criteria of ANSI N14.6-1978 is also consistent with the requirements of this guideline.

The preventive maintenance program that includes inspection by qualified personnel and nondestructive examination prior to use appears to address the need for continuing compliance testing set forth in Section 5 of ANSI N14.6.

The Licensee's decision to require visual inspection by nonoperating or maintenance personnel prior to each use is in keeping with ANSI N14.6-1978 requirements. In addition, load tests to be performed for the head strongback and lifting beams for the cavity shield plug and the equipment storage pool plug satisfy the guideline requirements, as does the Licensee requirement that cask owners comply with Section 5.0 of ANSI N14.6-1978. No load test is needed for the fuel transfer shield sling since it is only subject to compliance with ANSI B30.9-1971.

c. Conclusion and Recommendations

The Oyster Creek plant complies with Guideline 4.

2.1.6 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612 Section 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [9]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

a. Summary of Licensee Statements and Conclusions

The Licensee has stated that, to ensure that appropriate slings are selected for use in handling miscellaneous loads and that slings are properly maintained, the following program changes have been made:

1. Load handling procedures require the use of ANSI B30.9 criteria for sling selection and rigging techniques.

2. A new preventive maintenance procedures has been developed for annual inspection of slings.

3. Load handling procedures require a visual inspection of slings for damage prior to making a lift.

4. A tagging procedure has been developed for slings to identify sling rating, application, last examination, and expiration date of examination.

Based on an analysis performed, dynamic loading on slings associated with the reactor buildng crane were found to be approximately 3% of the static load. This 3% increase in loading is insignificant and may be disregarded.

b. Evaluation

Sling installation and usage at the Oyster Creek plant complies with NUREG-0612, Section 5.1.1(5). On the basis of information provided by the Licensee, dynamic loads are a reasonably small percentage of the overall static load and may be disregarded in rating the slings.

c. Conclusion

The Oyster Creek plant complies with Guideline 5 of NUREG-0612.

2.1.7 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Section 5.1.1(6)]

"The crane should be inspected, tested and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use when it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane

inside a PWR containment may only be used every 12 to 18 months during refueling operations and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, tests, and maintenance should be performed prior to their use)."

a. Summary of Licensee Statements and Conclusions

The Licensee has stated that new procedures for inspection, testing, and maintenance of the recirculation pump monorail, spent fuel pool jib crane, and reactor building crane are being developed. In addition, provisions have been included in the new crane operation procedures, to include appropriate operator inspections prior to load movement. With these revisions and additions, the procedures will satisfy the criteria in ANSI B30.2-1976, Chapter 2-2 without exception.

b. Evaluation

Upon implementation, the Oyster Creek plant inspection procedures will be consistent with Section 5.1.1(6) of NUREG-0612.

c. Conclusion

The Oyster Creek plant complies with Guideline 6 of NUREG-0612.

2.1.8 Crane Design [Guideline 7, NUREG-0612, Section 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes, 'and of CMAA-70 [10], 'Specifications for Electric Overhead Travelling Cranes.' An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

a. Summary of Licensee Statements and Conclusions

The Licensee has stated that the reactor building crane was designed and fabricated by Whiting Corporation to the specifications in EOCI-61 [10], "Specifications for Electric Overhead Traveling Cranes-1961" and in accordance

with additional requirements specified by the architect-engineer. The Licensee performed a review of the original specifications versus CMAA-70 (1975) and ANSI B30.2-1976. The results of this detailed point-by-point comparison were submitted in Reference 5.

b. Evaluation

The reactor building crane at the Oyster Creek plant substantially complies with the criteria specified in Guideline 7 because the original procurement specification was based on EOCI-61. In addition, for those criteria in CMAA-70 noted to be more restrictive than requirements of EOCI-61, the Licensee has demonstrated compliance with CMAA-70 or provided reasonable assurance that the existing design meets the intent of the CMAA criteria.

c. Conclusion

The Oyster Creek plant complies with Guideline 7.

2.2 INTERIM PROTECTION MEASURES

The NRC has established six interim protection measures to be implemented at operating nuclear power plants to provide reasonable assurance that no heavy loads will be handled over the spent fuel pool and that measures exist to reduce the potential for accidental load drops to impact on fuel in the core or spent fuel pool. Four of the six interim measures of the report consist of Guideline 1, Safe Load Paths; Guideline 2, Load Handling Procedures; Guideline 3, Crane Operator Training; and Guideline 6, Cranes (Inspection, Testing, and Maintenance). The two remaining interim measures cover the following criteria:

1. Heavy load technical specifications
2. Special review for heavy loads handled over the core.

Licensee implementation and evaluation of these interim protection measures are contained in the succeeding paragraphs of this section.

2.2.1 Technical Specifications [Interim Protection Measure 1, NUREG-0612, Section 5.3(1)]

"Licenses for all operating reactors not having a single-failure-proof overhead crane in the fuel storage pool area should be revised to include a specification comparable to Standard Technical Specification 3.9.7, 'Crane Travel - Spent Fuel Storage Building,' for PWR's and Standard Technical Specification 3.9.6.2, 'Crane Travel,' for BWR's, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1 [of NUREG-0612]."

a. Summary of Licensee Statements and Conclusions

A review of the Oyster Creek Technical Specifications indicates that Section 5.3.1(d) prohibits the movement of loads greater than the weight of one fuel assembly over irradiated fuel in the fuel pool.

b. Evaluation and Conclusions

The Oyster Creek plant complies with Interim Protection Measure 1.

2.2.2 Administrative Controls [Interim Protection Measures 2, 3, 4, and 5, NUREG-0612, Sections 5.3(2)-5.3(5)]

"Procedural or administrative measures [including safe load paths, load handling procedures, crane operator training, and crane inspection]... can be accomplished in a short time period and need not be delayed for completion of evaluations and modifications to satisfy the guidelines of Section 5.1 [of NUREG-0612]."

a. Summary of Licensee Statements and Conclusions

Summaries of Licensee statements and conclusions are contained in discussions of the respective general guidelines in Sections 2.1.2, 2.1.3, 2.1.4, and 2.1.7.

b. Evaluations, Conclusions and Recommendations

Evaluations, conclusions, and recommendations are contained in discussions of the respective general guidelines in Sections 2.1.2, 2.1.3, 2.1.4, and 2.1.7.

2.2.3 Special Review for Heavy Loads Handled Over the Core [Interim Protection Measure 6, NUREG-0612, Section 5.3(6)]

"...special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operation, and content of procedures."

a. Summary of Licensee Statements and Conclusions

With regard to the implementation of interim actions, the Licensee has stated that the required changes to procedures have been developed and are currently being reviewed and approved. Full implementation of the approved procedures will be effected prior to the next refueling outage.

b. Evaluation

The Licensee has adequately addressed the requirement for a review of all load handling procedures. In light of responses to Guidelines 2 and 3, it is apparent that procedures for handling loads over the core and operator training have been reviewed and upgraded as appropriate. In addition, design of cranes at the Oyster Creek plant and programs for selection and use of slings have been reviewed and found to comply with NUREG-0612.

c. Conclusion

The Oyster Creek plant complies with Interim Protection Measure 6 based upon Licensee-provided information.

3. CONCLUSION

This summary is provided to consolidate the results of the evaluation contained in Section 2 concerning individual NRC staff guidelines into an overall evaluation of heavy load handling at Oyster Creek Nuclear Power Plant. Overall conclusions and recommended Licensee actions, where appropriate, are provided with respect to both general provisions for load handling (NUREG-0612, Section 5.1.1) and completion of the staff recommendations for interim protection (NUREG-0612, Section 5.3).

3.1 GENERAL PROVISIONS FOR LOAD HANDLING

The NRC staff has established seven guidelines concerning provisions for handling heavy loads in the area of the reactor vessel, near stored spent fuel, or in other areas where an accidental load drop could damage equipment required for safe shutdown or decay heat removal. The intent of these guidelines is twofold. A plant conforming to these guidelines will have developed and implemented, through procedures and operator training, safe load travel paths such that, to the maximum extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment. A plant conforming to these guidelines will also have provided sufficient operator training, handling system design, load handling instructions, and equipment inspection to ensure reliable operation of the handling system. As detailed in Section 2, it has been found that load handling operations at Oyster Creek Nuclear Power plant can be expected to be conducted in a highly reliable manner consistent with the staff's objectives as expressed in these guidelines.

3.2 INTERIM PROTECTION MEASURES

The NRC staff has established certain measures (NUREG-0612, Section 5.3) that should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Section 5.1 is complete. Specified measures include the implementation of a technical specification to prohibit the handling of heavy loads over fuel in the storage pool; compliance with

Guidelines 1, 2, 3, and 6 of NUREG-0612, Section 5.1.1; a review of load handling procedures and operator training; and a visual inspection program, including component repair or replacement as necessary of cranes, slings, and special lifting devices to eliminate deficiencies that could lead to component failure. The evaluation of information provided by the Licensee indicates that the Oyster Creek plant complies with the staff's measures for interim protection.

4. REFERENCES

1. "Control of Heavy Loads at Nuclear Power Plants"
NRC, July 1980
NUREG-0612

2. V. Stello, Jr. (NRC)
Letter to all Licensees
Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel
May 17, 1978

3. NRC
Letter to Jersey Central Power and Light Company (JCP&L)
Subject: Request for Review of Heavy Load Handling at Oyster Creek Nuclear Power Station
December 22, 1980

4. J. T. Carroll (JCP&L)
Letter to D. G. Eisenhut (NRC)
Subject: Control of Heavy Loads
September 22, 1981

5. P. B. Fiedler (GPU)
Letter to D. G. Eisenhut (NRC)
Subject: Control of Heavy Loads
February 18, 1983

6. Telephone conference call involving J. Lombardo and A. Singh (NRC), C. Bomberger (WSI/FRC), and GPU representatives
May 27, 1983

7. American National Standards Institute
"Overhead and Gantry Cranes"
ANSI B30.2-1976

8. American National Standards Institute
"Standard for Special Lifting Deivces for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials"
ANSI N14.6-1978

9. American National Standards Institute
"Slings"
ANSI B30.9-1971

10. Crane Manufacturers Association of America, 1975
"Specifications for Electric Overhead Traveling Cranes"
CMAA-70

Exhibit D

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

JUN 26 1985

TO ALL LICENSEES FOR OPERATING REACTORS

Gentlemen:

SUBJECT: COMPLETION OF PHASE II OF "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS" NUREG-0612. (GENERIC LETTER 85-11)

On December 22, 1980, NRC issued a generic letter (unnumbered) which was supplemented February 3, 1981 (Generic Letter 81-07) regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". This letter requested that you implement certain interim actions and provide the NRC information related to heavy loads at your facilities. Your submittals were requested in two parts; a six month response (Phase I) and a nine month response (Phase II).

All licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action is not required to reduce the risks associated with the handling of heavy loads (See enclosed NUREG-0612 Phase II). Therefore, a detailed Phase II review of heavy loads is not necessary and Phase II is considered completed. However, while not a requirement, we encourage the implementation of any actions you identified in Phase II regarding the handling of heavy loads that you consider appropriate.

For each plant which has a license condition requiring commitments acceptable to the NRC regarding Phase II, an application for license amendment may be submitted to the NRC to delete the license condition citing this letter as the basis. If you have any questions, contact your Project Manager or Don Neighbors (301) 492-4837.

Sincerely,

Hugh L. Thompson, Jr., Director
Division of Licensing

Enclosure:
As Stated

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NUREG-0612, "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS"
RESOLUTION OF PHASE II

Generic Technical Activity A-36 was established to systematically examine the staff's licensing criteria, adequacy of measures in effect at operating plants and recommend necessary changes to assure the safe handling of heavy loads. The task involved review of licensee information, evaluation of historical data, performance of accident analyses and criticality calculations, development of guidelines for operating plants, and review of licensing criteria. The review indicated that the major causes of load handling accidents include operator errors, rigging failures, lack of adequate inspection and inadequate procedures. The results of the review culminated in the issuance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" in July 1980. NUREG-0612 described a resolution of Task A-36.

NUREG-0612 presents an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. The approach is directed to preventing load drops. The following summarizes this defense-in-depth approach:

1. Assure that there is a well designed handling system.

2. Provide sufficient operator training, load handling instructions, and equipment inspection to assure reliable operation of the handling system.

3. Define safe load travel paths and procedures and operator training to assure to the extent practical that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.

4. Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

5. Where mechanical stops or electrical interlocks cannot be provided, provide a single-failure-proof crane or perform load drop analyses to demonstrate that unacceptable consequences will not result.

By Generic Letters dated December 22, 1980, and February 3, 1981 (Generic Letter 81-07), all utilities were requested to evaluate their plants against the guidance of NUREG-0612 and to provide their submittals in two parts; Phases I (six month response) and Phase II (nine month response). Phase I responses were to address Section 5.1.1 of NUREG-0612 which covers the following areas:

1. Definition of safe load paths
2. Development of load handling procedures
3. Periodic inspection and testing of cranes
4. Qualifications, training and specified conduct of operators
5. Special lifting devices should satisfy the guidelines of ANSI N14.6 6.
6. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
7. Design of cranes to ANSI B30.2 or CMAA-70

Phase II responses were to address Sections 5.1.2 thru 5.1.6 of NUREG-0612 which cover the need for electrical interlocks/mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas and the specific guidelines for single-failure-proof handling systems.

We have completed our review of the utilities' submittals for Phase I for nearly all operating reactors. Only one plant still remains to be reviewed. During our review we verified that the seven guidelines listed above were providing the desired level of safety indicated in NUREG-0612. By way of the utilities' responses to the criteria of NUREG-0612, Section 5.1.1 and through discussions with our consultants based on their experiences from the reviews, we have concluded that the Phase I guidelines have provided an increased awareness by the utilities of the importance of heavy load handling.

Our review has indicated that satisfaction of the Phase I guidelines assures that the potential for a load drop is extremely small. We have note

improvements in heavy load handling procedures and training and crane and handling tool inspection and testing. These changes have been geared to limiting the handling of heavy loads over safety-related equipment and spent fuel to the extent practical, but where this can not be avoided, to accomplishing it with the operational and other features of the program implemented in Phase I. We therefore conclude that the guidelines of Phase I are adequately providing the intended level of protection against load drop accidents.

To date we have received Phase II submittals from all licensees. We interpret Phase II of NUREG-0612 as an enhancement to Phase I. Thus, prior to undertaking a review of the utilities' Phase II response for all of the operating reactors, and as a test of the adequacy of the Phase I program, we decided to undertake a pilot program with a limited number of plants. The findings from the pilot program would then provide a basis for a decision on whether to proceed with the review of the Phase II submittals for all operating reactors, to reduce the scope of the review, or to totally eliminate the review.

The pilot program involved the review of operating reactors at 12 sites, a total of 20 reactors (eight BWRs and 12 PWRs). Of the 20 reactors, 5 BWRs (Browns Ferry 1, 2 and 3 and Peach Bottom 2 and 3) have single-failure-proof cranes for all heavy load lifts. "Single-failure-proof" is used to mean a crane which meets the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Three BWR units (Dresden 2 and 3 and Big Rock Point) have taken credit for a combination of single-failure-proof cranes in some plant areas and load drop analyses in others. Five PWR reactors (Millstone 2, Prairie Island 1 and 2, and Surry 1 and 2) have utilized the load drop analysis approach. One plant (Kewaunee) has taken credit for a combination of electrical interlocks in some plant areas and load drop analyses in others. The remaining six reactors (Davis Besse, Indian Point 2, Arkansas 1 and 2 and Calvert Cliffs 1 and 2) chose to take credit for a combination of administrative controls, procedures and Technical Specification restrictions in conjunction with some type of load drop analysis. This approach does not meet the criteria of Sections 5.1.2 to 5.1.6 of NUREG-0612. Rather, it is an amplification of the guidelines of the Phase I effort, reflecting Section 5.1.1 of NUREG-0612.

It should also be noted that we have completed our review of Phase II for five operating license applicants. Of these, two (WNP-2 and Fermi-2) have single-failure-proof cranes. The remaining three (Callaway, Wolf Creek and Catawba 1 and 2) employ a combination of electrical interlocks, mechanical stops, limit switches and load drop analyses.

In addition to the detailed reviews of the Phase II reports in the pilot program and in connection with the five operating license applications, we have performed a sufficient review of all other Phase II reports to flag any outstanding plant-specific concerns reported.

From our pilot program and OL Phase II reviews, together with the above-mentioned reviews of the other Phase II reports, we have concluded that the risks associated with damage to safe shutdown systems are relatively small because:

1. nearly all load paths avoid this equipment
2. most equipment is protected by an intervening floor
3. of the general independence between crane failure probability and safety-related systems which has been observed
4. redundancy of components

We did not identify any outstanding plant specific safety concern associated with heavy loads handling.

Therefore, most of the risk appears to be associated with carrying heavy loads over or in a location where spent fuel could be damaged. The single most important example of this concerns loads handled over the open reactor vessel during refueling (such as the reactor vessel head). However, as previously mentioned, this is limited to the extent practical and where necessary, is performed with a specifically implemented program in conformance with the Phase I guidelines.

From the pilot program and OL reviews, we noted that nine of the twenty reactors, all PWRs, do not have single-failure-proof cranes. To date, we have not identified any PWRs with single-failure-proof cranes. Further, since electrical interlocks and mechanical stops are not possible for PWR polar

cranes, these reactors would be required to perform costly detailed load drop analyses. If satisfactory results could not be demonstrated from these analyses, NUREG-0612 would call for installation of a single-failure-proof crane.

Based on the above, since a single failure proof crane becomes the only solution for satisfying the NUREG-0612 criteria, the cost/benefit should be examined. Because we are dealing primarily with PWRs, the cost for modification of a polar crane to meet single failure criteria (NUREG-0554) guidelines) is approximately $30 million. This includes, as the dominant cost element, the cost of the extended shutdown which is required in order to gain access to containment. On the benefit side, given the improvements obtained from the Phase I implementation and the information obtained in the course of the pilot program and OL Phase II reviews, we cannot perceive a significant enough benefit in conversion to single-failure-proof polar cranes to warrant the high costs. (See Attachment I for a cost-benefit analysis.) We believe that the cost/benefit analysis in NUREG-0612 is no longer valid because of the benefits realized by Phase I implementation.

We believe the above assessment is further borne out by the industry experience with handling of heavy loads over the years. Precautions have been and are being taken such that no heavy load drop accidents affecting any features of the defense-in-depth against severe core-damage accidents have occurred.* This determination is also supported by the recommendation of our contractor for the pilot program reviews (Franklin Research Center) and our benefit-cost analysis suggesting that we accept other, less strigent but less costly means for Phase II compliance as an alternative to the criteria of NUREG-0612 with respect to conversion to single-failure-proof cranes.

Conclusion and Recommendation

Based on the above, we believe the Phase I implementation has provided sufficient protection such that the risk associated with potential heavy load

*There have, however, been recent occurrences of lesser severity. (See for example, IE Information Notice No. 85-12: Recent Fuel Handling Events; LER 84-015, Fort Calhoun 1, Load Over the RCS; and LER 84-006, San Onofre 2, Polar Crane Malfunction). Accordingly, nothing in this determination should be regarded as a basis for any de-emphasis of continued attention to the safe handling of heavy loads.

drops is acceptably small. We further conclude that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense in depth" is satisfied by the Phase I compliance, and that the Phase II analyses did not indicate the need to require further generic action at this time. This conclusion has been confirmed by the results obtained from the Phase II pilot program and additional Phase II reviews, which identified no residual heavy loads handling concerns of sufficient significance to demand further generic action. All plants have examined their load handling practices against the recommendations of Phase II and submitted the Phase II report. In this way, the utilities were required to identify any unexpected problems to the staff.

ATTACHMENT I

SUMMARY OF COST-BENEFIT ANALYSIS OF PWR POLAR CRANE CONVERSION TO SINGLE-FAILURE-PROOF FEATURES

SCOPE

The safety benefit of converting the polar crane in the containment of an operating or completed or nearly completed PWR to single-failure-proof features and the cost of the conversion were estimated and compared.

The safety benefit was estimated in terms of the resulting reduction in the risk of a severe accident, involving major radioactive material release, during the remaining plant life. The risk was expressed as the product of the accident probability and the population radiation dose from the release, should the accident occur.

The cost estimate included the cost of shutdown (or extension of a non-operating period) needed to accomplish the conversion.

ACCIDENT FREQUENCY ESTIMATES

Crane Failure Frequency

There were 32 crane LER events in the approximately 400 reactor-years of U.S. power-reactor operation in the 10-year period July 1969 to July 1979 (NUREG-0612, p. 4-6). None resulted in radioactive release. Of the 32 events, 17 (i.e., just over half) were apparently due to hardware design or fabrication causes, the other 15 to human factors. (Navy crane statistics, cited in NUREG-0612, for 40 load-drop or potential load-drop events in 1974-77 show 80% of the events to be due to human factors.)

It may be assumed, as a rough approximation, that Phase I of NRC's heavy-loads generic program is addressed to all the human factors causes and one-half of the hardware causes and succeeds in reducing the affected part of the failure frequency to a quite small fraction of the frequency originally present. Since human factors and hardware each contribute about one-half of the failures, approximately 3/4 of the total crane failures can be expected to be eliminated by the Phase I program. Single-failure-proof (SFP) cranes should substantially reduce the remaining 1/4 of the failure frequency, though those failures would not be eliminated altogether, since the SFP feature (as defined in NUREG-0554) does not protect against all types of possible failure (e.g., the bridge is not SFP and the SFP feature itself is subject to defeat by some types of human error). On the other hand, the SFP feature would make the cranes more "forgiving" of imperfections in the Phase I implementation. Accordingly, one may reasonably assume that the SFP feature would have a net effect of eliminating 1/4 of the pre-Phase I failure frequency.

Frequency of Accidents Involving Radioactive Release

Not all LER events involve radioactive release. In over 600 reactor-years of U.S. power-reactor operation to date [1982] there have, to our knowledge, been no radioactive releases due to load drops. The 10-year period covered by the survey in NUREG-0612, which included 32 crane LER events, all without release, represents about 60% of all U.S. power-reactor operating time to date. An assumption of a pre-fix frequency of some radioactive release once in 1,000 reactor-years appears consistent with the LER-reflected failure experience, taken together with the absence of releases to date. With 1/4 of these releases averted by an SFP crane feature, the pertinent release frequency reduction would be 1 in 4,000 RY. For the most part, these can be assumed to be minor releases due to limited fuel damage in the spent-fuel storage pool or in the reactor.

Frequency of Accidents Involving Major Releases

For a load-drop event to cause a major accident, with major radioactive release, special circumstances need to be present -- circumstances that Phase I is intended to make much less likely to occur. A highly damaging heavy load drop, such as one that could destroy a core cooling feature through violation of -- or imperfections in -- Phase I provisions combined with other failures, should be unlikely, and very unlikely to lead to major release, because of back-up safety provisions (e.g., independent additional core cooling provisions).

Review of typical load paths and associated crane-operation frequencies suggests that of all load drops in a typical PWR plant that could have radiological consequences, some 1/4 could involve equipment with a role in safe reactor shutdown, including primary-system piping. If one assumes that there is typically a 1% probability that back-up revisions would also fail, then the pertinent major-release frequency is 1 in 1,600,000 reactor-years.

Frequency Reduction

Single-failure-proof cranes may reasonably be expected to eliminate most, perhaps 90%, of the residual load-drop probability after the Phase I improvements. Thus, the frequency reduction for major release is approximately 1 in 1,800,000 RY (90% of 1/1,600,000).

It should be noted that these estimates are sensitive to plant layout. Plant-specific evaluations could, depending on case specifics, point to a much higher or lower major-release frequency estimate for a specific case. For example, should layout of a specific plant be such that a particularly unfortunate load drop could destroy all means of core cooling or incapacitate the control room (possibilities suggested by the situations at Montecello and Arkansas Nuclear 1, respectively, before remedial actions were taken at those plants), the above generic analysis could be wide of the mark

for such a plant. The major-release accident frequency could well be an order of magnitude higher for such a plant (i.e., of the order of 1 in 100,000 reactor-years) -- or even higher, depending on plant and crane features, load paths, and operating practices.

CONSEQUENCES ESTIMATE

Potential radiological consequences of load-drop accidents encompass a wide range of possibilities, depending on specific features of plant design, operating practices, and the nature and location of the specific load-drop event. We assume that some -- though very rough -- indication of the severity of the load-drop accident risks may be gained by using in these simplified calculations certain selected release categories described in WASH-1400, Appendix VI, pp. 2-1 to 2-4. Category PWR 4 was selected for the major-release estimates for pressurized water reactors.

In PWR 4 core cooling and containment both fail. Core melt occurs. This release category is used to explore consequences of a load drop that incapacitates core cooling (during or promptly after reactor operation), with containment open.

The release estimates, stated as resulting public dose, based on representative generic estimates, for a hypothetical site with a projected Year 2000 mean U.S. power-reactor-site population density, developed by Battelle Pacific Northwest Laboratories (NUREG/CR-2800) is 2,700,000 person-rem.

COST ESTIMATE

Costs of change-over to single-failure-proof cranes are subject to wide plant-specific variation, depending on the number of features of the specific cranes involved and other aspects of plant design and status.

Based on advice from the Auxiliary Systems Branch, DSI, and limited vendor and utility contacts, we take the following estimates as representative (as of 1982, when the estimates were made).

For future plants, the cost differential for original inclusion of SFP features is estimated at about $250,000 for PWRs (based on information from Ederer Crane Co.).

At the pre-operating-license stage, with no startup delay, the costs -- including planning, engineering, hardware, installation, and testing -- are estimated at $2 million per plant. This is based on the Monticello experience (1 M in 1976, adjusted for inflation). (The Monticello information was obtained from the licensee through the NRC resident inspector.)

For operating PWRs the estimated costs are dominated by plant shutdown during modifications of the polar crane located inside the containment building. (The shutdown may be an extension of a shutdown for refueling or other purposes.) The cost effect of a startup delay for a completed or nearly completed plant would be similar. With a 3-month shutdown and with shutdown costs taken as determined by the cost of replacement power at $300,000 per day, representative total change-over costs for operating PWRs are estimated at about $30 million.

RISK REDUCTION

Based on the foregoing frequency and consequences estimates, the "expected value" of the risk subject to being affected by the possible Phase II SFP feature, i.e., the magnitude of release times the frequency of its

occurrence, integrated for the remaining plant life taken as 20 years, is as follows:

$$\text{Major release risk} = 20 \times \frac{2,700,000}{2,800,000} = 30 \text{ person-rem/reactor}$$

COST-BENEFIT RATIO

The cost-benefit ratio indicated by the foregoing estimates is approximately \$1,000,000/person-rem. This estimate is subject to wide plant-to-plant variation as well as large uncertainties in the underlying estimates of accident frequency and consequences. Nevertheless, it is possible to conclude with reasonable confidence that the benefit-cost ratio for the crane conversion would fail to meet a \$1,000/person-rem worthwhileness criterion by a large margin.

Exhibit E

NUREG-1382

Safety Evaluation Report

related to the full-term operating license for
Oyster Creek Nuclear Generating Station
Docket No. 50-219

GPU Nuclear Corporation and
Jersey Central Power & Light Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1991

TABLE OF CONTENTS (Continued)

The following modifications were made to upgrade the SFPCS in the reactor building:

(1) Six new supports and nine replacement pipe supports were added.

(2) A 6-inch manually operated gate valve was added where the original SFPCS connects with the augmented SFPCS.

(3) Seismic supports were added to the SFPCS head exchangers.

These modifications thus upgraded the original SFPCS from its non-seismic condition to a condition that would ensure that the pressure boundary would remain intact and functional.

In addition to the above modifications, several SFPCS valves were qualified for operability following a seismic event to ensure an isolated, seismically qualified cooling loop. The modification ensures that the equipment, valves, piping, and supports contained in the cooling loop meet operability criteria following a seismic event and that the boundary will remain intact and functional.

9.1.2 Control of Heavy Loads at Nuclear Power Plants (Generic Task A-36)

All plants have overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where if they are accidentally dropped, they may damage safe shutdown systems. Therefore, in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980 (Generic Task A-36), all plants should satisfy each of the following criteria for handling heavy loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel pool area, in the reactor building, and in other plant areas.

(1) Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practicable, structural floor members, beams, etc., so that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee.

(2) Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include identification of required equipment, inspections and acceptance criteria required before the load is moved, the steps and proper sequence to be followed in handling the load, defining the safe load path, and other special precautions.

(3) Crane operators should be trained and qualified and should conduct themselves in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes."

(4) Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." This standard should apply to all special lifting devices that carry heavy loads in areas as defined above. For operating plants certain inspections and loads tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device on the basis of the characteristics of the crane that will be used.* This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6, which base the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device.

(5) Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load.* The rating identified on the sling should be in terms of the "static load" that produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used.

(6) The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, except that tests and inspections should be performed before use where it is not practicable to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where the frequency of crane use is less than the specified inspection and test frequency.

(7) The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976 and of CMAA-70, "Specifications for Electric Overhead Travelling Cranes" (Crane Manufacturers Association of America). An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied.

A plant conforming to these seven guidelines will have developed and implemented, through procedures and operator training, safe load travel paths so that, to the maximum extent practicable, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment. A plant conforming to these guidelines will also have provided sufficient operator training, handling-system design, load-handling instructions, and equipment inspection to ensure reliable operation of the handling system. It has been found that load-handling operations at Oyster Creek can be expected to be conducted in a highly reliable manner consistent with the staff's objectives as expressed in these guidelines.

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*For the purpose of selecting the proper sling, loads imposed by the safe shutdown earthquake need not be included in the dynamic loads imposed on the sling or lifting device.

NUREG-0612, Section 5.3, also lists certain measures that should be initiated to provide reasonable assurance that the handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612 is complete. Specified measures include the implementation of a technical specification to prohibit the handling of heavy loads over fuel in the storage pool; compliance with Guidelines 1, 2, 3, and 6 identified above; a review of load-handling procedures and operator training; and a visual inspection program, including component repair or replacement as necessary of cranes, slings, and special lifting devices to eliminate deficiencies that could lead to component failure. The evaluation of information provided by the licensee indicates that Oyster Creek complies with the staff's measures for interim protection.

By Generic Letter 85-11 dated June 81, 1985, the staff concluded that the Oyster Creek station along with other plants has provided sufficient protection so that the risk associated with potential heavy-load drops is acceptably small and that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense in depth" is satisfied.

9.2 Water Systems (SEP Topic IX-3)

Under SEP Topic IX-3, the staff reviewed the licensee's turbine building closed cooling water system, reactor building closed cooling water system, service water system, and emergency service water system to ensure that the systems have the capability to meet their design objectives and, in particular, to ensure the following:

(1) Systems are provided with adequate physical separation so that there are no adverse interactions among those systems under any mode of operation.

(2) Sufficient cooling water inventory has been provided, or adequate provisions for makeup are available.

(3) Tank overflow cannot be released to the environment without monitoring and unless the level of radioactivity is within acceptable limits.

(4) Vital equipment necessary for achieving a controlled and safe shutdown is not flooded as a result of the failure of the main condenser circulating water system.

On the basis of its review of the station service and cooling water systems for Oyster Creek, the staff concluded that the essential system and function are the emergency service water system for torus heat removal.

In a letter dated November 13, 1981, the staff determined that the design of the above system conforms with current regulatory guidelines and with GDC 44 regarding the capability and redundancy of the essential functions of the system.

9.3 Ventilation Systems (SEP Topic IX-5)

10 CFR Part 50 (GDC 4, 60, and 61), as implemented by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that the ventilation systems have the capability to provide a safe environment for plant personnel and for engineered safety