January 23, 2002

Mr. Oliver D. Kingsley, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: HANDLING OF HEAVY LOADS OVER IRRADIATED FUEL STORED IN THE SPENT FUEL POOL (TAC NO. MB1747)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 223 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated April 4, 2001, as supplemented on October 12, November 28, November 30, December 7, and December 20, 2001.

The amendment deletes Technical Specifications (TSs) 5.3.1.B and 5.3.1.C. These TSs restricted the handling of heavy loads over irradiated fuel stored in the spent fuel pool. The basis for deleting these TSs is the upgrade of the reactor building crane and associated handling systems to a single-failure proof 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.

Please note that the Nuclear Regulatory Commission (NRC) staff found your original application lacking in providing the necessary information to conduct our evaluation. The NRC staff had to expend significant resources to obtain this information. For example, the staff held numerous teleconference calls, a meeting, and one very lengthy request for additional information. The NRC staff believes that the original application should have addressed issues beyond the cask drop protection system (CDPS). If part of the basis for supporting the amendment for the removal of the CDPS requirements is based on the single-failure proof crane, then the original submittal should have contained whatever information was necessary for the NRC staff to review the single-failure proof crane. This information was not provided. The NRC is facing significant challenges to its resources. It is imperative that licensees' submittals are of high quality such that the NRC can perform its review in an efficient and effective manner. We request that you take appropriate action to improve your quality of submittals.

If you have any questions, you may reach me at (301) 415-1261.

Sincerely,

/RA/

Helen N. Pastis, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 223 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

quality such that the NRC can perform its review in an efficient and effective manner. We request that you take appropriate action to improve your quality of submittals.

If you have any questions, you may reach me at (301) 415-1261.

Sincerely,

/RA/

Helen N. Pastis, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 223 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	SLittle	JMunday	HPastis	KManoly
PD1-1 R/F	JKnox	GHill (2)	GHatchett	
JZwolinski/TMarsh	OGC	JHannon	JMa	
EAdensam	ACRS	JRogge, RI	BThomas	

**See previous concurrence

Accession Number: ML013410156 *SE inputs dated 12/05/01 and 12/12/01 were provided and no major revisions were made.

OFFICE	PDI-1/PM	PDI-1/LA	OGC *	PDI-1/ASC	SPLB*	EMEB*
NAME	HPastis	SLittle	AHodgdon**	JMunday	BThomas	KManoly
DATE	01/22/02	01/22/02	12/18/01	01/23/02	12/05/01	12/12/01

OFFICIAL RECORD COPY

cc:

John Skolds
Chief Operating Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

William Bohkle
Senior Vice President Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

John B. Cotton
Senior Vice President - Operations Support
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Joseph J. Hagan
Senior Vice President -
Mid Atlantic Regional Operating Group
Exelon Generation Company, LLC
200 Exelon Way, Suite 305
Kennett Square, PA 19348

Kevin P. Gallen, Esquire
Morgan, Lewis, & Bockius LLP
1800 M Street, NW
Washington, DC 20036-5869

Kent Tosch, Chief
New Jersey Department of
Environmental Protection
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

Mr. Jeffrey A. Benjamin
Vice President -
Licensing and Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Mr. Ronald J. DeGregorio
Vice President
Oyster Creek Nuclear Generating Station
AmerGen Energy Company, LLC
PO Box 388
Forked River, NJ 08731

H. J. Miller
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 445
Forked River, NJ 08731

Michael P. Gallagher
Director - Licensing
Exelon Generation Company, LLC
Correspondence Control Desk
P.O. Box 160
Kennett Square, PA 19348

Ernest J. Harkness
Plant Manager
Oyster Creek Nuclear Generating Station
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

Manager Regulatory Assurance
Oyster Creek Nuclear Generating Station
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

Oyster Creek Nuclear Generating Station

cc:
Edward J. Cullen, Jr., Esquire
Vice President, General Counsel and
Secretary
Exelon Generation Company, LLC
300 Exelon Way
Kennett Square, PA 19348

J. Rogge, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
License No. DPR-16

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

   A. The application for amendment by AmerGen Energy Company, LLC, et al., (the licensee), dated April 4, 2001, as supplemented on October 12, November 28, November 30, December 7, and December 20, 2001, 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. 223, are hereby incorporated in the license. AmerGen Energy Company, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 23, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 223

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 223

TO FACILITY OPERATING LICENSE NO. DPR-16

AMERGEN ENERGY COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated April 4, 2001, as supplemented on October 12, November 28, November 30, December 7, and December 20, 2001, the AmerGen Energy Company, LLC, (AmerGen or the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station (Oyster Creek) Technical Specifications (TSs). The requested changes would delete TSs 5.3.1.B and 5.3.1.C. These TSs restrict the handling of heavy loads over irradiated fuel stored in the spent fuel pool. Specifically, TS 5.3.1.B restricts movement of heavy loads over stored irradiated fuel and requires the shield plug and its associated lifting hardware to be within a transfer cask while moving over stored irradiated fuel within the cask drop protection system (CDPS). TS 5.3.1.C limits the vertical height of a spent fuel shipping cask to 6 inches while being moved over the CDPS. In addition, TS 5.3.1.C requires vertical limit switches be operable to assure the 6-inch restriction is maintained. The basis for deleting these TSs is the upgrade of the reactor building crane and associated handling systems to a single-failure proof system. The November 28, November 30, December 7, and December 20, 2001, letters provided clarifying information within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Background

NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," dated May 1979, identifies features of the design, fabrication, installation, inspection, testing, and operation of single-failure-proof overhead crane handling systems that are used for handling critical loads. The NUREG superseded Draft Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," dated 1976.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines in two phases (Phase I and II) for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads and provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, design, testing, inspection, and maintenance of cranes and lifting devices, and analyses of the impact of heavy load drops. Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations. NUREG-0612, Appendix C provides alternative means of upgrading the reliability of the crane to satisfy the guidelines of NUREG-0554.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the guidelines of NUREG-0612 Phase II based on the improvements obtained from the implementation of NUREG-0612 Phase I. GL 85-11, however, encouraged licensees to implement actions they perceived to be appropriate to provide adequate safety.

In Nuclear Regulatory Commission Bulletin (NRCB) 96-02, "Movement of Heavy Loads over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated April 1996, the NRC staff addressed specific instances of heavy load handling concerns and requested licensees to provide specific information detailing their extent of compliance with the guidelines and their licensing basis.

Restriction on movement of transfer casks and other heavy loads, in the vicinity of the spent fuel storage pool, at Oyster Creek are specified in TSs 5.3.1.B and 5.3.1.C. Because use of the existing non-single-failure-proof crane and the CDPS reduces operational efficiency and increases radiological exposure, the licensee proposes to delete such restrictions from the TSs. As an alternative to the restrictions in the TSs and the CDPS, the licensee has upgraded its crane to single-failure-proof using the guidelines in NUREG-0612, Appendix C, and NUREG-0554 to support the deletion of TS 5.3.1.B and 5.3.1.C. The NRC staff's evaluation of the licensee's request is provided below.

2.2 <u>Deletion of TSs 5.3.1.B and 5.3.1.C</u>

A heavy load at Oyster Creek is considered to be any load greater than 800 pounds, which is the approximate weight of a fuel assembly and its associated handling tool. Oyster Creek has proposed to delete TS 5.3.1.B because it prohibits (1) heavy loads from being moved over stored irradiated fuel in the spent fuel pool and (2) requires the shield plug and its associated lifting hardware to only be moved over fuel assemblies within a transfer cask while it is in the CDPS. TS 5.3.1.C limits the height to which a cask can be moved above the CDPS to 6 inches as well as requires limit switches to be operable to assist in assuring the 6-inch height restriction is maintained during cask movement above the CDPS. As stated in the December 20, 2001 letter, the change to the Bases for TS 5.3.1 deletes only the reference to the analysis that states that the dropped waste cans will not damage the spent fuel pool liner. The licensee provided the following justification for deletion of each of the aforementioned TSs and portions of the Bases:

> The proposed Technical Specification change will eliminate restrictions on movement of heavy loads, i.e., loads greater than the weight of a spent fuel assembly and its handling tool, over spent fuel in the spent fuel storage

pool (SFSP). It also removes requirements relating to the design function of the CDPS for cask moves into the SFSP. To effect this, the reactor building crane has been upgraded to single-failure-proof as defined by NUREG-0612.

These sections of the TSs provide restrictions applicable to the safe handling of heavy loads at Oyster Creek using the non-single-failure-proof 100-ton main and 10-ton auxiliary (100/10) hoists and trolley to the reactor building crane. By upgrading the handling system to include single-failure-proof features consistent with NUREG-0612, Appendix C, and NUREG-0554, restrictions applicable to the previous reactor building crane will no longer be needed.

The new 105/10 ton single-failure-proof heavy load handling system, which is fully compliant with NUREG-0554 (see Section 3.2 of this safety evaluation (SE)) reduces the probability of a handling accident. Moreover, the licensee, in its October 12, 2001, response to the NRC staff's request for additional information (RAI), stated that its heavy load handling program would continue to meet NUREG-0612 Phase I guidelines as approved by the NRC staff in an SE dated June 21, 1983. The combination of the crane upgrade to single-failure-proof and meeting NUREG-0612 Phase I guidelines satisfies the defense-in-depth philosophy of NUREG-0612 and assures a consistent level of protection in handling heavy loads at Oyster Creek.

In addition, to satisfy the defense-in-depth methodology described in NUREG-0612, the licensee made the following commitments in its letters:

- In accordance with NUREG-0612, Section 5.1.1(1), "Safe Load Paths," procedures governing heavy load paths will be revised to require authorization from the Plant Manager and Engineering Director, or their designee, to use load paths over the reactor cavity with the shield blocks removed, or over the spent fuel storage pool that have not been previously analyzed;

- Plant procedures will be revised to prevent heavy load travel over "HOT" irradiated fuel;

- Plant procedures will be revised to minimize the length of travel of heavy loads over spent fuel; and

- All new lifting devices and interfacing lift points, associated with heavy loads handled by the reactor building crane, will meet the guidelines in NUREG-0612, Section 5.1.6 (5.1.6(1) and 5.1.6(3)).

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. The NRC staff has determined that the commitments do not warrant the creation of regulatory requirements which would require prior NRC approval of subsequent changes. The NRC staff has agreed that NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," provides reasonable guidance for the control of regulatory commitments made to the NRC staff. (See Regulatory Issue Summary 2000-17, Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff, dated September 21, 2000.) The commitments should be controlled in accordance with the industry guidance or comparable criteria employed by a specific licensee. The NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

Accordingly, the lifting devices and interfacing lift points will meet the requirements in American Nuclear Standard Institute (ANSI) N14.6 1978, “Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Power Plants.” Therefore, if a single lifting device is provided, it will have twice the design safety factor of 6 times the tensile strength and 10 times the ultimate strength as required by ANSI N14.6, Subsection 6.2.1. Interfacing lift points, such as lifting lugs or cask trunions, will have a design safety factor of 5 times the maximum static and dynamic load. In the case of a non-redundant lift point system, a design safety factor of 10 is required for the combined static and dynamic load.

The licensee’s commitments to (1) restrict the handling of heavy loads over or in proximity to the spent fuel pool, and (2) meeting the Phase I guidelines of NUREG-0612 combined with increasing the handling system reliability to meet the single-failure guidelines of NUREG-0612, Appendix C and NUREG-0554 will provide reasonable assurance that handling of heavy loads at Oyster Creek will be performed in a safe manner. Therefore, the NRC staff finds the proposed deletion of TSs 5.3.1.B and 5.1.3.C acceptable.

2.3 Single-Failure-Proof Load Handling System Upgrade

As stated in NUREG-0554, when reliance for the safe handling of heavy loads is placed upon the crane system, the crane should be designed such that a single failure will not result in the loss of the capability of the system to safely retain the load. NUREG-0554 identifies the features of the design, fabrication, installation, inspection, testing and operation of the single-failure-proof overhead crane systems which are used for handling of heavy loads. These features are limited to the hoisting system and the braking systems for the trolley and the bridge. Other load bearing components (e.g., girders) should be conservatively designed, but are not required to be considered single-failure-proof. Also, NUREG-0612, Appendix C, provides alternative means to satisfy certain guidelines of NUREG-0554 when an existing crane system is upgraded to single-failure-proof.

In response to the NRC staff’s RAI, the licensee submitted a matrix, dated October 12, 2001, detailing how the crane handling system meets the guidelines established in NUREG-0554. Also, included in the licensee’s response was detailed information concerning how specific requirements of the guidelines of NUREG-0612, Section 5.1.6 and Appendix C were met in the upgrade of the crane. The NRC staff reviewed the matrix to determine if the licensee adequately addressed the guidelines and whether the licensee took any exceptions to the guidelines of NUREG-0554. During discussions with the licensee, the NRC staff indicated that exceptions were noted; however, the licensee had not provided adequate justification. In addition, an entire subsection of the matrix had been inadvertently omitted from the October 12, 2001, response to the NRC staff’s RAI. As a result, the licensee, in a letter dated November 28, 2001, provided justification for exceptions taken to the guidelines in NUREG-0554 and submitted the subsection omitted from the October 12, 2001, response to the NRC staff’s RAI. These specific Oyster Creek exceptions to NUREG-0554 are listed and evaluated below:

- Section 2.4 specifies Charpy tests per the American Society for Testing and Materials (ASTM) A-370; however, as an alternative for cranes already fabricated or operating, coldproof testing is recommended to assure the absence of brittle-fracture tendency in the crane material. NUREG-0612, Appendix C, omitted coldproof testing because the minimum ambient temperature is 70 °F (21 °C), which exceeds the Non-Destructive Testing Temperature + 60 °F requirement for most structural steels of comparable dimensions. Recognizing that this is not required by NUREG-0612, Appendix C, Oyster Creek conducted Charpy tests on the old trolley since the trolley and the bridge were made from the same type of steel (ASTM A7). Also, Oyster Creek performed metallurgical examination of a test plug cut from the existing crane bridge and the Charpy V-notch test from the discarded trolley for comparison and found them to be similar. The NRC staff finds that the licensee meets the intent of the NUREG-0554 guidelines to verify the structural integrity of the crane bridge through comparative sampling and testing of the discarded trolley materials.

- Section 4.9 specifies that the static and dynamic alignment of all hoisting machinery components, including gearing, shafting, couplings, and bearings, should be maintained throughout the range of loads to be lifted, with all components positioned and anchored on the trolley machinery platform. The licensee chose the American Society of Mechanical Engineers (ASME) Nuclear Overhead and Gantry Cranes (NOG)-1-1998 to meet the guidelines of the NUREG. The licensee compared the criteria found in NOG-1-1998 to Regulatory Guide (RG) 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis;" and RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants;" and determined that the values used in the NOG satisfies the intent of the guidelines in the regulatory guides. In addition, the criteria found in ASME NOG-1-1998 satisfy position 2 of RG 1.29, "Seismic Design Classification." As a result, the NRC staff finds the criteria used in the ASME NOG-1-1998 acceptable for meeting the above guidelines identified in NUREG-0554.

- Section 7.2 specifies that, following installation, the integrity of all control, operating, and safety systems be verified as to satisfaction of installation and design requirements. The licensee used ASME NOG-1-1998 in lieu of ANSI B30.2, "Overhead and Gantry Cranes." The NRC staff notes, however, that most requirements of B30.2 are captured in ASME NOG-1-1998. Consequently, pre-operational test, inspection, and no-load test performed in accordance with NOG-1-1998 provide an acceptable alternative to the test specified under B30.2.

- Section 8.1 specifies that a complete check be made of all the crane's mechanical and electrical systems to verify the proper installation and prepare the crane for testing. The licensee used certain sections of ASME NOG-1-1998, as indicated in their compliance matrix, to conduct the required testing. As stated previously, use of the ASME NOG-1-1998 provides an acceptable alternative to crane testing specified by the NUREG as directed in ANSI B30.2.

- Section 8.2 specifies that the crane system be static load tested at 125 percent of maximum critical load (MCL) and subjected to full performance tests at 100 percent of the MCL for all speeds and motions for which the system was designed. The licensee did not conduct the specified test; however, the licensee proposed some alternative testing to meet the intent of the NUREG. The licensee upgraded the crane controls in 1996 on the recently discarded 100/10-ton hoist and trolley. The 100/10-ton hoist and trolley were tested at 125 percent of the design rated load and its travel was limited to the area of the equipment hatch due to safety considerations. In addition, the new 105/10-ton hoist and trolley were tested in accordance with the NUREG at the manufacturer's facility and shipped whole to Oyster Creek and installed on the bridge. Because the new trolley weighed over 12,000 pounds less than the old system, complete testing was performed at the manufacturer's facility, and the new system was shipped to Oyster Creek without change in configuration. The NRC staff finds the alternative testing to be acceptable.

- Section 9 specifies that the crane designer and manufacturer provide a manual of information and procedures for use in checking, testing, and operating the crane. The licensee's manual was developed using the criteria of ASME NOG-1-1998, which the NRC staff finds acceptable.

As stated by the licensee in response to the NRC staff's RAI on October 12, 2001, the CDPS complicated cask-handling activities, resulting in potential increased dose, additional hang time during cask moves, and the potential for additional lifts to assure that limit switches were properly adjusted. By upgrading the handling system to meet the single-failure criteria of NUREG-0554 and continuing compliance with the NUREG-0612 Phase I guidelines, the licensee no longer required the design function of the CDPS. Thus, although the CDPS would not be removed from the SFSP, reliance on the single-failure capability of the handling system would provide for uncomplicated handling activities and reduce occupational doses. Because the licensee satisfies the defense-in-depth philosophy of NUREG-0612 Phase I and II, removal of the CDPS design function is acceptable.

Based upon this evaluation, the NRC staff finds that the upgraded handling system at Oyster Creek satisfies the applicable requirements of NUREG-0612, Appendix C, and NUREG-0554 for single-failure-proof cranes.

2.4 Single-Failure-Proof Load Handling System Seismic Design

NUREG-0554 recommends specific design requirements for above the wheels of the proposed replacement overhead crane during a seismic event. The structures above the wheels should be considered to retain control of and hold the load. The bridge and trolley should be designed to remain in place on their runways with the wheels prevented from leaving the tracks. If a seismic event comparable to a safe-shutdown earthquake (SSE) occurs, the bridge is required to remain on the runway with brakes applied, and the trolley is required to remain on the runway with the crane girders with brakes applied. The pendulum and swinging effects due to seismic and other operational loads including the maximum critical load (MCL) are to be considered in the seismic design of these crane components.

The licensee used the response spectrum method of dynamic analyses to analyze and design the new trolley and to evaluate the existing bridge girders. The analysis used the 4 percent operational basis earthquake (OBE) and 7 percent SSE spectra as the input for the seismic analysis. The NRC staff reviewed the design information submitted for the trolley and hoist, and the evaluation of the bridge structure, and determined that the licensee had not adequately addressed whether any of the load cases analyzed would result in uplift of the gantry trucks. In its October 12, 2001, response to the NRC staff's RAI, the licensee indicated that under all load cases no uplift (tension) was experienced from the wheels of the trolley on the bridge and bridge on the crane rails. As a result, the bridge and trolley are designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event in accordance with Section 2.5 of NUREG-0554.

Based upon the licensee's evaluation, the NRC staff agrees that the components above the wheels on the upgraded crane at Oyster Creek satisfy the seismic and structural guidelines of NUREG-0554. The licensee's analysis demonstrates that the capability of components to withstand a seismic event are within acceptable limits. Therefore, the new trolley and hoist, and the existing bridge, will safely perform its intended function of retaining a MCL of 105 tons under OBE and SSE conditions.

2.5 Structural and Seismic Evaluation

2.5.1 Reactor Building Steel Superstructure Analysis

The purpose of this analysis was to qualify the Oyster Creek reactor building steel superstructure for increased loading due to an upgrade to a single-failure-proof 105-ton capacity crane. The licensee indicated in Reference 5 that the requirements in NUREG-0554 necessitate the qualification of the reactor building steel superstructure for the increased crane load during normal operation and taking into account the seismic effects.
A finite element model was utilized for the analysis of the reactor building steel superstructure including the crane. The model includes the geometry, mass, and stiffness of roof trusses, building columns and their cross bracing around the perimeter of the operating floor, horizon purlins holding the building siding, and the trolley, crane girders and end trucks. The computer code, SAP 2000 Plus, Version 7.21, was used to perform the analysis. The weight of the structure was included in the model. A live (snow) load of 7 pounds per square foot (psf) was provided in the analysis. The 105-ton crane live load was included in the mass of the trolley. The seismic inputs were the 4-percent damping OBE and 7-percent damping SSE floor response spectra from the Oyster Creek Updated Final Safety Analysis Report (UFSAR) at elevation 119'-3", where the steel superstructure is anchored at the concrete. The loading combinations used in the analyses include the weight of the structure, the live load, and the crane live load plus OBE or SSE loads.

The analysis results indicated that the maximum stress for columns, column bracing, crane rail girders, roof truss chords, connections, and roof truss bracing for OBE condition are no greater than the allowable stresses specified in the American Institute Steel Construction (AISC) manual and for SSE condition are no greater than 1.6 times the allowable stresses specified in the AISC manual. The analysis results also indicate that the shear forces generated at the column bases are less than the allowable stress in the shear lugs, weld, and bolts and the maximum uplift forces in columns due to SSE are less than the capacity of anchor bolts.

The NRC staff finds the mathematical model, which represents the geometry, mass, and stiffness of the steel superstructure adequate. The SAP computer program is bench-marked and its use is acceptable for structural analysis applications. The use of 4-percent damping for OBE floor response spectra and 7-percent damping for SSE floor response spectra for bolted steel structures is acceptable because it complies with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." The licensee has correctly applied the seismic input at the bottom of the steel superstructure. The loading combinations used for the analysis are acceptable because they represent the potential loads the structure will be subjected to and are consistent with the design basis in the UFSAR. The analysis results have demonstrated that the reactor building steel superstructure is adequate for an upgrade to a single-failure proof 105-ton capacity crane.

Reference 2 states that the analysis provided an allowance of 7 psf for snow load on the roof of the reactor building steel superstructure. The NRC staff requested a justification for the adequacy of the licensee’s provision of the snow load. The licensee provided a static analysis for the roof trusses for an additional 5 psf snow load (i.e. for a total of 12 psf) in Reference 4, and the results indicate that the increased stresses in the members of the roof truss are not significant. In Reference 5, the licensee provided its justification that the use of 12 psf for snow load meets the requirement of the building code. The NRC staff considers the licensee’s responses to the snow load adequate.

2.5.2 Seismic Qualification Of Oyster Creek Reactor Building Crane

The purpose of this analysis was to qualify the Oyster Creek reactor building crane for increased loading due to an upgrade to a single-failure-proof 105-ton capacity. Reference 5 indicates that NUREG-0554 requires a qualification for the crane subjected to normal operating loads plus seismic effects.

A new trolley was mounted on the existing bridge, which is spanned 103'-4" between runway girders. Seismic qualification of the new trolley was performed in accordance with CMAA 70 (1999), Class D for Crane Manufacturer Association of America, Inc. (CMAA) Cases 1, 2, and 3. Seismic qualification of the existing bridge and bogeys was performed in accordance with CMAA 70 (1975) for CMAA Cases 1, 2, and 3. All structural components of the existing crane and the new trolley were seismically qualified for a 105-ton lifted load. A postulated “broken rope” accident condition was analyzed. The STARDYNE computer program was used to perform the analysis. The model is a three-dimensional linear elastic, small displacement finite elements model. The model includes the entire crane assembly as well as reactor building runway rails. The seismic inputs were the 4-percent damping OBE and 7-percent damping SSE floor response spectra at the top of reactor building runway girder. Loading combinations specified in CMAA 70 (1999) were used for the analysis. The analysis results indicate that the loading combination of dead load of the bridge and attached equipment plus dead load of the trolley and attached equipment plus SSE produces the maximum stresses and, therefore, controls the design. The maximum stresses are less than 1.33 times the normal CMAA 70 (1975) allowable stresses. In the telephone conference held on November 26, 2001, the licensee stated that the values of 1.33 times the normal CMAA 70 (1975) allowable stresses are less than yield stress of the steel material. A broken rope accident analysis was also performed.

The analysis results indicate that all trolley members meet the acceptance criteria and all connections are considered to be adequate. The NRC staff finds the use of STARDYNE computer program acceptable because it is benched-marked for structural analysis applications of this type. The NRC staff finds the model that combines the crane assembly and the runway girders adequate because it includes the dynamic interaction between the two substructures. The NRC staff finds the use of 4-percent damping for OBE and 7-percent damping for SSE floor response spectra for bolted steel structures acceptable because it complies with RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants.” The NRC staff finds that the analysis results have demonstrated the adequacy of the Oyster Creek reactor building crane.

The connector between the end trucks was initially modeled (Reference 3) as a fixed end member which resulted in significant bending stress at the point of fixity. A more realistic representation (Reference 4) of the connector member was made by considering it as a pinended member. Reference 5 indicates that the connector would not buckle under a SSE. The NRC staff considers the licensee's responses adequate.

2.6 Concluding Remarks

The NRC staff finds that the proposed revisions to the TSs are acceptable and will not represent a decrease in oversight and control of the crane travel by the licensee. To ensure the continued safety of the spent fuel in the pool during movement of the transfer cask and the handling of other heavy loads, the licensee has committed to (1) revise plant procedures to prevent heavy load travel over "HOT" irradiated fuel, (2) revise procedures to minimize the length of travel of heavy loads over spent fuel, and (3) continue to follow the guidelines of NUREG-0612, Phase I.

Based upon the preceding discussions, the NRC staff concludes that the proposed revisions to the TS associated with upgrading the existing crane at Oyster Creek to single-failure-proof are in accordance with NUREG-0612 and NUREG-0554. Based upon this evaluation, the NRC staff finds that the use of the proposed crane, coupled with the special lifting devices, will enable the licensee to handle heavy loads with little or no risk to irradiated fuel stored in the spent fuel pool. The staff further finds that the licensee's proposed personnel training, equipment inspections, definition of safe load paths, and procedural controls, which preserve the NUREG-0612 Phase I guidelines, provide adequate defense-in-depth to maintain safety during heavy load handling operations in the vicinity of the spent fuel pool at Oyster Creek.

Based on the information provided by the licensee, the NRC staff finds the licensee's qualification of the Oyster Creek reactor building steel superstructure and the Oyster Creek reactor building crane for increased loading due to an upgrade to a single-failure-proof 105-ton capacity crane acceptable.

3.0 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.

4.0 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, and there has been no public comment on such finding (66 FR 31702). 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.

5.0 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 regulations, 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.

6.0 REFERENCES

1. Letter from Ron J. DeGregorio of AmerGen to Document Control Desk of NRC, dated April 4, 2001.

2. Letter from Ron J. DeGregorio of AmerGen to Document Control Desk of NRC, dated October 12, 2001.

3. Letter from Ron J. DeGregorio of AmerGen to Document Control Desk of NRC, dated November 28, 2001.

4. Letter from Ron J. DeGregorio of AmerGen to Document Control Desk of NRC, dated November 30, 2001.

5. Letter from Ron J. DeGregorio of AmerGen to Document Control Desk of NRC, dated December 7, 2001.

Principal Contributors: G. Hatchett
J. Ma

Date: January 23, 2002