December 21, 2004

Mr. Dennis L. Kehl
Site Vice-President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6590 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 07200005/2004-003(DNMS)

Dear Mr. Kehl:

On November 19, 2004, the NRC completed a routine spent fuel storage team inspection at the Point Beach Nuclear Power Plant. The purpose of this routine team inspection was to determine whether four dry fuel casks were loaded and transferred to the Independent Spent Fuel Storage Installation safely and in accordance with applicable regulations. At the conclusion of onsite inspections on October 28 and November 19, 2004, the NRC inspectors discussed the preliminary inspection findings with members of your staff. On December 13, 2004, at the conclusion of our in-office review, a final exit meeting was conducted by telephone between members of your staff and the inspectors to discuss the final disposition of the issues identified during the inspection.

The inspection consisted of examinations of dry fuel storage activities at the Point Beach Nuclear Power Plant as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Areas examined during the inspection are identified in the enclosed report. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities in progress, and interviews with personnel.

Based on the results of this inspection, the NRC has identified an issue associated with your seismic analysis of the auxiliary building and the auxiliary building crane. The NRC has also determined that a violation is associated with this issue. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy. The current Enforcement Policy is included on the NRC web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Enforcement Policy.** The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, DC 20555-0001, and the Resident Inspector at the Quad Cities Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's

document system (ADAMS). The NRC's document system is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Please note that on October 25, 2004, the NRC terminated public access to ADAMS and initiated an additional security review of publicly available documents to ensure that potentially sensitive information is removed from the ADAMS database accessible through the NRC's web site. Interested members of the public may obtain copies of the referenced documents for review and/or copying by contacting the Public Document Room pending resumption of public access to ADAMS. The NRC Public Documents Room is located at NRC Headquarters in Rockville, MD, and can be contacted at (800) 397-4209 or (301) 415-4737 or pdr@nrc.gov. We will gladly discuss any questions you may have regarding this inspection.

Sincerely,
/RA by W. Snell Acting for/

Kenneth G. O'Brien, Chief
Decommissioning Branch

Docket No. 07200005

Enclosure: Inspection Report 07200005/2004-002

cc w/encl: F. Kuester, President and Chief
Executive Officer, We Generation
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D. Cooper, Senior Vice President, Group Operations
J. McCarthy, Site Director of Operations
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Regulatory Affairs Manager
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Site Assessment Manager
Site Engineering Director
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REGION III

Docket No.	07200005
Report No.	07200005/2004-003
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Power Plant
Location:	6590 Nuclear Road Two Rivers, WI 54241-9516
Dates:	October 25 through 28, 2004, and November 17 through 19, 2004. In-office review: November 29 through December 13, 2004.
Inspectors:	Ross B. Landsman, Project Engineer Magdalena R. Gryglak, Reactor Inspector Christopher G. Martin, Reactor Inspector
Approved by:	Kenneth O'Brien, Chief Decommissioning Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Point Beach Nuclear Power Plant
NRC Inspection Report 07200005/2004-003(DNMS)

This inspection included direct observation by the inspectors of various phases of the loading activities associated with the Independent Spent Fuel Storage Installation (ISFSI) as well as a review of the plant conditions to support the loading activities. The objective of this inspection was to establish whether the Point Beach Nuclear Power Plant transferred fuel to a spent fuel storage cask and subsequently to the ISFSI safely and in accordance with applicable regulations. Overall, the loading activities were performed satisfactorily.

Operation of Independent Spent Fuel Storage Installation (IP 60855, IP 60855.1)

- The inspectors determined that the licensee's loading and transfer of four casks to the ISFSI was adequate and met the requirements in the CoC and associated Technical Specifications. (Section 1.1)

- The inspectors identified a Non-Cited Violation of 10 CFR 72.122 (b), Section (2)(i), "Overall Requirements." Specifically, between October 25 and October 30, 2004, the licensee utilized the crane without performing a complete written evaluation to demonstrate that the crane was adequate to lift the weight of a fully loaded NUHOMS-32PT cask under a seismic event. The violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, the inspectors identified one Unresolved Item associated with the licensee's analysis of the auxiliary building structure and the crane design basis during a seismic event. (Section 1.2)

Report Details

1.0 Operation of Independent Spent Fuel Storage Installation (IP 60855, IP 60855.1)

1.1 Dry cask loading activities

a. Inspection Scope

The inspectors evaluated portions of the licensee's loading of four casks and transfer of the casks to the Independent Spent Fuel Storage Installation (SFSI) to verify compliance with the applicable Certificate of Compliance (CoC) conditions and associated Technical Specifications.

b. Observations and Findings

The inspectors observed the licensee perform activities associated with the loading of four Dry Shielded Canisters (DSCs). The activities included loading of the DSC and the transfer cask with spent fuel, decontamination of the cask, movement of the transfer cask to the ISFSI pad, and placement of the DSC inside the Horizontal Storage Module (HSM). During the loading campaign, the inspectors observed the licensee promptly respond to and successfully resolve several unexpected conditions that occurred. These are discussed below. Overall, the inspectors noted that the licensee performed the loading activities adequately and in accordance with the applicable regulations.

During the loading of the first cask, the licensee observed radiation levels that were much higher than expected. The dose rate was 80 millirem (mrem) at a distance of three feet, as the licensee lifted the DSC and the Transfer Cask (TC) to the top of the water surface in the spent fuel pool. The licensee predicted this value in its pre-operational calculations. After the DSC and the TC broke the spent fuel pool water surface, the licensee removed 200 gallons of water from the DSC. Removal of 200 gallons of water was a procedural requirement. Immediately following this step, the dose rate rose to 700 millirem per hour (mrem/hr) on contact and 400 mrem/hr at three feet. The licensee immediately stopped all work to evaluate the condition. The licensee verified that the Technical Specification requirements were met. The vendor's CoC Technical Specifications state that "dose rated from the transfer cask shall be limited to levels which are less or equal to: 1) 200 mrem/hr at three feet with water in the DSC cavity, and 2) 500 mrem/hr at three feet without water in the DSC cavity." The licensee had 80 mrem/hr and 400 mrem/hr, respectively, which did not exceed the Technical Specification values. In addition, the staff verified that the correct fuel assemblies were placed inside the DSC and consulted with the vendor to confirm that the radiation levels are within the expected values seen throughout the industry. Also, the licensee contacted the dry cask storage project personnel from the Palisades Nuclear Plant to ascertain the differences in radiation levels since both plants used the same dry cask storage design and similar procedures. After further review, the licensee concluded that the different radiation levels resulted from the different fuel design utilized at each plant. The Point Beach fuel mass at the top nozzle was greater by approximately 4 pounds, which resulted in a greater content of fission products. Draining 200 gallons of water resulted in uncovering the top portion of fuel assemblies in the DSC, which caused greater radiation levels. After learning the source of greater than expected radiation levels, the licensee continued loading the first cask. The personnel used lead shielding blankets to decrease the radiation levels and revised its procedures and its As-Low-As-Reasonably-Achievable (ALARA) plan to account for the differences in expected dose for future loadings. Also, the loading procedures were revised to not drain any water out

at this point in the loading process since this requirement was plant specific due to crane capacities and was not applicable to the Point Beach plant.

During preparations to load the second cask, the licensee discovered a defect in the neutron shield tank of the TC. The defect was a hole located between 340 and 350 degrees on the TC approximately three feet up from the bottom. As a result, water leaked out of the neutron water shield at a rate of approximately one drop per minute. The licensee suspected that the site of the leak might have been a previous weld repair. The licensee prepared an Operational Decision-Making Issue Evaluation Document to demonstrate adequacy of performing the weld repair with the DCS inside the TC. This approach required no heavy load lifts. The heat input from the welding process into the DSC did not effect the DSC. Subsequently, the damaged neutron shield was repaired under the vendor's guidance and the loading activities continued.

c. Conclusions

The inspectors determined that the licensee's loading and transfer of four casks to the ISFSI was adequate and met the requirements in the CoC and associated Technical Specifications.

1.2 Control of Heavy Loads (IP 60855, IP 60855.1)

a. Inspection Scope

The inspectors evaluated the crane and the auxiliary building structure to verify that they were able to sustain the loads associated with a fully loaded DSC and TC.

b. Observations and Findings

During the previous inspection, the inspectors determined and documented in Inspection Report No. 07200005/2004-002(DNMS), Section 2.4, that the Auxiliary Building structure was qualified to sustain a seismic event for a 101 ton lifted load based on an evaluation of the VSC-24 cask design. Subsequent to the previous inspection, the licensee changed to a new cask design with different weight specifications. The weight of the new NUHOMS-32PT cask exceeded the 101 ton value that was used to qualify the VSE-24 cask design. However, the licensee did not update the evaluation and the calculations reflecting the current cask weight prior to using the crane during the loading of the first NUHOMS-32PT cask. The inspector raised questions regarding the adequacy of the structural evaluations related to the building and the crane. In response, the licensee generated a condition report, CAP 60198, documenting this issue and stopped the use of the crane for heavy load lifts. Subsequently, the licensee performed an assessment of the current conditions associated with a 125-ton load on the crane and the auxiliary building combined with a seismic event. Using previous calculations as a reference and scaling techniques, the licensee concluded that the crane was qualified to lift a maximum load of 125 tons in the event of an earthquake. Following this assessment, the licensee resumed dry cask loading activities.

Section (2)(i), "Overall Requirements," of 10 CFR 72.122(b), states, in part, that structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes without impairing their capability to perform their intended design functions.

Contrary to the above, the licensee failed to demonstrate that the crane, a component important to safety, was designed to withstand the effects of an earthquake without impairing its capability to perform its intended function. Specifically, between October 25 and October 30, 2004, the licensee utilized the crane without performing a complete written evaluation to demonstrate that the crane was adequate to lift the weight of a fully loaded NUHOMS-32PT cask under a seismic event. This is considered to be a Non-Cited Violation of 10 CFR 72.122, Section (2)(i) (Violation 07200005/2004-003-01).

Upon further review, the inspectors identified other deficiencies in the structural analysis of the building and the crane. There was no response spectra analysis performed on the building to model its response due to an earthquake at different elevations, such as that of the crane. Also, the inspectors could not independently verify that the basis for the horizontal accelerations in all of the calculations used for the auxiliary building and the crane were adequate. In response to these questions, the licensee constructed a detailed building model of the steel portion of the auxiliary building. The preliminary results from performing modeling showed that the original acceleration values were conservative and adequate to demonstrate compliance with regulations and the ability of the building and the crane to sustain up to a 125-ton load under an earthquake scenario. In addition, the licensee hired an independent consultant who confirmed the licensee's results. The inspectors were not able to validate these conclusions since the appropriate documentation was not available at the time of the inspection and a complete analysis was not completed. However, the licensee committed to perform a full analysis of the auxiliary building and the crane response under a seismic event with the current plant conditions. The licensee stated that this analysis would be completed before the next dry cask loading campaign. The adequacy of the auxiliary building structure and the crane design basis during a seismic event will be the subject of further review by the inspectors and is considered an Unresolved Item (URI 07200005/2004-003-01).

c. Conclusions

The inspectors identified a Non-Cited Violation of 10 CFR 72.122 (b), Section (2)(i), "Overall Requirements." Specifically, between October 25 and October 30, 2004, the licensee utilized the crane without performing a complete written evaluation to demonstrate that the crane was adequate to lift the weight of a fully loaded NUHOMS-32PT cask under a seismic event. The violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, the inspectors identified one Unresolved Item associated with the licensee's analysis of the auxiliary building structure and the crane design basis during a seismic event.

2.0 Inspector Follow-up Items

(CLOSED) IFI 07200005/2004-002-01; Seismic analysis of the reactor auxiliary building structure

During the previous inspection, Inspection Report No. 07200005/2004-002(DNMS), Section 2.4, the inspectors determined that the auxiliary building structure was not evaluated to sustain new cask design loads under a seismic event. During this inspection, the inspectors discussed the issue with the licensee (Section 1.2 of this report). This issue will be tracked as an Unresolved Item and will be assigned a new tracking number; therefore, the inspector follow-up item is closed.

3.0 Exit Meeting

The inspectors presented the preliminary results of the inspections to the licensee on October 28, and November 19, 2004. On December 13, 2004, during a phone call, the inspectors presented the final inspection results to the licensee. The licensee acknowledged the findings presented and did not identify any documents or processes reviewed by the inspectors as proprietary.

Partial List of Persons Contacted

D. L. Koehl, Site Vice-President
J. Shaw, Plant Manager
R. Davenport, Production Planning Group Manager
S. Leblang, NMC High Level Waste Manager
J. Becka, Dry Fuel Storage Supervising Engineer
J. Connolly, Regulatory Affairs Manager
L. Schofield, Licensing

Inspection Procedures Used

IP 60855	Operation of Independent Spent Fuel Storage Installation
IP 60855.1	Operation of Independent Spent Fuel Storage Installation at Operating Plants

Items Opened, Closed, and Discussed

Opened

07200005/2004-003-01	NCV	Utilization of crane without a safety analysis performed on the crane and the auxiliary building in accordance with 72.122(b), Section (2)(i) under a seismic event.
07200005/2004-003-01	URI	Seismic analysis of the reactor auxiliary building structure.

Closed

07200005/2004-002-01	IFI	Seismic analysis of the reactor auxiliary building structure.
07200005/2004-003-01	NCV	Utilization of crane without a safety analysis performed on the crane and the auxiliary building in accordance with 72.122(b), Section (2)(i) under a seismic event.

Discussed

None

List of Acronyms Used

ALARA	As-Low-As-Reasonably-Achievable
CoC	Certificate of Compliance
CFR	Code of Federal Regulations
DSC	Dry Shielded Canister
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
SFP	Spent Fuel Pool
TC	Transfer Cask

Licensee Documents Reviewed

Calculation, “Structural Design Criteria for the Point Beach Nuclear Plant,” Revision 2, July 1967.

Bechtel Calculation, Book 43, “Auxiliary Building,” 1969.

Bechtel Calculation No. 10447-068-C-1, “Structural Adequacy of Auxiliary Building Superstructure,” Revision 0, dated February 13, 1995.

Calculation, “Bridge Crane Seismic Analysis TR833," Revision A, dated May 4, 1983.

Calculation, “Seismic Analysis of Steel Superstructure,” dated December 1978.

Calculation, white paper, “Assessment of PAB Crane,” dated October 30, 2004.

Condition Report, CAP 60198, “PAB Crane Adequacy Review,” dated October 28, 2004.

Condition Report, CAP 60291, “Dry Fuel-Transfer Cask Neutron Shield tank has leak,” dated November 11, 2004.

Condition Report, CAP 60127**, “**Higher Dose Rates than Expected during the Dry Fuel First Load,” dated October 26, 2004.