August 11, 2005

Mr. Dennis Koehl
Site Vice President
Point Beach Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 2 - REVISION TO SAFETY EVALUATION FOR AMENDMENT NO. 225 DATED JUNE 24, 2005

Dear Mr. Koehl:

On June 24, 2005, the Nuclear Regulatory Commission (NRC) issued Amendment No. 225 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit 2. The amendment incorporated a reactor vessel head drop accident analysis into the Final Safety Analysis Report, in response to your application dated April 29, 2005, as supplemented by letters dated May 13, May 19, June 1, June 4, June 9, June 20, and June 23, 2005. This letter transmits a revision to the NRC staff's safety evaluation (SE) associated with Amendment No. 225. The revision includes a clarification of the NRC staff's core damage frequency discussion and correction of administrative errors.

Enclosure 1 describes the revisions to the SE. The revised pages are included in Enclosure 2 and replace the associated pages in the original NRC staff's SE. The revisions do not change the conclusions of the original NRC staff's SE.

If there are any questions concerning this matter, please contact Harold K. Chernoff at (301) 415-4018.

Sincerely,

/RA/

Harold K. Chernoff, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-301

Enclosures: As stated

cc w/encls.: See next page

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POINT BEACH NUCLEAR PLANT, UNIT 2 CORRECTIONS TO THE NRC STAFF'S SAFETY EVALUATION FOR AMENDMENT NO. 225			
PAGE #	ORIGINAL TEXT	REVISED TEXT	REASON FOR THE CHANGE
3	10 CFR 50.59(c)(vi)	10 CFR 50.59(c)(2)(v)	Typographical error
5	5.6E-5	5.6E-5 per lift	Include proper units
5	5.6E-5/yr * 1.4E-1 = 7.8E-6/yr	(5.6E-5/lift) * (2 lifts / 1.5 yr) * (1.4E-1) = 1.05E-5/yr	Correctly reflect annualized RVH head drop probability
11	commitments 7 and 11	commitments 5 and 8	Reference correct commitments
11	Cooland	Coolant	Typographical error

Enclosure 1

ENCLOSURE 2: REVISED SE PAGES

equipment, (2) most equipment is protected by an intervening floor, (3) there is redundancy of components, and (4) crane failure probability is generally independent of safety-related systems. As is demonstrated by Oyster Creek's proposed activities, this conclusion may not always be valid.

Since the 1982 RVH drop analysis was completed based on a request from the NRC staff, 10 CFR 50.71(e) required that the results of the evaluation be incorporated into the FSAR. The failure to meet this regulatory requirement was brought to the licensee's attention by the NRC staff in April 2005. Subsequently, the licensee completed a 10 CFR 50.59, "Changes, tests, and experiments," review of the proposed incorporation of the 1982 RVH drop analysis into the FSAR. This review concluded that the proposed change to the FSAR required prior NRC approval in accordance with the requirements of 10 CFR 50.59(c)(2)(v).

In accordance with the requirements of 10 CFR 50.59, the licensee submitted a license amendment request (LAR) in accordance with the requirements of 10 CFR 50.90. In the June 20, 2005, letter the licensee stated that:

> NMC proposes changing the PBNP licensing basis to incorporate a revised RVH (heavy load) drop event analysis, specifically for PBNP Unit 2, within the scope of a revision that incorporates PBNP actions taken in response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980. The heavy loads analysis was performed based upon the guidance contained in NUREG-0612 as directed by an unnumbered NRC generic letter dated December 20, 1980, as supplemented by Generic Letter 81-07.

While the proposed inclusion of a RVH drop accident into the PBNP FSAR does meet the criterion of 10 CFR 50.59(c)(2)(v) and requires prior NRC approval pursuant to 10 CFR 50.90, the NUREG-0612, Phase I load handling measures and controls the licensee has committed to incorporate into the PBNP FSAR are not within the scope of this safety evaluation.

3.0 TECHNICAL EVALUATION

In summary the postulated RVH drop accident involves the concentric drop of the RVH onto the reactor vessel flange from a height of no more than 26.4 feet. The resultant impact displaces the reactor vessel downward. Downward movement of the reactor vessel creates the potential for damage to piping and tubing directly or indirectly connected to the reactor vessel, thereby creating the potential for a decrease in reactor coolant inventory. The following sections describe the NRC staff's technical evaluation of the licensee's analysis of this postulated accident.

3.1 Initiating Event

In NUREG-0612, the NRC provided guidelines to minimize the occurrence of the principal causes of load handling accidents and to control heavy load lifts 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. The defense-in-depth philosophy emphasized in these guidelines includes

While the licensee did not propose a conditional core damage probability (CCDP), using the assumptions of the licensee's accident analysis, the NRC staff estimated a CCDP to assess the risk implications. This CCDP estimate used the licensee's following assumptions: (1) a 300 gallon per minute loss-of-coolant accident (LOCA) from reactor vessel bottom mounted instrumentation (BMI) tube penetrations, (2) both RHR pump trains operable, and (3) both safety injection (SI) pump trains available but not operable. The CCDP estimate for the medium LOCA event, which best bounds the postulated PBNP conditions, was determined to be 1.4E-1 based on calculations using the SAPHIRE Code, Version 7.25 and the PBNP Standardized Plant Analysis Risk (SPAR) model, Version 3.11. It is noted that the SPAR model is developed for risk analysis of power operations, while the postulated RVH lift activities are conducted during refueling operations. Using the RVH drop probability of 5.6E-5 per lift and the CCDP estimate of 1.4E-1, the increased core damage frequency (CDF) from a lift was estimated to be (5.6E-5/lift) * (2 lifts / 1.5 yr) * (1.4E-1) = 1.05E-5/yr. When compared to the Regulatory Guide (RG) 1.174[1] risk acceptance guidelines, the RVH drop scenario falls on the threshold of no changes allowed and increased management attention.

Since the licensee did not submit the proposed LAR as a "risk-informed" submittal[2], the licensee's submittal did not address the elements of RG 1.174 to support the licensing basis change request. Based on the NRC staff evaluation, a reasonable risk assessment would show that the risk implications of a postulated RVH drop with an assumed probability of 5.6E-5 per lift would place the risk in the range of RG 1.174 risk acceptance guidelines where management attention is warranted. The acceptability of the licensee's submittal was primarily based on deterministic considerations.

3.2 <u>Mechanical and Structural Aspects of the Reactor Vessel Head Drop Accident</u>

The 1982 RVH drop analysis was limited to elastic behavior of the structures, piping, and components that are impacted. The licensee with support from Sargent & Lundy (S&L) and Westinghouse, determined that inelastic structure and piping behaviors would absorb significant energy such that there would be no structural or piping failure that would cause loss of core cooling.

S&L performed a finite element analysis (FEA) to evaluate the reactor vessel behavior during a postulated RVH drop scenario. Westinghouse performed a plastic analysis of the PBNP, Unit 2 reactor coolant main piping based on specified reactor vessel downward vertical displacements.

Enclosure 3 to the June 20, 2005, letter, contains the revised FEA of the postulated RVH drop scenario prepared by S&L, "Analysis of Postulated Reactor Head Load Drop Onto the Reactor Vessel Flange", Revision 1, dated June 19, 2005. Enclosure 4 to the June 20, 2005, letter, contains Westinghouse Report, "Plastic Analysis of Point Beach Reactor Coolant Piping for

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[1]RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002 (ML023240437).

[2]The licensee's letter dated April 29, 2005, stated that the analysis was based upon a plant specific risk-informed evaluation. In its letter dated June 20, 2005, the licensee removed references to a risk-informed evaluation.

Prior to moving the RVH over the vessel, the licensee will ensure that containment closure is established per commitments 5 and 8 in Section 4.0, herein. The postulated RVH drop does not result in pressurization of the containment building. Therefore, the licensee did not model a release through containment leakage. The NRC staff finds this acceptable based on guidance in Standard Review Plan (SRP) 15.7.4, "Radiological Consequences of Fuel Handling Accidents," for analyzing the FHA, which is consistent with Regulatory Position 5.1 of Appendix B to RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Revision 0, dated May 2003 (ML031490640).

Because the RVH drop may result in a loss of coolant through damaged lines connected to the reactor vessel (such as the BMI tubes), the ECCS provides core cooling. The fission products that are retained in the coolant and sump fluid are available for release to the outside environment by leakage from ECCS components outside of containment. The licensee evaluated the radiological consequences, both offsite and in the control room, of this release through ECCS leakage. The licensee used the PBNP FSAR LOCA ECCS leakage pathway dose analysis as the basis for the RVH drop dose analysis.

The licensee's source term was based upon the current licensing basis LOCA total core inventory in PBNP FSAR Table 14.3.5-1, adjusted for 30 days of radiological decay and a non-LOCA gap fraction of 0.08 for each iodine isotope available for release. The licensee's iodine gap fraction assumption is the same as the assumption in the PBNP current licensing basis FHA analysis for I-131, and bounds the assumption for the remaining iodine isotopes. Iodine is retained in the fluid circulating through the ECCS, while the remainder of the fission products are released and retained in the containment. This assumption is in accordance with guidance on the LOCA in SRP 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident," Appendix B. The licensee conservatively assumed all iodine released from the fuel is retained in the ECCS fluid.

Considering the similarity in release modeling, The licensee determined the radiological consequences of the RVH drop by determining the minimum ratio of the RVH drop iodine source term as discussed above to the PBNP FSAR LOCA source term. The LOCA ECCS leakage pathway dose results were adjusted by this scaling factor, by dividing the values by 75. The licensee's RVH drop dose results are listed in Table 3 of the licensee's June 20, 2005, letter.

The NRC staff had questions about the applicability of two of the PBNP FSAR LOCA analysis assumptions for the control room dose. The PBNP FSAR LOCA analysis does not bound the recent results of control room envelope unfiltered inleakage tracer gas testing, and the control room analysis assumed an ECCS leakage rate half that assumed for the offsite dose analysis. In response to these questions, the licensee evaluated the impact on the control room dose results of (1) increasing the assumed unfiltered inleakage from 10 cubic feet per minute (cfm) to 100 cfm to account for the testing results, and (2) increasing the ECCS leakage rate from 400 cubic centimeters per minute (cc/min) to 800 cc/min. The licensee showed that the control room dose would increase by a factor of 2.7, which is still bounded by the LOCA results and meet GDC-19 , "Control Room," dose criteria. The licensee's adjusted control room dose results are 3.8 rem thyroid and 0.0055 rem whole body. These are within the GDC-19 dose criteria of 5 rem whole body or its equivalent to any part of the body, given as 30 rem thyroid in