June 24, 2005

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

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT RE: INCORPORATION OF REACTOR VESSEL HEAD DROP ACCIDENT ANALYSIS INTO THE FINAL SAFETY ANALYSIS REPORT (TAC NO. MC6729)

Dear Mr. Koehl:

The Commission has issued the enclosed Amendment No. 225 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant (PBNP), Unit 2. The amendment changes Facility Operating License No. DPR-27 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. The April 29, 2005, application proposed amendments to PBNP Units 1 and 2, Facility Operating Licenses Nos. DPR-24 and DPR-27, respectively. However, your letter dated May 13, 2005, withdrew the proposed amendment for PBNP, Unit 1. This license amendment incorporates a PBNP, Unit 2 reactor vessel head drop accident into the PBNP Final Safety Analysis Report.

Your letter dated June 20, 2005, as supplemented by letter dated June 23, 2005, requested that this amendment be processed as an exigent amendment in accordance with the provisions of Title 10 *Code of Federal Regulations* 50.91(a)(6). The proposed amendment was noticed in the *Federal Register* on May 13, 2005 (70 FR 25621). Thus, the 30-day period for public comment had elapsed prior to your June 20, 2005, letter. Therefore, the NRC staff has determined that exigent circumstances do not exist and your request for exigent processing is denied. Furthermore, the NRC staff has determined that your discussion of exigent circumstances has not been substantiated.

As noted above, you submitted several supplements to your original application. Amendment applications that require multiple supplements present unnecessary challenges to licensees and the NRC staff. We understand that you will be reviewing your activities related to this licensing action. We are willing to discuss with you any lessons learned.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's *Federal Register* notice.

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: 1. Amendment No. 225 to DPR-27
2. Safety Evaluation
3. Notice of Issuance

cc w/encls: See next page

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's *Federal Register* notice.

Sincerely,

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Harold K. Chernoff, Sr. Project Manager, Section 1
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Point Beach Nuclear Plant, Units 1 and 2

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 225
License No. DPR-27

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated April 29, as supplemented by letters dated May 13, May 19, June 1, June 4, June 9, June 20, and June 23, 2005, 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, this license amendment authorizes changes to the design basis and Final Safety Analysis Report (FSAR) related to a postulated reactor vessel head drop accident. The licensee shall update the FSAR to reflect the revised design basis authorized by this amendment in accordance with 10 CFR 50.71(e). The description of a postulated reactor vessel head drop accident incorporated into the FSAR shall include discussion of: occurrences that lead to the initiating event; the event frequency classification; the sequence of events from initiation to the final stabilized condition; plant characteristics considered in the safety evaluation; assumed protective system actions; core and system performance; barrier performance; and radiological consequences.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance. Implementation of the amendment is the incorporation into the FSAR of the information related to a postulated reactor vessel head drop accident described in the licensee's application dated June 20, 2005, as supplemented by letter dated June 23, 2005 and as evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of issuance: June 24, 2005

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 225 TO FACILITY OPERATING LICENSE NO. DPR-27

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-301

1.0 INTRODUCTION

By 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, the Nuclear Management Company (NMC or the licensee), requested an amendment to Facility Operating License DPR-27 for Point Beach Nuclear Plant (PBNP), Unit 2[1]. The proposed license amendment would incorporate a Unit 2 reactor vessel head (RVH) drop accident into the PBNP Final Safety Analysis Report (FSAR).

2.0 REGULATORY EVALUATION

In a letter dated November 22, 1982, Wisconsin Electric Power Company[2] (WE or the licensee) submitted to the NRC the results of a RVH drop analysis. This letter stated that:

> The results of this analysis show that upon impact of the head drop the initial reactor vessel nozzle stresses are well within allowables. However, the loads imposed upon the reactor vessel supports caused by the impact of the head are greater than the critical buckling load of the support columns. These supports cannot be relied upon to absorb enough of the energy of impact to prevent severe damage to the safety injection lines attached to the reactor vessel or to the primary coolant loop piping.
>
> The results of the head drop analysis are presently being reviewed. This review is comprised of the following actions:
>
> 1) A review of the consequences of the head drop event for comparison with the guidelines of NUREG-0612, "Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," Section 5.1.

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[1]The April 29, 2005, application proposed amendments to PBNP Units 1 and 2, Facility Operating License Nos. DPR-24 and DPR-27, respectively. However, the letter dated May 13, 2005, withdrew the proposed amendment for PBNP, Unit 1.

[2]Prior to the formation of NMC, WE was the licensee of the PBNP.

2) An identification of alternative measures which may be used to remove decay heat from the core should normal methods of residual heat removal [RHR] become inoperative.

3) A determination of the probability of a head drop event based upon a lift frequency and current reactor operating history.

4) A determination of any potential modifications which could be made to limit the probability of occurrences of a head drop event.

5) A detailed review of the containment polar crane to determine areas of potential single failure that could be upgraded to provide increased reliability.

> It is anticipated that the review process will be concluded within our originally proposed time frame for NUREG-0612 compliance, that is, January 1984. However, it is unlikely that equipment modifications could be accomplished within this time frame. Should they be needed, such modifications would be completed as expeditiously as possible.

The licensee performed this analysis in response to a request from the NRC as detailed in an NRC letter dated December 20, 1980, Generic Letter (GL) 80-113. Although this analysis presented results that did not meet the acceptance criteria of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated July 1980, the licensee’s submittal identified potential courses of action that were being taken to comply with the criteria of NUREG-0612. With the exception of Items 3 and 5, the licensee was not able to provide records showing that these actions were completed prior to April 2005.

In a letter dated April 12, 2005, the licensee discussed its understanding of the licensing basis associated with the 1982 RVH drop analysis. In subsequent discussions with the licensee, the NRC staff informed the licensee that its April 12, 2005, letter did not properly characterize the PBNP licensing basis related to the 1982 RVH drop analysis. As stated in GL 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-related Equipment”:

> ... the [NRC] staff is concerned that other licensees may believe that their heavy load operations are in compliance with the regulations because they have completed Phase I of the generic letter of December 22, 1980, and the closeout of Phase II by GL 85-11. GL 85-11 did not relieve licensees of their responsibility under 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.90 to evaluate new activities with respect to the SAR [safety analysis report] and the Technical Specifications to determine whether the activity involves an unreviewed safety question or a change in the Technical Specifications. In addition GL 85-11 concluded that the risks associated with damage to safety-related systems are relatively small because (1) nearly all load paths avoid this

> 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)(vi).

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)(vi) 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

measures for reducing the likelihood of dropping heavy loads including design, operation, training, maintenance of cranes and lifting devices and increased handling system reliability.

The licensee has taken several measures to enhance the reliability of the handling system in preventing a load drop. The NRC staff requested that the licensee describe measures to minimize the potential for “two-blocking” as defined in NUREG-0612. In the supplemental information provided in an attachment to the letter dated May 19, 2005, the licensee described that the main hoist of the Unit 2 polar crane is equipped with two independent upper travel limit switches to prevent the possibility of a “two-blocking” incident. The two independent upper travel limit devices are of different design and are activated by independent mechanical means. These devices independently de-energize either the hoist drive motor or the main power supply. The redundant limit switches have been set to ensure that sufficient margin exists between the actuation of these switches and physical contact of the upper and lower blocks. This design feature satisfies the design criteria in NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” with regard to protection from potential “two-blocking” events.

The licensee described that the counterweight-activated limit switch is functionally tested in accordance with routine maintenance procedure RMP 9118-1, “Containment Building Crane OSHA Operability Inspections,” prior to use if the crane has been idle greater than six months. The licensee also described that the gear-actuated upper limit switch is tested and polar crane controls are checked daily when the containment crane is in use in accordance with procedure RMP 9118-1. Prior to lifting the head, procedure 2RMP 9096, “Reactor Vessel Head Removal and Installation,” requires a pre-lift inspection to be performed that includes a functional check of the main hoist gear-actuated upper limit switch. The licensee stated that operational restrictions have been added to procedure 2RMP 9096 that limit the height of the bottom of the RVH flange to prevent actuation of the upper travel limit switches. The restriction will maintain the lift approximately 3.5 inches below the actuation setpoint of the first upper travel limit switch. This will be accomplished by using physical references and visual level checks during the lift.

The NRC staff found that the two independent limit switch designs, combined with the crane testing and operational restrictions specified in procedures, provides assurance that the potential for a “two-blocking” incident is negligible.

The licensee discussed the likelihood of an RVH drop while the RVH is suspended over the reactor vessel. The licensee proposed that the PBNP plant-specific probability of an RVH drop is less than the upper bound estimate of 5.6E-5 per crane lift as provided in NUREG-1774, “A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1998 through 2002.” This proposal was based on a plant-specific review of PBNP operating experience data and comparison to the generic data included in three areas of NUREG-1774: (a) Very Heavy Load drop probability, (b) load slip probability, and (c) human error probability per lift.

The licensee’s plant-specific probability estimate of 5.6E-5 for an RVH drop is based on recent operating experience at commercial nuclear power plants with loads over 30 tons. This value was calculated by considering 3 load drops out of an estimated 54,000 crane lifts of very heavy loads, as described in the NUREG-1774 event database contained in its Appendix A. None of the 3 load drops were of an RVH. The NRC staff concludes that this probability value is a reasonable best-estimate probability for an RVH drop.

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. Annualizing the RVH drop probability of 5.6E-5 per lift and using the CCDP estimate of 1.4E-1, the increased core damage frequency (CDF) from a lift was estimated to be 5.6E-5/yr x 1.4E-1 = 7.8E-6/yr. When compared to the Regulatory Guide (RG) 1.174[3] risk acceptance guidelines, the RVH drop scenario falls in the range of increased management attention.

Since the licensee did not submit the proposed LAR as a "risk-informed" submittal[4], 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 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 Mechanical and Structural Aspects of the Reactor Vessel Head Drop Accident

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,

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[3]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).

[4]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.

contains Westinghouse Report, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop", Revision 1, dated June 20, 2005 (Proprietary). The NRC staff reviewed the evaluations included in the S&L and Westinghouse reports. The NRC staff's evaluation of these two reports is described below.

Sargent & Lundy Finite Element Analysis

The licensee stated that the S&L analysis considers a flat vertical impact of the new RVH, which weighs 194,000 pounds, dropping from a height of 26.4 feet onto the reactor vessel flange. This analysis also includes an evaluation of the structural integrity of supporting elements in the load path, and predicts the vertical downward displacement of the reactor vessel.

The licensee also stated that the load path consists of the reactor vessel, reactor vessel supports at the four nozzles and two brackets, the support girder box frame, and the six pipe columns and their supports, which rest on the concrete foundation. The reactor coolant system (RCS) piping provides additional stiffness to the reactor vessel nozzles under vertical impact loading, and also transfers a portion of the impact load to the steam generator (SG) and the reactor coolant pump (RCP) support structures under a postulated RVH scenario. The concrete shelf is not considered in the vertical load path in this analysis. Thus, the shear capacity of the concrete and the embedded rebars are not credited with absorbing any portion of the impact energy to reduce the impact loads on the reactor vessel nozzles and the RCS piping in the event that the reactor vessel displacement exceeded 3.375 inches. The NRC staff finds the licensee's description of the impact load path acceptable.

The analysis models used in the S&L analysis are static analysis models for stiffness calculations of various components and substructures, and a dynamic impact model. The finite element analyses are performed using the ANSYS computer code.

The static analysis models include:

(1) A detailed model of reactor vessel flange and reactor vessel shell below the flange, including a nozzle resting on a supporting shoe.

(2) A similar detailed model of reactor vessel flange and Reactor vessel shell below the flange with a support bracket resting on a supporting shoe.

(3) A detailed model of the hexagonal girder box frame supported by six pipe columns at the vertices.

(4) Piping models for the RCS hot legs and cold legs.

These models are used to construct static load-displacement diagrams for all steel components that are within the impact load path. Static vertical displacement is applied to the components uniformly and a reaction force is calculated to construct the force-displacement diagram of the affected components. In the static analysis, non-linear material properties are modeled with a strength increase factor of 10 percent to account for the strain rate effects due to the dynamic impact. The large deformation analysis option was selected to account for potential buckling and yielding in the structural components along the impact load path.

The results of the static analysis are used as part of the input for dynamic analysis. In calculating the stiffness of RCS hot leg or cold leg, two bounding cases are analyzed. In the first case, fixed boundary condition is used at either the SG location or the RCP location; and in the second case, pinned boundary condition is used at the SG location or the RCP location. In both cases, the pipe axial movement is released to account for the potential horizontal movement of the SG or the RCP.

The dynamic impact model consists of a two-mass model with springs and dash-pot in a vertical configuration. The top mass represents the falling head, and the bottom mass represents the target reactor vessel model supported by various springs, which represent the stiffness of the nozzle/bracket support, the girder box frame/column supports and the RCS piping.

In the dynamic impact analysis, the licensee stated that an impact damping of 5 percent of the critical damping is used. This damping assumption is judged by the licensee to be reasonable for this application in consideration of: (1) energy loss due to plastic damage at the impact surface between the RVH and the reactor vessel flange; (2) energy loss due to imparted damage to six lateral supports for the hexagonal girder box frame; and (3) energy loss due to local damage to the liner and concrete crushing at the top of the six support columns. The NRC staff finds the licensee's assumptions and approximations in the static analysis and the dynamic impact models reasonable and acceptable for the PBNP RVH drop scenario.

Results of the dynamic transient analysis indicate that the maximum dynamic downward displacement of the reactor vessel is 2.72 and 3.20 inches for cases 1 and 2, respectively. The licensee indicated that these displacements are less than 3.375 inches and, therefore, the hexagonal girder box frame will not come in contact with the concrete shelf, which is consistent with the assumption made in the dynamic impact model.

By letter dated June 23, 2005, the licensee submitted an evaluation of RCS piping nozzles for the postulated RVH drop scenario. In the analysis, the reaction forces and moments from the RCS piping at the reactor vessel nozzles are extracted for a deflection of 3.2 inches from the force-deflection analyses. The RCS piping boundary loads are obtained from the bounding cold leg fixed-fixed model at 3.2 inches reactor vessel deflection and are used in the reactor vessel nozzle stress evaluation. The licensee stated that the result of this calculation indicate that the maximum Von Mises stress in the nozzle due to membrane plus bending is less than the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Section III, Appendix F allowable stress for membrane stress intensity of 0.7 Su. Similarly, the Von Mises stress in the reactor vessel support brackets is also less than 0.7 Su. The NRC staff finds the licensee's calculated results reasonable and acceptable.

The S&L analysis also evaluated the maximum impact load on the column foundation, and the capability of the concrete shelf to provide lateral support for the stability of the support columns located within the shelf, and found the results to be acceptable. The NRC staff finds the S&L's conclusions reasonable and acceptable.

<u>Westinghouse Plastic Analysis of Reactor Coolant Loop Piping</u>

The licensee provided an evaluation of the impact of a postulated RVH drop on the structural integrity of reactor coolant loop piping in proprietary Westinghouse Calculation CN-RCDA-05-68, Rev. 1, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop." The evaluation consisted of a plastic analysis of the PBNP, Unit 2 reactor coolant loop piping for a downward vertical displacement of the reactor vessel nozzles. Two displacements were analyzed: (1) a 4-inch displacement, which bounds the displacement calculated from the S&L model, and (2) a 6.5-inch displacement, which represents the maximum possible displacement of the reactor vessel nozzles before the RCS piping comes in contact with the biological shield wall.

The results of the analysis were compared to the criteria specified in the 1998 Edition of ASME Code, Section III, Appendix F, paragraph F-1340. The Appendix F criteria allow for large reactor coolant loop piping deformations. However, the criteria are intended to assure that violation of the reactor coolant loop piping pressure boundary does not occur. The Appendix F criteria contain limits on general primary membrane stress intensity (0.7 Su), maximum primary stress intensity (0.9 Su) and average primary shear stress (0.42 Su) for a section loaded in pure shear stress. The general primary membrane stress intensity limit applies to the average value across the thickness of a section, whereas the maximum primary stress intensity limit applies to the highest value across the thickness of a section. The NRC staff finds the ASME Code Appendix F plastic analysis allowable limits appropriate for this evaluation.

Westinghouse used a relatively simple ANSYS finite element model of the hot and cold legs. The hot and cold legs were fixed at the reactor vessel nozzles and the SG, and the RCP nozzles respectively. Each leg was modeled as a straight run of piping with one elbow. The hot and cold leg material properties were represented by a piecewise linear stress strain curve. Westinghouse used two sets of material properties to represent the upper bound and lower bound properties of the piping and elbow materials. The NRC staff considers the Westinghouse model adequate for calculating the general primary membrane stress intensities in the piping since the general primary membrane stress intensity represents an average value through the thickness and over a finite cross sectional area of the piping. Westinghouse argues that the stress intensities from the ANSYS model should be compared to the ASME Code, Appendix F limit maximum primary stress intensity because calculated stress includes local through-wall bending effects in the elbow. While the NRC staff agrees that the reported stresses from the ANSYS finite element model contain local effects due to elbow flexure, the Westinghouse model is not sufficiently refined to accurately predict the maximum primary stress intensity. Therefore, the NRC staff concludes that it is appropriate to compare the maximum stress intensity from the ANSYS finite element model to the Appendix F general primary membrane stress limit. The NRC staff also concludes that comparing the calculated stress intensity from the ANSYS finite element model to the Appendix F general primary membrane stress limit will provide adequate assurance that the Appendix F maximum primary stress limit is also satisfied. The NRC staff agrees with Westinghouse's argument that the Appendix F shear stress criteria are not relevant for this analysis since the loading is primarily controlled by bending of the reactor coolant loop piping.

The results of the analyses indicate that the maximum calculated stress intensity in the hot and cold leg piping is within the ASME Code, Appendix F limit of 0.7 Su for general primary membrane stress for the 4-inch reactor vessel nozzle displacement. Since the 4-inch reactor vessel nozzle displacement bounds the maximum calculated vessel displacement predicted from the S&L model, the NRC staff finds that there is reasonable assurance that pressure boundary integrity of the reactor coolant loop piping will be maintained in the event of a postulated RVH drop.

The results of the analyses also indicate that the 0.7 Su limit is exceeded in the cold leg for the 6.5-inch vessel nozzle displacement. The maximum stress intensity was calculated in the cold leg elbow. Although the maximum calculated stress intensity exceeds the ASME Code general primary membrane stress intensity limit, the NRC staff concludes that loss of reactor coolant loop piping pressure boundary integrity would not be expected even if the vessel nozzle displaced 6.5 inches, because the maximum calculated stress intensity is still well below the material ultimate strength.

Bottom Mounted Instrument (BMI) Tubes

As a result of the predicted maximum dynamic downward displacement of 3.2 inches for the reactor vessel, and recognizing the potential impact between the BMI tubes and the floor with clearance varying from 1 inch to 4.5 inches, the licensee assumed that all 36 BMI tubes are severed. Therefore, the structural integrity of the BMI tubes is not considered in this analysis.

Based on the results of the S&L and Westinghouse analyses, the NRC staff concurs with the licensee’s determination that concludes that there is reasonable assurance that the pressure boundary integrity of the RCS piping for PBNP, Unit 2 will be maintained in the event of a postulated RVH drop from a height of 26.4 feet above the reactor vessel flange.

3.3 Protective System Actions

In its letter dated June 20, 2005, the licensee postulated failure of all 36 BMI tubes and determined that RCS leakage due to gravity drain was well within the makeup capability of a single RHR or SI pump. (Each RHR pump is stated to have a design capacity of 1560 gpm at a design head of 280 ft, and an SI pump is stated to have a design capacity of 700 gpm at a design head of 2600 ft. In its June 20, 2005, letter, the licensee stated that the loss rate due to complete failure of all 36 BMI tubes is approximately 300 gpm. The NRC staff concludes that the boiloff rate is minor compared to the loss due to complete failure of all 36 BMI tubes and well within the capacity of a single RHR or SI pump.

The equipment that the licensee has committed to maintain operable or available (both trains of RHR and SI, respectively), and the senior reactor operator initially stationed inside containment, will reasonably ensure continued provision of water into the reactor vessel as long as at least one RCS pipe remains connected to the reactor vessel with capability to transport injection water into the reactor vessel.

In its June 20, 2005 application, the licensee predicts, based upon curves provided in SEP-1, "Degraded RHR System Capability," that the time to boil, 65 hours after shutdown, with RCS level at reduced inventory and starting at 140 degrees F, is approximately 16 minutes. With the RCS level at one foot below the flange, which is the procedural requirement for lifting and setting the RVH and the same conditions as above, the time to boil is just over 20 minutes. Based on simulator validation of steps in SEP 2.3, the time to inject water into the RCS using SI pumps is approximately 10 minutes. The licensee stated that adequate time is available to diagnose the problem, enter the associated shutdown emergency procedure, and establish makeup flow following a postulated RVH drop that results in RCS leakage.

The NRC staff reviewed a procedure identified in the licensee's June 20, 2005, letter, SEP-2.3, "Cold Shutdown LOCA," which is initiated based upon indication of a loss of reactor coolant. The procedural guidance within SEP-2.3 will attempt to restore core cooling/level using the charging and SI systems. When inventory is depleted, containment sump recirculation can be aligned using RHR.

SEP-2.3 will direct checking whether charging or SI flow is adequate to stabilize or restore RCS inventory. Upon exhaustion of the refueling water storage tank inventory (the licensee has committed to maintain a minimum borated water volume of 197,000 gallons to be available as a suction source), RHR pumps will be realigned to take a suction from the containment sump to ensure long-term cooling. In this regard, the licensee has committed to the containment sump screen being installed and the flowpath for aligning RHR pump suction to the containment sump being available during movement of the RVH.

The NRC staff concludes that SEP-2.3 is adequate to reasonably ensure provision of water to the RCS and reactor vessel should makeup be needed. The NRC staff finds that the licensee has reasonably ensured that makeup will be provided if needed and that the licensee has reasonably ensured that a core damage accident will not occur due to a postulated RVH drop.

In addition to the provision of in-depth makeup capability, the licensee has ensured that containment is closed, the purge supply and exhaust fans are off, the associated containment isolation valves closed, and the personnel airlock door interlocks are functional (ensuring one door at each airlock is closed) prior to movement of the RVH. These measures provide reasonable assurance of adequate core cooling and limit the release of radioactive material.

3.4 Radiological Consequences

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of the postulated RVH drop, performed by the licensee in support of its proposed license amendment, reviewing the assumptions, inputs, and methods used by The licensee to assess these impacts. The NRC staff also performed independent calculations to confirm the conservatism of the licensee's analyses.

The licensee's analysis of the radiological consequences of the RVH drop assumes that, although some loss of coolant may occur as a result of the RVH drop, cooling of the core can be maintained through use of the emergency core cooling systems (ECCS), so that no fuel melting occurs. Potential mechanical damage to the fuel clad may occur as a result of the RVH drop, with a subsequent release of the fission products from the fuel gas gap to the reactor coolant. This release mechanism can be considered to be similar to the release mechanism in an FHA. To provide a bounding radiological source term, the licensee assumed the RVH drop results in clad damage to 100 percent of the fuel assemblies in the core, such that a complete gap release occurs.

Prior to moving the RVH over the vessel, the licensee will ensure that containment closure is established per commitments 7 and 11 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-Cooland 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 SRP 6.4, "Control Room Habitability System." The licensee's dose results at the exclusion area boundary (EAB) are 0.8 rem thyroid and 0.003 rem whole body, and at the low population zone (LPZ) are 0.5 rem thyroid and 0.0008 rem whole body. The offsite dose results are well within the dose criteria in 10 CFR Part 100.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of an RVH drop at PBNP, Unit 2. The NRC staff finds that NMC used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the NRC staff finds the licensee has shown with reasonable assurance that the radiological consequences of the postulated RVH drop are acceptable.

4.0 REGULATORY COMMITMENTS

The following commitments were contained in the licensee's letter dated June 20, 2005:

1. The Site Vice President has directed an administrative hold on RVH lift activities until the associated issues are resolved.

2. A Senior Reactor Operator will be stationed in containment during RVH lift activities and will have communications capability with the control room.

3. The containment sump screen shall be installed and the flowpath for aligning RHR pump suction to the containment sump is available.

4. A minimum borated water volume of 197,000 gallons shall be available as a suction source.

5. Containment purge supply and exhaust fans are off and associated containment isolation valves are closed when the RVH is suspended greater than 24 inches over the reactor vessel flange.

6. The maximum allowable lift height for the RVH (i.e., 26.4 feet above the reactor vessel flange when over the fuel) shall not be exceeded.

7. The Programmed and Remote (P&R) reactor vessel inservice inspection device will not be lifted over a core containing fuel assemblies.

8. Both personnel airlock door interlocks will be functional (ensuring one door at each airlock is closed).

9. Both SI trains shall be available.

10. Both RHR trains shall be operable.

11. Technical Specification Limiting Condition for Operation (LCO) 3.7.9, "Control Room Emergency Filtration System (CREFS)," and LCO 3.3.5, "CREFS Actuation Instrumentation," shall be met.

12. One standby emergency power source capable of supplying each 4.16 kV/480 V Class 1E safeguards bus on PBNP, Unit 2 shall be operable.

13. The licensee will incorporate an analysis of the RVH drop into the PBNP FSAR.

14. The licensee will incorporate the PBNP method of NUREG-0612 Phase I compliance into the PBNP FSAR.

The above compensatory measures have been entered as regulatory commitments in the licensee's Commitment Management System, which complies with Nuclear Energy Institute's Document 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes." The NRC staff has reviewed the compensatory measures and how they will be controlled, and finds that the licensee's commitments provide adequate assurance that safe plant operation will not be affected by movement of the RVH inside containment.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulation at 10 CFR 50.92(c) states that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the 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) result in a significant reduction in a margin of safety. The NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91. The NRC staff's final determination is presented below:

1. Would the proposed amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The proposed change incorporates a postulated RVH drop accident into the FSAR. This postulated accident involves the drop of the RVH over a reactor vessel containing fuel assemblies. Assuming that the BMI tubes are severed as a result of displacement of the reactor vessel, a decrease in reactor coolant inventory will occur. Thus, a RVH drop accident can be considered as a LOCA under shutdown conditions.

The RVH drop accident meets the frequency classification[5] of an infrequent incident (I. e., an incident that may occur during the lifetime of the plant. Therefore, it does not represent a significant increase in the probability of a LOCA, above the current licensing basis large break LOCA.

Prior to commencement of RVH movement, at an elevation greater than 24 inches over a reactor vessel containing fuel assemblies, containment closure will be established. Redundant trains of the safety-related RHR system will be operable, with makeup water capacity well in excess of the postulated LOCA. The calculated radiological consequences of the RVH drop accident are well within those calculated for the current licensing basis large break LOCA. Therefore, the consequences of a LOCA are not increased. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

2. Would the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change incorporates a postulated RVH drop accident into the FSAR. This postulated accident involves the drop of the RVH over a reactor vessel containing fuel assemblies. Assuming that the BMI tubes are severed as a result of displacement of the reactor vessel, a decrease in reactor coolant inventory will occur. Thus, a RVH drop accident can be considered as a LOCA under shutdown conditions.

As described in the response to Question 1, the frequency and consequences of the RVH drop accident are comparable to or within those of the current licensing basis large break LOCA. Therefore, while the RVH drop accident assumes a unique initiating event and operating conditions, it does not represent a new or different kind of accident from any accident previously evaluated,

3. Would the proposed amendment result in a significant reduction in a margin of safety?

Response: No

The proposed change incorporates a postulated RVH drop accident into the FSAR. This postulated accident involves the drop of the RVH over a reactor vessel containing fuel assemblies. Assuming that the BMI tubes are severed as a result of displacement of the reactor vessel, a decrease in reactor coolant inventory will occur. Thus, a RVH drop accident can be considered as a LOCA under shutdown conditions.

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[5]Chapter 15 of RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3 (ML011340116), provides guidance on frequency classifications for accident analyses.

As described in the response to Question 1, the frequency and consequences of the RVH drop accident are comparable to or within those of the current licensing basis large break LOCA.

The proposed change adds an accident analysis to the FSAR, it does not alter any safety limits, limiting safety system settings or limiting conditions for operation as defined in the Technical Specifications. Therefore, the proposed amendment does not result in a significant reduction in a margin of safety.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the 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 amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has concluded that the amendment involves no significant hazards consideration as addressed in Section 5.0, herein. Accordingly, these amendments meet 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 these amendments.

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 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.

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7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-301

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 225 to Facility Operating License No. DPR-27 issued to Nuclear Management Company, LLC (the licensee), which modified the Point Beach Nuclear Plant (PBNP), Unit 2, Final Safety Analysis Report to include a reactor vessel head drop accident for operation of the PBNP, Unit 2, located in Two Rivers, WI. The amendment is effective as of the date of issuance.

The amendment authorized changes to the design basis and Final Safety Analysis Report (FSAR) related to a postulated reactor vessel head drop accident in accordance with 10 CFR 50.71(e).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on May 13, 2005 (70 FR 25621).

For further details with respect to this action see (1) the application for amendment 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, (2) Amendment No. 225 to License No. DPR-301, and (3) the Commission's related Safety Evaluation dated June 24, 2005. The Commission made a final no significant hazards consideration determination in its Safety Evaluation dated June 24, 2005. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, Public File Area O1 F21,11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/NRC/ADAMS/index.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC Public Document Room Reference staff by telephone at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 24th day of June 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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