NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

June 20, 2005

NRC 2005-0079
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR 27

Supplement 4 to Request for Exigent Review of Heavy Load Analysis

References:
1. NMC Letter to NRC Dated April 29, 2005
2. NMC Letter to NRC Dated May 13, 2005
3. NMC Letter to NRC Dated May 19, 2005
4. NMC Letter to NRC Dated June 1, 2005
5. NMC Letter to NRC Dated June 4, 2005
6. NMC Letter to NRC Dated June 9, 2005

In Reference 1, Nuclear Management Company, LLC (NMC), requested review and approval, in accordance with the provisions of 10 CFR 50.90 and 50.91(a)(6), of a proposed amendment to the licenses for Point Beach Nuclear Plant (PBNP), Units 1 and 2, to support a change to the PBNP Final Safety Analysis Report (FSAR) regarding control of heavy loads. The review for PBNP Unit 2 was requested on an exigent basis.

References 2, 3 and 6 submitted supplements to the proposed amendment to provide the results of additional assessments and to incorporate additional technical justification for the proposed amendment. Additionally, Reference 2 retracted the proposed amendment for PBNP Unit 1 and proposed to apply the reactor vessel head (RVH) lift assessment for the upcoming lift of the Unit 2 RVH. References 4 and 5 provided responses to questions posed by Nuclear Regulatory Commission (NRC) staff.

To facilitate NRC staff review of the proposed amendment, this supplement compiles and replaces the original request, transmitted in References 1 through 6, in its entirety. Previously submitted reference material remains applicable.

Enclosed for Commission review and approval are the revised PBNP analyses for control of heavy loads associated with the planned lift of the Unit 2 RVH. Enclosure 1

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provides a description, justification, and a significant hazards determination for the RVH drop event analysis. Enclosure 2 describes administrative controls established during the Unit 2 RVH lift. Enclosure 3 submits revised finite element analysis (FEA) of the postulated RVH drop scenario prepared by Sargent & Lundy, "Analysis of Postulated Reactor Head Load Drop Onto the Reactor Vessel Flange", Revision 1, dated June 19, 2005. Enclosure 4 submits Westinghouse Report, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop", Revision 1, dated June 20, 2005 (Proprietary).

Also provided in Enclosure 4 is a Westinghouse authorization letter, accompanying affidavit, Proprietary Information Notice and Copyright Notice for the analysis provided in Enclosure 4. The specified pages (marked "Proprietary") of the calculations contained in Enclosure 3 were previously provided in Reference 6 and are supported by the authorization letter and affidavit submitted therein.

Since the document pages listed above as Proprietary contain information proprietary to Westinghouse Electric Company, they are supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity, for each, the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the above documents, or the supporting Westinghouse affidavit, should reference the appropriate authorization letter (CAW-05-2015) and be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Summary of Commitments:

This letter replaces commitments made in References 1 through 6 with the following:

1. Prior to Unit 2 RVH lifting activities associated with the spring 2005 refueling outage, NMC will establish administrative controls for the RVH lift as described in Enclosure 2.

2. NMC will incorporate an analysis of the RVH drop into the PBNP FSAR.

3. NMC will incorporate the PBNP method of NUREG-0612 Phase I compliance into the PBNP FSAR.

In accordance with 10 CFR 50.91, a copy of this submittal, with attachments, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 20, 2005.

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures (4)

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

Westinghouse

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Our ref: CAW-05-2015

June 20, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: CN-RCDA-05-68, Rev. 1, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced letter and its attachment is further identified in Affidavit CAW-05-2015 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2015, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. S. Galembush, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. S. Galembush, Supervisory Engineer
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 20th day
of June, 2005

Sharon L Fiori
Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Supervisory Engineer, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked CN-RCDA-05-68, Rev. 1, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop" (Proprietary), as an attachment to WEP-05-193, Rev. 1, dated June 20, 2005. The information is provided in support of a submittal to the Commission, being transmitted by Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Point Beach Unit 2 contains design information that is proprietary to Westinghouse and is provided in response to certain NRC requirements for justification of reactor vessel head drop analyses.

This information is part of that which will enable Westinghouse to:

(a) Show the impacts upon the reactor coolant main loop piping from a downward vertical displacement of up to 6.5 inches at the reactor vessel nozzles. This postulated displacement is conjectured to occur due to a drop of the reactor vessel head onto the reactor vessel during installation or removal.

(b) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of this information to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Nuclear Management Company

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC:

Enclosed are 5 copies of Calculation Note Number CN-RCDA-05-68, Rev. 1, "Plastic Analysis of Point Beach Reactor Coolant Piping for Reactor Vessel Head Drop" (Proprietary)

Non-Proprietary copies of the Calc Note are not provided.

Also enclosed is Westinghouse authorization letter CAW-05-2015 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Calculation Note Number CN-RCDA-05-68, Rev. 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's' regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-05-2015 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

ENCLOSURE 1

SUPPLEMENT 4 TO REQUEST FOR EXIGENT REVIEW OF HEAVY LOAD ANALYSIS

1.0 DESCRIPTION OF PROPOSED CHANGE

In accordance with 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests exigent review and approval of a revised analysis regarding the control of heavy loads at Point Beach Nuclear Plant (PBNP) Unit 2 necessary to reflect the reactor vessel head (RVH) drop event in the PBNP current licensing basis. Approval of this change to the licensing basis is requested for use for Unit 2 RVH lifting activities.

PBNP Unit 2 has completed routine refueling and is currently shutdown with the old RVH removed. The lifting and setting of the new RVH are awaiting completion of associated analyses. Consequently, exigent NRC approval of the proposed amendment is requested to preclude delays in resumption of PBNP Unit 2 power operation.

To facilitate the staff's review, the following description of references to previous submittals is provided. All six references are superceded by this Supplement.

References:

1. NMC Letter to NRC Dated April 29, 2005 (NRC 2005-0055)
   Original amendment request.

2. NMC Letter to NRC Dated May 13, 2005 (NRC 2005-0063)
   Supplement 1.

3. NMC Letter to NRC Dated May 19, 2005 (NRC 2005-0064)
   Supplement 2 and Response to Request for Additional Information.

4. NMC Letter to NRC Dated June 1, 2005 (NRC 2005-0067)
   Response to Request for Additional Information.

5. NMC Letter to NRC Dated June 4, 2005 (NRC 2005-0070)
   Response to Request for Additional Information.

6. NMC Letter to NRC Dated June 9, 2005 (NRC 2005-0072)
   Supplement 3 and Response to Request for Additional Information.

2.0 PROPOSED CHANGE

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[1], as supplemented by Generic Letter 81-07.

The heavy load analysis is based on a plant specific evaluation that continues to demonstrate the low probability of occurrence of a RVH drop as originally evaluated. The analysis is supplemented by a finite element analysis (FEA) to demonstrate the affects on piping attached to the reactor vessel following a postulated RVH drop scenario. For the specific case of a RVH lift, this analysis also requires that administrative controls be maintained during lifting of the RVH over the reactor vessel containing fuel assemblies in order to maintain defense-in-depth.

Administrative controls will be in effect whenever the RVH is not fully resting on the reactor vessel flange and any part of the head is over a reactor vessel containing fuel assemblies.

The administrative controls, which address equipment requirements for potential accident mitigation, are contained in Enclosure 2 to this letter.

3.0 BACKGROUND

On December 22, 1980, NRC issued GL 80-113, which was supplemented on February 3, 1981, (Generic Letter 81-07) regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." These Generic Letters discussed a two-phase set of investigations and submittals by licensees. Phase I was to identify the load handling equipment within the scope of NUREG-0612 and to describe the associated load paths, procedures, operator training, special and general purpose lifting devices; the maintenance, testing and repair of equipment; and handling equipment specifications. Phase II was intended to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment had been provided.

Wisconsin Electric Power Company (WEPCo), then licensee for PBNP, responded to these letters via submittals dated June 19 and September 30, 1981; January 11, February 25, June 16, June 30, July 23, September 16, October 22, and November 22, 1982; and February 15 and September 28, 1983.

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[1] The December 22, 1980, Generic Letter was issued as an unnumbered document but was later numbered as GL 80-113 and hereinafter will be referred to as GL 80-113.

Additional information was provided in WEPCo's response to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment" (Wisconsin Electric Power Company Letter Dated May 9, 1996).

The November 22, 1982, letter provided the results of the PBNP RVH drop analysis as follows:

> "The results of this analysis showed that upon impact of a head drop, the initial reactor vessel nozzle stresses would be well within allowables. However, the loads of the head impact 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 1982 analysis conservatively limited itself to elastic behavior of the structures that are impacted. No attempt was made to reflect inelastic material behavior, which was later determined to absorb significant energy such that there would be no structural failure that would cause loss of core cooling.

On March 27, 1984, the NRC issued a safety evaluation that addressed the Phase I actions taken by WEPCo. The safety evaluation permitted deferral of the annual inspections of the containment polar cranes if the applicable inspections (e.g., daily, monthly, and annual) were performed prior to use. Additionally, it required interim Technical Specifications to restrict movement of heavy loads over spent fuel until a single-failure-proof crane was installed. Interim Technical Specifications for restrictions on the movement of heavy loads over the spent fuel pool were proposed by WEPCo via a submittal to the NRC dated March 16, 1984, as modified September 25, 1984 (Technical Specification Change Request 104). In response, the NRC issued Facility Operating License Amendments 91 and 95 for Units 1 and 2, respectively, on April 8, 1985.

On June 28, 1985, the NRC issued Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," in which the NRC concluded that the actions taken by licensees in response to Phase I provided the intended level of protection against load drop accidents in pressurized water reactor (PWR) containments. Generic Letter 85-11 concluded that only installation of a single-failure-proof containment polar crane could further reduce the possibility of a heavy load handling accident in PWR containment buildings and thus satisfy the guidance in NUREG-0612 with respect to Phase II investigations. Therefore, the NRC concluded that there remained no residual heavy load handling concerns of sufficient significance to demand further generic action. Accordingly, the NRC indicated that there was no need to conduct further analyses pursuant to Phase II or submit further reviews and analyses on this subject. However, the NRC encouraged the implementation of any actions identified in Phase II regarding the handling of heavy loads that were considered appropriate.

On April 6, 2005, during the course of reviews associated with replacement of the RVH for PBNP Unit 2, it was determined that the PBNP Final Safety Analysis Report (FSAR) had not been updated as required by 10 CFR 50.71(e) to reflect the handling of heavy loads. This condition was entered into the PBNP corrective action program on April 7, 2005. Additional information regarding this condition is contained in letters from NMC to NRC dated April 15 and 20, 2005.

As corrective action, NMC initiated a change to the PBNP FSAR to incorporate the heavy load analysis. 10 CFR 50.59 states that a licensee may make changes in the facility as described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to 10 CFR 50.90 only if a change to the technical specifications incorporated in the license is not required, and the change does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59.

An evaluation of the heavy load analysis, performed by NMC pursuant to 10 CFR 50.59, concluded that incorporation of the analysis into the FSAR does meet one or more of the criteria in paragraph (c)(2) of 10 CFR 50.59. NMC's evaluation of paragraph (c)(2) of 10 CFR 50.59 concluded that the heavy load lift activity was not previously evaluated in the PBNP FSAR. Although this accident was described in a letter to the NRC dated November 22, 1982, it was not appropriately incorporated into the PBNP FSAR at that time, as required, by 10 CFR 50.71(e).

As discussed in the April 20, 2005, letter, NMC has assessed the ability to safely conduct RVH removal and replacement activities at PBNP in a manner that provides reasonable assurance of public health and safety. The risk associated with this activity has been assessed using plant specific data and is managed and further mitigated by appropriate levels of contingency planning. The analyses demonstrate that the potential for a load drop is extremely small and that NUREG-0612 Phase I implementation provides sufficient protection, especially when coupled with effective mitigation measures, such that the risk associated with potential heavy load drops is acceptably small. The analyses also conclude that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense-in-depth" is satisfied. NMC will incorporate both the PBNP method of NUREG-0612 Phase I compliance and an analysis of the RVH drop into the PBNP FSAR.

The NMC assessment of a postulated RVH drop event identified that systems and components remain available to mitigate its consequences (i.e., in service or can be placed in service in a functional or operable state by immediate manual or automatic actuation to meet its required shutdown key safety function, as defined in NP 10.3.6, "Shutdown Safety Review and Safety Assessment"). The proposed change provides additional assurances that these systems and components will be available during Unit 2 RVH replacement.

To support this assessment, NMC contracted with Sargent & Lundy to develop a finite element analysis (FEA) to model vessel behavior during a postulated RVH drop scenario. Enclosure 3 provides a revised FEA of the postulated RVH drop scenario prepared by Sargent & Lundy. NMC also contracted with Westinghouse to develop a FEA to determine effects on RCS piping based on specified reactor vessel deflection.

These detailed and independently performed analyses have been reviewed and accepted by NMC. The Sargent & Lundy analysis followed the same methodology as used for the Prairie Island Nuclear Generating Plant FEA. The Sargent & Lundy analysis demonstrated that reactor vessel deflection following a postulated RVH drop would not exceed 3.2 inches. This amount of deflection would be insufficient to cause a loss of decay heat removal capability. Although not used as a basis for this submittal, NMC obtained an additional independent FEA to calculate RVH deflection to provide a confirmatory calculation. The results of this confirmatory FEA were consistent with the Sargent & Lundy analysis.

In parallel with the Sargent & Lundy calculation, Westinghouse was contracted to analyze the ability of RCS piping to withstand the deflection caused by a RVH drop. Westinghouse was requested to perform this analysis for a 4-inch deflection, which was considered to bound the projected reactor vessel deflection. The results of the analysis show that stress values are less than the more restrictive criteria of $.7S_u$ specified in ASME Section III, Appendix F. In addition, Westinghouse was requested to analyze an additional deflection value of 6.5 inches, which is equivalent to the gap that exists between the RCS piping and the shield wall. The results of this analysis yielded stress values of greater than $.7S_u$ but did not predict failure of the RCS piping.

The combined results of the Sargent & Lundy and the Westinghouse analysis show that the damage from a RVH drop would not result in a loss of decay heat removal. Based on these results, NMC has concluded that adequate reactor core cooling and makeup capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop.

NMC reassessed lifting the new Unit 2 RVH and determined that existing controls ensure the probability of a head drop remains extremely low and additional administrative controls allow appropriate action to be taken to mitigate and manage the adverse consequences should such an unlikely event occur. The assessment demonstrated the acceptability of lifting the new Unit 2 RVH. Therefore, this condition does not significantly affect reactor safety.

This proposed amendment will continue to ensure a defense-in-depth approach is maintained during the lift of the new Unit 2 RVH and thereby provide reasonable assurance of public health and safety.

4.0 TECHNICAL ANALYSIS

The technical justification for the heavy load lift analysis applicable to the Unit 2 RVH is contained below. The heavy load analysis includes an abridgement of requirements and commitments established pursuant to NUREG-0612. The scope of the proposed licensing basis change includes both Phase I and Phase II submittals pursuant to Generic Letter 80-113 and Generic Letter 81-07.

The proposed licensing basis change involves the handling of heavy loads, with particular focus on reducing the consequences for a drop of the RVH. Heavy loads, if dropped on irradiated fuel, safe shutdown equipment, or equipment necessary for the continued removal of decay heat from either the reactor core or the spent fuel pool, could challenge the ability of the plant to maintain the integrity of fission product barriers as credited in the FSAR.

As such, heavy load handling equipment (such as cranes and special lifting devices) are support components credited in the FSAR.

In addition, failure of such equipment could initiate a transient, depending upon the equipment impacted by the failure and the operating mode of the facility at the time of the failure.

Finally, the control of heavy loads is implemented by a combination of design, inspection, testing, training, and procedural controls in order to comply with guidance established by NUREG-0612.

NUREG-0612 Phase I Controls

The PBNP license basis, as it addresses a postulated RVH drop event, is based on the prevention of such an event. Measures taken to prevent a RVH drop event included those taken under Phase I of NUREG-0612, and evaluated in the NRC Safety Evaluation dated March 27, 1984. An independent assessment was performed to evaluate the effectiveness of implementation of NUREG-0612 Phase I requirements. This assessment was completed on April 14, 2005, and concluded that NUREG-0612, Phase I guidance as described in the safety evaluation, has been satisfactorily implemented at PBNP.

The frequency of very heavy load drops of 5.6E-5 described in NUREG-1774 represents an estimate based on recent operating experience with loads over 30 tons. Although recent causes of load drops have predominantly involved sling failure, other causes are not discounted, as evidenced by Appendix A to NUREG-1774 and Section 4 of NUREG-0612. Of particular applicability to industry-standard handling systems is the potential for the wire rope supporting the load block to be cut or overloaded. This is of special concern with low head room lifts where the load block is deliberately raised near the upper block in order for the load to clear obstructions. From this position, a stuck relay or operator error combined with failure of the upper limit switch could cause a load drop before corrective measures, such as removing power to the crane could be implemented.

The new integrated head replacement involves a low head room lift. Therefore, NMC has developed measures to specifically minimize the potential for "two-blocking" as defined in NUREG-0612. These include testing of controls and limit switches and operational restrictions when the load is near its maximum lift height.

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 first limit switch that would be activated is a gear-actuated travel limit. This limit switch is activated by the rotation of the main hoist drum. It is currently set to actuate approximately 25 inches prior to a potential "two-blocking" incident. This limit switch de-energizes the hoist drive motor and prevents further movement in the upward direction.

A second counterweight-activated limit switch would be relied upon if the geared limit switch fails. This limit switch would be activated by the physical contact of the lower block with a counterweight connected to the limit switch. The initial contact of the lower block with the counterweight will occur approximately 21 inches prior to "two-blocking." The actual close of the limit switch will occur approximately 15 inches prior to "two-blocking." The limit switch circuit will de-energize the main power supply and consequently apply the hoist braking system.

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. 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 gear-actuated upper limit switch is tested daily when the containment crane is in use in accordance with RMP 9118-1. In addition, prior to lifting the RVH, 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 evolution to move the reactor head from the vessel to the storage stand involves lifting the head vertically directly above the vessel to an elevation that permits clearance of the 66-foot elevation interferences. The head is then moved horizontally from above the vessel to a location directly above the storage stand and finally lowered onto the stand. The movement from the stand to the vessel is a reverse of the above movements. The movement of the RVH is controlled by Safe Load Path (SLP) procedures. The SLP provides a path that minimizes crane manipulations and movement while the head is over the vessel. The head is lifted straight up to the height needed to clear the containment elevation 66-foot elevation before any bridge or trolley moves are made. The PBNP refueling cavity design does not have adequate room to allow the head to be moved to a position that completely clears the reactor vessel once the head is above the guide studs. When the head is replaced it is again lifted to a height necessary to clear the containment 66-foot elevation and then moved directly above the vessel, using crane reference marks and then lowered onto the vessel. This sequence minimizes crane manipulations and thus the potential for a crane failure or human performance induced failure.

The replacement RVH assembly has an overall height that is taller than the original RVH assembly. To move the head from the storage stand to the vessel (and vise-versa), the bottom flange of the head needs to clear the 66-foot elevation and other physical obstructions attached to the 66-foot elevation, which results in a lift height of 26.4 feet (67-foot elevation). The installed polar crane main hook has a maximum lift height, as determined by the physical design and limit switch settings. Based on the maximum physical hook elevation prior to two blocking (approximately 112-foot 5.75-inch plant elevation) and the replacement head assembly height (approximately 43-foot 1-inch, including the lift rig), the maximum height of the bottom flange of the replacement head is approximately the 69-foot 5-inch plant elevation (without limit switch settings). The inclusion of the main hook limit switch settings results in a maximum height of the bottom flange of the replacement head at approximately the 67-foot 3.5-inch plant elevation.

Operational restrictions have been added to procedure 2RMP 9096, which limit the height of the bottom of the RVH flange to the 67-foot elevation 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 activity is accomplished by using physical references and visual level checks during the lift.

The two independent limit switch designs, combined with the testing and operational restriction of the lift height, provide assurance that a "two-blocking" incident potential is minimal.

Polar crane controls are also checked in accordance with procedure RMP 9118-1. These checks include initial checks following installation of the radio controls, as well as daily checks when the crane is in use.

Core Cooling Configuration

The emergency core cooling system (ECCS) and normal core injection paths remain available during the postulated event. Core cooling water remains available to ensure adequate cooling and makeup is maintained to remove decay heat and keep the core covered.

Upon exhaustion of the Refueling Water Storage Tank (RWST) inventory, the residual heat removal (RHR) pumps would be realigned to take suction from the containment sump; with the safety injection pump(s) drawing from the RHR pump discharges as needed. This provides assurance that core cooling and makeup can be maintained for a prolonged period.

For Low Temperature Overpressure Protection of the reactor vessel, LCO 3.4.12 requires one train of safety injection to be disabled when the RVH is installed on the vessel. Prior to head installation, one train of safety injection is configured to prevent inadvertent start and pressurization once the RVH is installed in order to satisfy this requirement. PBNP will maintain the second train as available using administrative controls currently defined in its Shutdown Safety Review and Safety Assessment procedure (NP 10.3.6). In order to maintain the second train as available during head lift operations, PBNP has specified administrative requirements which will ensure prompt recovery if required. Pre-briefed and stationed operators will take the necessary local manual actions to ensure a timely recovery of the second train as necessary.

Postulated RVH Drop Sequence

The 1982 analysis referred to in the Wisconsin Electric Power Company letter dated November 22, 1982, established an acceptable bounding head drop scenario for evaluation of this event for replacement of the PBNP Unit 2 RVH during the spring 2005 refueling outage.

The "acceptable bounding scenario" referred to was predicated on the limiting reactor head drop being a concentric (or "symmetric") drop of the RVH onto the vessel. Such a drop would result in the greatest impact forces to the reactor vessel and supports, and therefore pose the greatest challenge to continued reactor coolant system integrity. In all postulated scenarios of slightly asymmetric drops, it is considered that efforts taken to expand the scope of evaluation beyond a symmetric drop, to include explicit evaluation, would ultimately be bound by the previously acceptable bounding scenario previously noted as having already been reviewed by NRC.

The 1982 analysis referred to in the Wisconsin Electric Power Company letter dated November 22, 1982, is contained in Westinghouse letter WEP-82-584, "Reactor Vessel Head Drop Analysis," dated November 15, 1982. Westinghouse performed the evaluation of the effects of a postulated RVH drop accident as described in NUREG-0612 for PBNP. A RVH drop is postulated to occur during refueling when the head is manipulated above the reactor vessel. The RVH is assumed to fall concentrically onto the reactor vessel.

Administrative controls have been established to limit the maximum RVH drop height to 26.4 feet. This drop height has been utilized in the analyses discussed below.

Analysis of Reactor Vessel Deflection

Based on NMC's assessment of the Sargent & Lundy FEA provided in Enclosure 3 and the Westinghouse analysis provided in Enclosure 4, the following bounding conditions apply:

Following the postulated RVH drop, the reactor vessel deflection would not exceed 3.2 inches. The RCS piping remains intact following the postulated Reactor Vessel deflection.

The impact of the postulated reactor vessel deflection on the attached RCS piping was assessed. This assessment was performed by Westinghouse and is contained in Enclosure 4. Westinghouse was requested to perform this analysis for a 4-inch deflection, which was considered to bound the projected reactor vessel deflection. The results of the analysis show that stress values are less than the more restrictive criteria of $.7S_u$ specified in ASME Section III Appendix F. In addition, a second case to analyze a deflection value of 6.5 inches, which is equivalent to the gap that exists between the RCS piping and the shield wall was conducted. The results of this analysis yielded stress values of greater than $.7S_u$ but did not predict failure of the RCS piping.

The combined results of the Sargent & Lundy and the Westinghouse analyses show that the damage from a RVH drop would not result in a loss of decay heat removal. Based on these results, NMC has concluded that adequate reactor core cooling and makeup capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop.

Infrequently Performed Tests or Evolutions Process

The control and conduct for the RVH installation will be accomplished in accordance with appropriate plant procedures and work instructions. Procedure NP 1.2.6, "Infrequently Performed Tests or Evolutions (IPTEs)," will be the governing document in controlling the pre-job briefings prior to commencement of work activities. 2RMP 9096, "Reactor Vessel Head Removal and Installation," is the routine maintenance procedure used to control work activities and provides contingencies in the event of a polar crane malfunction.

The commencement of work is in accordance with station work schedules and begins with the selection of the crew. After the crew is selected, an IPTE high-risk briefing is conducted in accordance with station procedures. Information discussed during the briefing include, but are not limited to, the scope of the job, communications, hazards, radiological conditions, operating experience, key error traps, and ACEMAN principles. The roles and responsibilities are clearly identified such that all personnel are aware of each other's duties for the evolution.

Work then progresses per 2RMP 9096 until the work activity is completed. In the event of a crane malfunction, the crew performing the work would use 2RMP 9096, Attachment G, "Failure Modes/Contingency Actions for Polar Crane During Head Lifts." In the event of a RVH drop, the personnel located in containment would evacuate the containment.

Mitigation Contingency Strategy Evaluation

In the event of a RVH drop, operators will first be alerted by reports from personnel who witnessed the event. The operators will assess damage using indications available to them in the control room.

The mitigating strategy will retain core cooling and makeup using RHR, charging, and safety injection flow paths. Minimum equipment availability is established as described within Enclosure 2. The Shutdown Emergency Procedures (SEPs) will direct mitigating actions as necessary to include injection into the core and eventual sump recirculation.

The Operations staff would implement SEP-2, "Shutdown LOCA Analysis," in the event of indications of lowering reactor vessel level. Upon entry to SEP-2, the procedure will direct a transition to SEP-2.3, "Cold Shutdown LOCA," 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 safety injection systems. When inventory is depleted, containment sump recirculation can be aligned using RHR.

SEP-2.3 will direct checking whether charging or safety injection flow is adequate to stabilize or restore RCS inventory. Upon exhaustion of the RWST inventory, RHR pumps will be realigned to take a suction from the containment sump to ensure long-term cooling.

Long-term plant status and future recovery actions will be determined in accordance with PBNP's processes and practices.

RCS Makeup Capability

As an upper bounding scenario, all bottom mounted instrument (BMI) tubes are postulated to sever. The resultant inventory loss from the RCS is postulated to be approximately 300 gpm. This value is based on all 36 BMI tubes being severed in a manner that causes minimal flow restriction from the break and the resultant gravity fed flow of RCS water through the tubes' nominal interior diameters.

There are numerous flow paths available to provide makeup to the reactor vessel. The various flow paths are listed in the table below. Available pumps include:

- Two RHR pumps, each with a design capacity of 1560 gpm at a design head of 280 feet.
- Two safety injection pumps, each with a design capacity of 700 gpm at a design head of 2600 feet.
- Two charging pumps with a capacity of 60 gpm each.

	System	RV inlet
1	Safety Injection Train A via SI-878D	A Cold Leg
2	Safety Injection Train A via SI-878B	B Cold Leg
3	Safety Injection Train B via SI-878A	B Core Deluge
4	Safety Injection Train B via SI-878C	A Core Deluge
5	RHR via 2RH-720	B Cold Leg
6	Charging via 2CV-1298	A Cold Leg
7	Charging via 2CV-1296	B Cold Leg
8	Charging via Auxiliary Spray Line	B Hot Leg

In the event of a loss of coolant event during cold shutdown, procedure controls are established to ensure adequate core inventory and cooling are maintained. The procedure performs the following:

1) Check RHR Pump conditions and secure if required due to system voiding
2) Establish charging from the RWST to maintain RCS inventory
3) Establish safety injection flow from the RWST via one SI pump if necessary
4) Verify adequate injection flow based on inventory and RCS temperature
5) Establish low head injection flow if required based on temperature
6) Establish containment sump recirculation if required based on RWST level.

In the event of a shutdown LOCA, operations would diagnose the problem based on communication of the drop by the containment SRO and by indications of a loss of reactor vessel level.

Time to boil curves provided in SEP-1, "Degraded RHR System Capability," show 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. Thus, 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.

Dose Assessment

An evaluation was performed to assess the offsite and control room dose consequence following a RVH drop. The event sequence assumes that the RVH drops onto the vessel causing clad damage to all of the fuel assemblies in the core, which results in a gap release. In addition, damage to the bottom mounted instrumentation tubes is assumed such that coolant is lost through these penetrations. Initial makeup of the RCS to the vessel is via suction from the Refueling Water Storage Tank (RWST) to the safety injection pumps, RHR pumps, or charging pumps. Once the RWST volume is exhausted, the RHR system is realigned to recirculate the coolant in the containment sump to maintain the core sub cooled.

A RVH drop is generally treated as a fuel handling accident; however, for PBNP the RVH drop is also treated as an initiator for a Loss of Coolant Accident (LOCA) at shutdown since damage to the bottom mounted instrumentation is assumed to occur resulting in an uncontrolled release of coolant. Therefore, this dose evaluation uses a combination of LOCA and FHA input assumptions as they apply to the event scenario.

For purposes of providing a bounding source term, the postulated RVH drop is assumed to result in clad damage to 100% of all fuel assemblies, such that a complete gap release occurs.

Prior to moving the RVH, the following conditions are assumed to be in effect:

- Reactor has been shut down for greater than 30 days;
- Containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks operable);
- Purge supply/exhaust system isolation valves closed;
- All penetrations that allow containment atmosphere to communicate with outside or PAB atmosphere are closed.

There are two possible release paths for a postulated RVH drop at PBNP: containment through open penetrations or leakage through containment barriers and ECCS leakage due to eventual operation of the recirculation mode of the RHR system. These release paths were evaluated using the guidance provided in Regulatory Guide (RG) 1.195. The guidance is appropriate for PBNP since 10 CFR 100 is the licensing basis for all of the design basis accident analyses except the fuel handling accident, which is licensed to 10 CFR 50.67 (NRC SE dated April 2, 2004).

The guidance in RG 1.195 does not provide specific accident offsite dose acceptance criteria for the heavy load drop event. However, NUREG-0612 does categorize the heavy load drop event to be in the same class of limiting faults for which the radiological dose acceptance criteria are stated to be "well within" 10 CFR 100, meaning 25% of the 10 CFR 100 limits. The FHA is within this class of accidents, which generally would also be used to assess a heavy load drop such as a RVH. Therefore, the offsite dose criteria are 6.3 rem whole body and 75 rem thyroid as listed in Table 4 of RG 1.195. The PBNP control room dose criteria is provided by 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 and interpreted to mean 5 rem whole body, and 30 rem thyroid.

Containment Release Evaluation

Prior to moving the RVH over the vessel, containment closure is established. Containment closure is defined as the containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks operable), the purge supply/exhaust system isolation valves closed, and all penetrations that allow containment atmosphere to communicate with outside or PAB atmosphere closed. The postulated RVH drop does not result in pressurization of the containment building. This is because the event occurs while the reactor coolant system is open to the containment building atmosphere and sufficient RCS makeup is available to provide cooling to the core. Heat is removed from containment by makeup to the RCS and containment sump recirculation. Since the containment configuration meets Footnote 2 to Position 5.1 of Appendix B to RG 1.195 and there is no pressure differential induced by the accident, there is no release via containment leakage.

ECCS Leakage Release Evaluation

The postulated RVH drop occurs when the RCS is open to the containment atmosphere. Therefore, the damage to the fuel assemblies is not driven by a thermal-hydraulic event but is assumed to occur due to impact of the head. Since RCS makeup provides cooling to the vessel and maintains the core covered, core cooling and makeup will be maintained by the heatup of the injected coolant and heat removal by nucleate boiling. A departure from nucleate boiling will not occur with the core covered. The coolant will not exceed saturation temperature at the RCS pressure, which is expected to be near atmospheric pressure. Due to the good conductivity of zirconium alloys, the cladding temperature will only be slightly above the coolant temperature. This is well below the temperature needed to cause cladding damage. Therefore, the non-LOCA gap fractions provided in Table 2 of RG 1.195 are applicable to the RVH drop event since no additional release from the fuel (e.g., due to fuel melt) will occur. RG 1.195 provides a larger gap fraction for I-131 than for the other isotopes of iodine. For this evaluation, the I-131 gap fraction of 8% is applied to all isotopes of iodine.

As previously stated, in order to establish a clearly bounding source term assumption, it is conservatively assumed that the drop of the RVH results in a 100% gap release to the coolant. The gap release is based on the total core inventory assumed for the licensing basis LOCA adjusted for the decay time from shutdown and the nuclide gap release fractions. For the ECCS leakage path, all of the gap activity of iodine is assumed to be retained in the coolant while the noble gases are not retained in the coolant. Therefore, the evaluation did not consider a noble gas release through the ECCS leakage. Assuming that all of the iodine activity remains in the coolant will result in a conservative dose assessment of the ECCS leakage pathway. The source term for the RVH drop in the coolant (i.e., the sump source term) is based upon the current licensing basis (CLB) LOCA total core inventory adjusted for a 30-day decay and the non-LOCA gap fraction. The CLB LOCA core inventory at shutdown can be found in PBNP FSAR Table 14.3.5-1. Table 1 of this enclosure provides the 30-day adjusted activities and the gap inventory assuming 8% of the total inventory. At 30 days post-shutdown, I-134 and I-135 have decayed away.

Table 1: 30-day Post-Shutdown Total Gap Inventory Nuclide		
Nuclide	30-day Total Inventory (Ci)	Gap Inventory (Ci)
I-131	3.47E+06	2.78E+05
I-132	1.11E+05	8.88E+03
I-133	3.56E-03	2.85E-04

The offsite and control room doses due to the RVH drop ECCS leakage are estimated by adjusting the CLB LOCA ECCS leakage doses for the resulting RVH drop source term. Additionally, the control room dose estimates take into account the ECCS leakage rate factor of two multiplier (Position 4.2 of Appendix A of RG 1.195) and the measured unfiltered inleakage rate. The offsite CLB LOCA ECCS leakage doses already take into account the factor of two multiplier for the ECCS leakage rate and are not impacted by unfiltered inleakage.

No other adjustments are made on CLB LOCA ECCS leakage doses, other than those discussed above. Therefore, all other input assumptions used in the CLB LOCA radiological consequence analysis are, in effect, used in the RVH drop ECCS leakage analysis.

Since dose is directly proportional to the amount of activity present, a scaling factor can be used to estimate the dose due to RVH drop ECCS leakage. The scaling factor is calculated by dividing the CLB LOCA ECCS sump source term by the RVH drop sump source term (Table 2 of this enclosure). The CLB LOCA ECCS sump source term is one-half the total core inventory values for iodine provided in FSAR Table 14.3.5-1.

Table 2: CLB LOCA and RVH Drop Sump Source Term Comparison			
Nuclide	CLB LOCA (Ci)	RVH Drop (Ci)	Scaling Factor
I-131	2.07E+07	2.78E+05	75
I-132	2.96E+07	8.88E+03	3.3E+03
I-133	4.23E+07	2.85E-04	1.5E+11

In order to assess the impact of the measured unfiltered inleakage value on the control room operator dose post-RVH drop, a review of the constituents of the control room dose via the ECCS leakage pathway for PBNP is provided. The control room operator inhalation and whole body dose (due to activity internal to the control room) for PBNP is driven primarily by the amount of activity that passes through the control room ventilation filter into the control room. When the control room emergency filtration system (CREFS) is in operation (referred to as mode 4), it is assumed that 4950 cfm of outside air is supplied to the control room to provide filtered air which will pressurize the control room. Since the efficiency of the CREFS filters is 95% for elemental/organic and 99% for particulate, the activity is entering the control room via the ventilation system at an estimated rate of 250 cfm for elemental/organic and 50 cfm for particulate. The current licensing basis control room habitability analysis assumes that while in mode 4, the unfiltered inleakage is 10 cfm.

The CLB ECCS leakage pathway is assumed to contain only elemental iodine. Therefore, under the current licensing basis analysis, the dose to the operator via the ECCS leakage path is primarily due to the elemental iodine activity delivered through the ventilation system. Recent tests of unfiltered inleakage to the control room while in mode 4 determined that unfiltered inleakage is approximately 100 cfm. This information was previously discussed in NMC letters to the NRC dated December 5, 2003, and September 29, 2004. Incorporation of the measured unfiltered inleakage value into the RVH drop dose assessment results in an increase that is proportional to the ratio of x/y, where x is the rate of activity delivered to the control room including measured unfiltered inleakage (250 cfm + 100 cfm) and y is the rate of activity delivered under current licensing basis assumptions for unfiltered inleakage (250 cfm + 10 cfm). Therefore, the elemental dose increases by a factor of 1.35 (350 cfm / 260 cfm). Since the ECCS leakage pathway is assumed to contain only elemental activity, the control room doses increase at most by a factor of 1.35.

The offsite RVH ECCS leakage doses are conservatively estimated by dividing the CLB LOCA ECCS leakage dose values by the I-131 scaling factor. The control room external and internal cloud RVH ECCS leakage doses are conservatively estimated by dividing the CLB LOCA ECCS leakage dose values by the I-131 scaling factor then multiplying by 2.7, accounting for the ECCS leakage rate multiplier and the measured unfiltered inleakage. Including the impact of the measured unfiltered inleakage on the external cloud whole body dose is conservative, since this assumption does not actually impact the external cloud dose.

Table 3 of this enclosure provides the CLB dose values and the scaled dose values for the RVH drop event. The CLB LOCA ECCS leakage doses are taken from FSAR Table 14.3.5-6, except for the "control room (external)" values. The external cloud control room dose is discussed in FSAR Chapter 11.6 and represents the contribution to the control room operator whole body dose due to the activity external to the control room from release pathways: containment and ECCS leakage. FSAR 11.6 Reference 2, "WE Calculation 97-0115, Point Beach Nuclear Plant Control Room Internal Dose Rates Due to External Cloud, dated May 30, 1997," provides the pathway breakdown of the external cloud dose and was used to determine the ECCS leakage contribution to the external cloud dose. Per Reference 2 of FSAR 11.6, the dose value at 5 feet from the shielded control room window is 0.20 rem and at 10 feet from the shielded control room window is 0.12 rem. The value for control room (external) provided in Table 3 of this enclosure was calculated by summing the location doses (5 feet and 10 feet from shielded control room window) weighted by the time spent in front of the window (25% occupancy at 5 feet and 75% occupancy at 10 feet). Since these values are derived for the design basis LOCA, the accident duration is taken to be 30 days.

Table 3: RVH Drop Dose Consequence (Rem)				
Location	CLB LOCA ECCS Leakage (rem)		RVH Drop ECCS Leakage (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
Site Boundary	57.12	0.24	0.8	3E-03
Low Population Zone	37.0	0.06	0.5	8E-04
Control Room (Internal)	106.7	0.004	3.8	1E-04
Control Room (External)	-	0.14	-	5.4E-03

Note: The designation of Internal and External for the control room indicates the activity cloud location.

Conclusion

As seen in Table 3 of this enclosure, the site boundary and low population zone RVH ECCS leakage thyroid and whole body doses are less than the acceptance criteria of 75 rem thyroid and 6.3 rem whole body. The control room doses are less than the acceptance criteria of 30 rem thyroid and 5 rem whole body. Therefore, the bounding dose consequence of the RVH drop can be considered "well within" the 10 CFR 100 limits. These dose estimates are conservative because credit was not taken for the presence of the fresh assemblies. The current licensing basis radiological consequence of a design basis LOCA bounds the dose consequence of the postulated Unit 2 RVH drop, and the LOCA remains the limiting event for control room habitability. The source term assumed for the RVH drop event provides a significant margin of safety.

References for Dose Assessment

1. USNRC Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.

2. USNRC Safety Evaluation, "PBNP, Units 1 and 2 - Issuance of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated with Handling of Irradiated Fuel Assemblies and Use of Selective Implementation of the Alternative Source Term for Fuel Handling Accident," April 2, 2004.

Probabilistic Risk Analysis

Although this is not a risk-informed submittal, the following information is provided:

An assessment of the risk of a RVH drop event from a height great enough to cause severe RCS damage was performed in 1983. That assessment concluded that the probability of a RVH drop event was approximately 5E-5 per lift. A new assessment was performed using current probabilistic risk assessment (PRA) methods to estimate the core damage probability associated with the lift of the new RVH over the reactor vessel.

Initiating Event

The initiating event in this assessment is the drop of the RVH while it is suspended over the reactor vessel. The RVH is assumed to fall onto the reactor vessel flange, resulting in damage to the reactor vessel support structure.

NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," was written to address NRC Candidate Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Crane operating history from 1968 through 2002 was reviewed as part of this report to provide a risk assessment associated with lifts of Very Heavy Loads (VHL). The risk analysis included in NUREG-1774 considers VHL lifts for any crane at any operating nuclear station. The analysis considers a postulated drop of load at any point during the movement of a load from the initial lift until set-down. In addition, the risk assessment included in NUREG-1774 was set-up to determine the probability of a number of different end states (consequences).

The probabilistic analysis contained within NUREG-1774 is primarily concerned with the probability of a VHL drop at an operating commercial nuclear power plant. A VHL is defined as any load over 30 tons. The generic probability for any VHL drop is given as 5.6E-5 per lift. This value is based upon three (3) drops per 54,000 VHL lifts.

A plant-specific review has been performed to demonstrate that operational characteristics with respect to crane failures due to mechanical failures or human performance are not significantly different than the average of plants considered within NUREG-1774. Three (3) areas were reviewed and compared to the generic data included within NUREG-1774.

- VHL Drop Probability – PBNP data review indicates that approximately 429 VHL lifts were performed using the turbine building, primary auxiliary building or containment cranes between the period of January 1, 1995, and April 14, 2005 (average of approximately 20 per reactor per year). There were no drops identified during this time. NUREG-1774 provides a probabilistic value of 5.6E-5 per lift. Statistically, given the small number of VHL lifts performed at PBNP, it is not expected that a drop event would have occurred. This review suggests that PBNP does not deviate from the values contained within NUREG-1774.

- Load Slip Probability – Plant review indicates 429 VHL lifts. During this timeframe, PBNP experienced zero (0) VHL slips. NUREG-1774 states that there were six (6) load slips in the 54,000 VHL lifts considered. This review suggests that PBNP does not deviate from the values contained within NUREG-1774.

- Human Error Probability (probability of human error per lift) – A review of plant data shows there were 29 human error and procedural events out of 50 lift-related events in more than 14,000 lifts of any size that took place between January 1, 1995, and April 14, 2005. The majority of the events are human error and procedure related. This is similar to the observation noted in NUREG-1774, demonstrating that PBNP is not an outlier compared to the data contained in this assessment.

  All three drops referenced within the NUREG-1774 involve a failure of rigging and all involved a human error associated with the rigging. It is considered that this value is conservative and bounding for an RVH drop for the following reasons:

  1. If a load was to drop as a result of a rigging problem, there is a likelihood that the load drop will happen at the beginning of the lift because of the lift rig failing when it is first put under stress. For this RVH lift, there is some likelihood that the load drop may occur when the RVH is not suspended over the reactor vessel, or that it occurs from a low height. During the RVH set, only the end of the lift takes place over the reactor vessel. If rigging failure occurs during the RVH installation, it is much more likely to occur at the beginning of the lift when the RVH is not above the reactor vessel. Because these split fractions are not known with great certainty, it is assumed for this assessment that any drop that occurs takes place over the reactor vessel.

  2. The three VHL drops cited in NUREG-1774 were all failures of nylon or Kevlar sling-type riggings being used on cranes not located in containment. These rigging failures were, at least in part, attributed to human error resulting in the slings being overstressed or unprotected from damage during the lift. The rigging used for the RVH lift is constructed of steel, is specifically designed for this lift and is used exclusively for this lift. The RVH rigging and crane is inspected prior to the lift. The RVH lift is rigorously controlled by procedure, and key personnel involved are experienced with this particular lift.

Considering the factors discussed above, it can be stated with a high degree of confidence that the PBNP plant-specific probability of a RVH drop is less than the upper bound estimate of 5.6E-5 per lift provided in NUREG-1774. The three VHL drops that have occurred in the industry were attributed to a failure mode that cannot occur for a RVH lift because a single purpose, steel lifting rig is used rather than a general use, nylon or Kevlar sling. Even though these three failures can be eliminated because the specific failure mode does not apply to a RVH lift, the entire population of 54,000 VHL lifts can be used because all of the remaining possible failure modes are still applicable to all of these VHL lifts.

With these three VHL drops eliminated, the correct number of failures for the numerator is now some value between 0 and 1. It is a common PRA practice in the situation where no failures have occurred to use an estimated value of 0.5 in the numerator. Assuming 0.5 drops in a sample size of 54,000 VHL lifts results in a more appropriate VHL drop probability of 9.3E-06 per lift.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

In accordance with 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests exigent review and approval of a revised analysis regarding the control of heavy loads at Point Beach Nuclear Plant (PBNP) Unit 2 needed to reflect inclusion of a Reactor Vessel Drop (RVH) drop event. Approval of this change to the licensing basis is requested for use to complete activities associated with the spring 2005 refueling outage.

PBNP Unit 2 has completed routine refueling and is currently shutdown with the RVH removed. The lifting and setting of the RVH are awaiting completion of associated analyses. Consequently, exigent NRC approval of the proposed amendment is requested to preclude delays in resumption of PBNP Unit 2 power operation.

NMC has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of PBNP Unit 2, in accordance with the proposed amendments, presents no significant hazards. The NMC evaluation against each of the criteria in 10 CFR 50.92 follows:

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 the heavy load analysis supporting lifting of the PBNP Unit 2 RVH over a reactor vessel containing fuel assemblies. A RVH drop can be postulated as an initiator of a Loss of Coolant Accident (LOCA) under shutdown conditions. The consequences of this postulated accident are bounded by the licensing basis analysis for a LOCA. A RVH drop is of sufficiently low probability such that the probability of a LOCA is not significantly increased. Administrative controls have been established to assure continued availability of multiple independent sources of water to provide core cooling and makeup. Therefore, the consequences of a LOCA are not increased. There will be no pressurization of the reactor coolant system as a result of this postulated event. Containment closure will be established during this evolution. The proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences. All Technical Specifications are satisfied and required equipment is operable. Therefore, this change would not significantly increase the probability of occurrence or 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 the heavy load drop analysis supporting lifting of the PBNP RVH over a reactor vessel containing fuel assemblies. The drop of a RVH can be postulated as an initiator of a LOCA under shutdown conditions. Supplemental controls have been established to assure continued availability of multiple independent sources of water to provide core cooling and makeup and to ensure the closure of containment. Adequate core cooling and makeup water remains available from core cooling water systems. Maintaining core cooling and makeup and closing containment assures that the drop of a RVH is bounded by the existing licensing basis analysis for a LOCA. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   The proposed change incorporates the heavy load analysis supporting lifting of the PBNP RVH over a reactor vessel containing fuel assemblies. A plant specific analysis was performed that demonstrates no significant reduction in margin to safety would occur. All the recommended margins regarding containment building polar crane loading are satisfied for this activity. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. Therefore, the proposed amendment does not result in a significant reduction in a margin of safety.

Conclusion

Operation of PBNP in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of any accident previously analyzed; would not create the possibility of a new or different kind of accident from any accident previously analyzed; and, would not result in a significant reduction in any margin of safety. Therefore, operation of PBNP in accordance with the proposed amendment presents no significant hazards.

5.2 Applicable Regulatory Requirements

10 CFR 50.59 states that a licensee may make changes in the facility as described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to 10 CFR 50.90 only if, a change to the technical specifications incorporated in the license is not required, and the change does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59.

An evaluation of the heavy load analysis, performed by NMC pursuant to 10 CFR 50.59, concluded that incorporating the analysis into the FSAR does meet one or more of the criteria in paragraph (c)(2) of 10 CFR 50.59. Specifically, the heavy load lift activity was not previously evaluated in the final safety analysis report. Although this event was described in a letter to the NRC dated November 22, 1982, it was not appropriately incorporated into the PBNP FSAR as required by 10 CFR 50.71(e).

10 CFR 50.71(e) requires that licensees shall periodically update their final safety analysis report (FSAR), to assure that the information included in the report contains the latest information developed. This update shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement. The update shall also include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", presents an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. On December 22, 1980, the NRC issued Generic Letter 80-113 which was supplemented on February 3, 1981, with the issuance of Generic Letter 81-07 regarding NUREG-0612.

Wisconsin Electric Power Company (WEPCo), then licensee for PBNP, submitted information to the Nuclear Regulatory Commission (NRC) regarding the handling of heavy loads at PBNP in accordance with the guidelines of NUREG-0612. However, the PBNP FSAR was not updated with this analysis.

NMC concludes that incorporation of the heavy load analysis into the licensing basis, to include an analysis of the RVH drop into the PBNP FSAR, requires a license amendment pursuant to 10 CFR 50.90. Because this analysis supports RVH lift over a reactor vessel containing fuel assemblies, exigent approval is required to prevent delays in resumption of PBNP Unit 2 power operation.

Based upon the considerations discussed above, (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 accordance 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.

5.3 Commitments

This letter contains three commitments as follows:

1. Prior to Unit 2 RVH lifting activities associated with the spring 2005 refueling outage, NMC will establish administrative controls for the RVH lift as described in Enclosure 2.

2. NMC will incorporate an analysis of the RVH drop into the PBNP FSAR.

3. NMC will incorporate the PBNP method of NUREG-0612 Phase I compliance into the PBNP FSAR

6.0 ENVIRONMENTAL EVALUATION

NMC has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure.

Accordingly, this proposed 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 this proposed amendment.

7.0 EXIGENT CIRCUMSTANCES

NMC considers that the 1982 failure to incorporate the RVH drop analysis into the PBNP Final Safety Analysis Report (FSAR) resulted in an unavoidable situation. NMC personnel conducted a review of the replacement RVH modification package for 10 CFR 50.59 applicability in February 2005. After reviewing the FSAR, it was determined that a 10 CFR 50.59 screening could be performed for the proposed modification. This review did not adequately address requirements associated with heavy loads that were contained in licensing basis correspondence. The requirements were not addressed because the PBNP FSAR did not reflect the results of a RVH drop analysis, which had been submitted to the NRC in November 1982.

On April 6, 2005, while collecting documents associated with NRC RVH replacement inspection activities, NMC staff raised questions regarding the licensing basis for handling heavy loads at PBNP. On April 7, NMC completed a review of relevant licensing basis documents and 10 CFR 50.59 applicability review for the replacement RVH and discovered that the PBNP FSAR had not been updated to reflect the 1982 NRC submittals for handling heavy loads. The condition was entered into the corrective action program, along with a recommendation to update the FSAR to reflect the previous analyses of heavy loads. Further evaluation led to the recognition that NMC had not formalized sufficient mitigative strategies for the control of heavy loads. Numerous additional corrective action program documents were initiated as NMC continued to investigate issues associated with this issue. NMC initiated corrective actions to develop and implement administrative controls to address the immediate need to lift the existing RVH.

NMC could not have reasonably avoided the situation. The 1982 error of not updating the FSAR led to missed opportunities to address current licensing basis requirements for handling heavy loads. Since April 7, 2005, the PBNP staff actively pursued the following activities:

- Deriving the licensing basis for control of heavy loads,
- Independently reassessing all NUREG-0612 Phase I commitments,
- Performing a risk-informed assessment of the current probability of a heavy load drop event,
- Enhancing the mitigating strategy for addressing a heavy load handling event,
- Developing and implementing administrative controls to lift the existing RVH so that the lifting evolution ensured a "defense-in-depth" mitigative strategy and provided a greater assurance of public health and safety,
- Re-evaluating NUREG-0612 Phase II guidance in order to develop a long-term programmatic approach to lifting and handling of heavy loads at PBNP,
- Obtaining additional information from the vendor regarding the impact forces related to the replacement RVH load drop event,
- Reevaluating the 10 CFR 50.59 evaluation for the replacement RVH,
- Performing revised finite element analyses (FEA)
- Reassessing the RVH drop event.

PBNP Unit 2 is currently shutdown following routine refueling. Therefore, NMC requests NRC approval of this amendment for Unit 2 on an exigent basis in accordance with 10 CFR 50.91(a)(6).

ENCLOSURE 2

REGULATORY COMMITMENTS TO ADMINISTRATIVE CONTROLS ASSOCIATED WITH LIFT OF THE NEW UNIT 2 REACTOR VESSEL HEAD AND ITS PLACEMENT ON THE REACTOR VESSEL

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

2. A Senior Reactor Operator will be stationed in containment during reactor vessel head 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 reactor vessel head is suspended greater than 24 inches over the reactor vessel flange.

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

7. The Programmed and Remote (PaR) reactor vessel ISI 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 safety injection trains shall be available.

10. Both residual heat removal trains shall be operable.

11. Technical Specification 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 Unit 2 shall be operable.