May 13, 2005

NRC 2005-0063
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 1 to Request for Exigent Review of Heavy Load Analysis

Reference: Nuclear Management Company, LLC Letter Dated April 29, 2005

In the reference, 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 license 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 of the analyses for PBNP Unit 2 was requested on an exigent basis.

Following submittal of the referenced amendment request, NMC continued to perform assessments regarding control of heavy loads. The results of these assessments have provided additional information that has affected the previously postulated scenario. Therefore, NMC is submitting this supplement to the proposed amendment to provide the results of these assessments and to incorporate additional technical justification for the proposed amendment. Additionally, since assessment of this activity remains ongoing, NMC proposes to apply the Reactor Vessel Head (RVH) lift assessment being provided in this submittal on a one-time basis for the upcoming lift of the Unit 2 RVH. The previously requested review of these analyses for Unit 1 is hereby withdrawn. To facilitate NRC staff review of the proposed amendment, this supplement replaces the original request, transmitted in the reference, in its entirety.

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 following completion of refueling activities. Enclosure 1 provides a description, justification, and a significant hazards determination for the RVH drop event analysis. Enclosure 2 provides administrative controls that will be established during the Unit 2 RVH lift. Enclosure 3 submits Westinghouse Report, "Assessment of Reactor Vessel Head Drop,

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Point Beach Unit 1 and Unit 2", (LTR-RCDA-05-428, Revision 1, dated May 13, 2005) (Proprietary).

Also enclosed are a Westinghouse authorization letter, accompanying affidavit, Proprietary Information Notice and Copyright Notice, as applicable, for the updated Westinghouse assessment of the RVH drop event.

Since the report listed above as Proprietary contains information proprietary to Westinghouse Electric Company, 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 2.390.

Correspondence with respect to the copyright or proprietary aspects of the above reports, or the supporting Westinghouse affidavit, should reference the appropriate authorization letter (CAW-05-1996) 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.

NMC requests exigent approval of the analyses for PBNP Unit 2 by May 24, 2005. Approval of this analysis is requested solely for use on a one-time basis for Unit 2 RVH lifting activities upon completion of refueling activities associated with the spring 2005 refueling outage.

NMC remains confident in our ability to safely conduct reactor vessel head removal and replacement activities in a manner that provides reasonable assurance of protecting public health and safety, plant personnel and equipment.

<u>Summary of Commitments</u>:

This letter contains two commitments as follows:

1. NMC will establish administrative controls for the RVH lift as described in Enclosure 2.
2. NMC will continue assessment of the RVH drop event for future RVH lifts and submit the associated analyses for NRC review, as appropriate.

In accordance with 10 CFR 50.91, a copy of this application, 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 May 13, 2005.

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

Enclosures (3)

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
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ENCLOSURE 1

SUPPLEMENT 1 TO REQUEST FOR 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 needed to reflect inclusion of a reactor vessel head (RVH) drop event. Approval of this change to the licensing basis is requested solely for use on a one-time basis for Unit 2 RVH lifting activities upon completion of refueling activities associated with the spring 2005 refueling outage.

PBNP Unit 2 is currently conducting refueling operations with the reactor vessel head removed. The subsequent lift of the reactor vessel head is planned for May 2005. Consequently, exigent NRC approval of the proposed amendment is required to preclude delays in resumption of PBNP Unit 2 power operation.

2.0 PROPOSED CHANGE

NMC proposes changing the PBNP licensing basis to incorporate a revised RVH (heavy load) drop event analysis, specifically for one-time use on 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 risk-informed evaluation that continues to demonstrate the low probability of occurrence of a reactor vessel head drop as originally evaluated. For the specific case of a reactor vessel head lift, this analysis also requires that administrative controls be maintained during lifting of a reactor vessel head over a reactor vessel containing fuel assemblies in order to maintain defense-in-depth.

Administrative controls will be in effect whenever the reactor vessel head 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.

[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.

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 the 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. 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 reactor vessel head (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 reactor vessel head 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 § 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 § 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 § 50.59. NMC's evaluation of paragraph (c)(2) of § 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 reactor vessel head 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.

The April 20, 2005, letter document's one-time regulatory commitments associated with the lift of the previous Unit 2 RVH at the start of the Unit 2 spring 2005 refueling outage. During a telephone conversation conducted on April 16, 2005, NRC concurrence with NMC's mitigative strategy and commitments associated with this one-time RVH lift was obtained. The previous Unit 2 RVH was successfully removed from the vessel and transported to its lay down area.

The NMC assessment of a postulated RVH drop event identified numerous systems and components that would be available to mitigate its consequences. The proposed change provides additional assurances that these systems and components will be available during Unit 2 RVH replacement.

This proposed amendment will continue to ensure a defense-in-depth approach is maintained during the lift of the new Unit 2 RVH during the spring 2005 refueling outage and thereby provide reasonable assurance of public health and safety.

NMC reassessed lifting of 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 demonstrates the acceptability of lifting the new Unit 2 RVH during the spring 2005 refueling outage. Therefore, this condition does not significantly affect reactor 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 new appendix 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 potential for a heavy load drop of the reactor vessel head. 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.

Phase II of implementation of NUREG-0612 required either single-failure-proof cranes, when they are capable of lifting heavy loads over irradiated fuel, or that the drop of any such loads be analyzed to be acceptable by meeting four separate criteria:

I. Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits);

II. Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95;

III. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated); and

IV. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

Crane/load combinations were evaluated as part of the Phase II submittal as not requiring such analyses with one exception. The exception was found to be a heavy load drop over the reactor vessel. An analysis was initiated to evaluate whether the above four criteria could be met for the most limiting lift above the reactor vessel (the reactor head). It was found that a drop of the reactor vessel head could result in severe enough damage that Criterion IV could not be met (i.e., that a loss of decay heat removal could result). As a result of the inability to meet Criterion IV, analyses to determine ability to meet Criterion I, II, and III were not fully developed at that time.

The drop of a reactor vessel head is expected to bound all other heavy loads that may need to be handled above the reactor. Specific heavy loads that would generally be bounded due to their lower mass are the reactor upper internals, the PaR device, and a reactor coolant pump (RCP) motor. This would ensure that Criterion II was met. Drops of the upper internals, reactor head, and RCP motor were not evaluated for Criterion II because the geometry of the dropped load (for the head and internals) would preclude contact with the fuel, or because administrative controls preclude handling the load over an exposed (and therefore unprotected) core.

While these results were submitted to the NRC for review in the NUREG-0612 Phase II submittals, PBNP received no acknowledgment that the content of the submittals had been reviewed or that they had been approved.

The above discussion supports incorporation of the analysis for control of heavy loads into the PBNP FSAR. Additional analyses that support the proposed administrative controls are described later in this submittal.

NUREG-0612 Phase I Controls

The PBNP license basis, as it addresses a postulated reactor vessel head (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 1 has been satisfactorily implemented at PBNP.

The elements credited for minimizing the potential for a RVH handling event at PBNP are documented in NMC's April 20, 2005, letter to NRC.

Reactor Vessel Head Drop Assessment

The original PBNP reactor vessel head (RVH) drop analysis is contained in Westinghouse analysis WEP-82-584, "Reactor Vessel Head Drop Analysis." That analysis stated that an evaluation of impact effects on the primary coolant pipe, reactor coolant pump supports, and steam generator supports, due to a RVH drop, would require a detailed plastic analysis. However, in lieu of such a detailed analysis, the Westinghouse analysis developed its scenario based on engineering judgment. The scenario postulated a potential loss of primary coolant loop piping connections. A letter to NRC dated November 22, 1982, 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. An assessment of the PBNP RVH drop event was performed by Westinghouse in 2005 to determine if inelastic structural behavior would change the conclusions reached in 1982. This assessment, which is contained in Westinghouse letter WEP-05-161 dated May 11, 2005, is provided in Enclosure 3.

The 2005 assessment determined that the 1982 conclusion was not realistic as follows.

> Inelastic material behavior will absorb significant energy such that there would be no structural failure that would cause loss of coolant to the core. This evaluation has used energy principles to demonstrate that gross plastic failure of the support columns is not likely to occur. The maximum strain levels, for head assembly drop of approximately 45 feet above the flange through air is 2% and below 1% for a 20 foot drop. From this evaluation, it can be concluded that the columns can absorb significant energy and still remain functional. However, if the columns do fail, there is still reserve strength from the primary coolant loop piping that will support the reactor vessel.
>
> Therefore, it can be concluded that if a potential head drop was to occur, the reactor vessel will be adequately supported either by the supports, or by the primary coolant piping such that there will be no damage to the attached primary coolant piping which would prevent the removal of decay heat from the core. Further, the reactor vessel integrity is maintained since it is adequately supported so that it does not break and leak out water, except for the safety injection lines that potentially may be lost.

A calculation was performed to compare the impact forces at the vessel supports resulting from a postulated drop of the original PBNP RVH assembly against that of the replacement head assembly, using various combinations of replacement head assembly weights and drop heights.

Based upon the results of the calculation, it was concluded that the impact forces at the vessel supports, caused by the drop of the replacement head assembly, will be equivalent to or less than the impact caused by the postulated drop of the original head assembly provided:

1. The replacement head assembly drops from a height that is lower than the drop height of the original head, due to physical lift limitations associated with the greater overall height of the replacement head assembly.

2. The replacement head assembly weighs 194,000 lbs or less.

Based on the resultant impact forces at the vessel supports of the replacement head assembly drop being equivalent to or less than those of the original head assembly drop, the postulated damage would be the same or less. The maximum allowable lift height for the RVH is administratively controlled in station procedures.

Load Drop Scenario

In order to establish a bounding scenario to enable development of mitigating strategies for the RVH drop event, the nuclear steam supply system (NSSS) vendor, Westinghouse, was requested to provide greater details of the sequence of events and postulated damage beyond that previously reported in the 1982 Westinghouse letter (WEP-82-584).

Westinghouse employed the services of a recognized expert in the field of assessing damage to plant components using elastic and inelastic methodology. Westinghouse letter WEP-05-161, dated May 11, 2005, concluded the 1982 evaluation reached extremely conservative conclusions regarding structural effects since it was limited to evaluated impact on structures based on elastic behavior.

Conclusions From Westinghouse Letter WEP-05-161

Inelastic material behavior will absorb significant energy such that there will be no structural failure that would cause loss of coolant to the core. This evaluation has used energy principles to demonstrate that gross plastic failure of the support columns is not likely to occur.

The maximum strain levels, for head assembly drop of approximately 45 feet above the flange through air is 2% and below 1% for a 20 foot drop. From this evaluation, it can be concluded that the columns can absorb significant energy and still remain functional.

However, if the columns do fail, there is still reserve strength from the primary coolant loop piping that will support the reactor vessel.

A second drop scenario was evaluated wherein the columns and other potential restraining structures are assumed to fail and only the reactor coolant pipes are available to restrain the vessel from further drop. This is a conservative scenario since it assumes that the reactor vessel safety injection piping fails, columns fail, and the concrete below the vessel ring girder fails, without providing energy absorption.

In this evaluation, plastic hinges are assumed to form in the loop pipes and the resulting displacement of the reactor vessel assembly is determined. For a 50 foot drop through air, a maximum deformation of less than 7 inches is predicted, after the loop pipes contact the biological shield wall. For a 15 foot drop, the maximum deformation is less than 3 inches. Assuming that the available flow area is reduced by these deformations, the loss in flow area is under 20%. That is for a 50 foot drop through air, about 80% of the flow area remains, and for a drop of 15 feet the remaining flow area is 95% of the original flow area.

This assessment confirmed that the concrete supporting the steel structures/reactor coolant piping would not fail, considering the maximum loads determined.

Therefore, it can be concluded that if a potential head drop occurred the reactor vessel will be adequately supported either by the supports, or by the reactor coolant piping such that there will be no damage to the attached reactor coolant piping which would prevent the removal of decay heat from the core.

Based upon this evaluation, conducted by a recognized industry expert, a previously postulated scenario of a total loss of decay heat removal capability and the magnitude of the reactor vessel drop is not considered credible.

RVH Drop with Loss of Normal Injection and Cooling

This scenario is not considered credible by the Westinghouse Letter, WEP-05-161, "Assessment of Reactor Vessel Head Drop," Point Beach Unit 1 and Unit 2, however, it is the scenario outlined in the 1982 Westinghouse Analysis WEP-82-584, "Reactor Vessel Head Drop Analysis." Therefore, it has been used for the purpose of establishing the bounding event for NMC's mitigating strategy.

In this scenario, all six upper vessel penetrations are completely lost such that no water can be injected via normal paths. The vessel remains suspended so bottom-mounted instrumentation (BMI) penetrations remain intact. Based on engineering judgment, the BMI connections remain intact because of the clearance between the lines and the floor and the flexibility of the lines. Based on the recent evaluation (WEP-05-161), this is a conservative bounding condition utilized to establish conservative mitigation strategies.

At the time of the planned lift, more than 38 days will have elapsed since reactor shutdown, and approximately 30% of the core has been replaced with unirradiated fuel. As a result, the total decay heat load has been calculated to be approximately 5.8E+6 BTU/hr and the requirements for makeup due to decay heat boil-off will be 12.5 gpm.

With this decay heat load, and assuming an initial temperature of 100°F, it has been calculated that there will be at least 50 minutes before the volume of water remaining in the reactor vessel heats to the point of boiling (neglects the sensible heat capacity of the fuel, the vessel and the internals). This heatup rate provides sufficient time to implement the mitigating strategies before the onset of boiling as validated through interviews conducted in support of the PRA analysis. Time to initiation of core uncovery was calculated to be an additional 58 minutes with no makeup.

As a compensatory measure for this bounding scenario, prior to suspending the Unit 2 RVH over the vessel, NMC will install a temporary modification that will provide two redundant water supply connections to the RVH.

Temporary Modification (TM 2005-008) will install two hoses connecting the Containment Spray system to the reactor head during the head lift evolution. The hose connections will facilitate the addition of water to the core in the event of an uncontrolled descent of the head and subsequent failure of the vessel supports and piping connections. One hose will be connected from a Containment Spray test flange to the head vent connection, and one hose will be connected from a second Containment Spray test flange on the opposite train to the Reactor Vessel Level Indicating System (RVLIS) connection on the reactor head.

The temporary modification has been designed to ensure that the 12.5 gpm cooling flow requirements necessary to offset boiling and core uncovery concerns can be satisfied with either hose connection.

The hose connections and fittings have been designed to withstand the highest expected pressure and temperature conditions, which would occur if the containment spray pumps are operating in the containment sump recirculation mode and are being fed by the RHR pumps taking suction from the containment sump. Applicable codes and standards have been used for the design. The hose routing has been selected to ensure that in the event of an uncontrolled descent of the reactor vessel, adequate length is available to prevent the hoses from being pulled from their connection points to the containment spray system. Additionally, the routing has been chosen to prevent flow reduction due to kinking of the hoses. Thus, there is assurance that the hoses will perform their required function and be available if needed.

Adequate venting from the vessel is expected given the nature of the failure of the vessel inlet and outlet piping. The severance of these pipes results in adequate openings in the vessel for venting. If piping were not severed, normal core injection paths remain available.

The temporary connections will be made up and verified aligned to the containment spray headers prior to suspending the RVH over the reactor vessel. In the event of a RVH drop event that causes severe damage to the RCS, manual initiation of either train of containment spray will ensure adequate core cooling is maintained to remove decay heat and keep the core covered.

Upon exhaustion of the RWST inventory, the residual heat removal pumps would be realigned to take a suction from the containment sump; with the containment spray pump(s) drawing from the residual heat removal pump discharges. This provides assurance that core cooling can be maintained for a prolonged period.

Mitigation Contingency Strategy Evaluation

In the event of a RVH drop, operators will first be alerted by reports from personnel who witnessed the event. In addition, in the bounding scenario, alarms will alert the operators to abnormal plant conditions. The operators will assess damage using indications available to them in the control room.

The mitigating strategy will attempt to restore core cooling using charging, safety injection and containment spray flow paths. Minimum equipment availability is established within the Technical Requirement Manual (TRM) and Enclosure 2. The

Abnormal Operating Procedure (AOP) and Shutdown Emergency Procedures (SEPs) will direct mitigating actions, including injection into the core and eventual sump recirculation.

The Operations staff would enter AOP-8J, "Reactor Vessel Head Drop", which will direct implementation of SEP-2, "Shutdown LOCA Analysis" while continuing in AOP-8J. 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 using the charging and safety injection systems. If these actions are successful, the containment sump recirculation can be aligned using RHR.

SEP-2.3 in conjunction with AOP-8J will check whether charging or safety injection flow is adequate to stabilize or restore RCS inventory. If RCS inventory is not restored, AOP-8J will initiate core cooling using the containment spray system temporary modification (TM 2005-008). If containment spray is supplying core cooling, AOP-8J will direct the residual heat removal pumps to be realigned to take a suction from the containment sump upon exhaustion of the RWST inventory; with the containment spray pumps drawing from the residual heat removal pump discharge to ensure long term cooling.

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

Dose Assessment (NUREG-0612 Criterion I)

NUREG-0612 Criterion I requires evaluation of releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

The dose assessment for postulated reactor vessel head drop event for PBNP Unit 2 is assessed under two segments regarding the potential for fuel damage: first, mechanical failure of the fuel rod cladding due to the impact of the head on the vessel and second, thermal hydraulic evaluation during coolant recirculation.

During normal refueling operations, drive shafts for the rod cluster control assemblies (RCCAs) are carefully inserted into the head while the reactor vessel head is being replaced onto the vessel. However, it cannot be assumed that all the drive shafts enter the head penetrations during a RVH drop event. As a reactor head undergoes a postulated drop onto a reactor vessel, the drive shafts extending out of the upper internals may not find their way properly into the corresponding head penetration. In such a case, the inadequately located drive shafts will be compressed, either by the edge of the alignment funnel or possibly by the inside of the head itself. The extreme weight and inertia of the falling head are expected to compress the drive shafts until a buckling/collapse situation is reached. This compressive load in the drive shaft will tend to be reacted by the fuel assembly. During this type of accident, the head assembly itself does not come in contact with the fuel assemblies. Therefore, the compressive load on the drive shafts is the only major force experienced by the fuel assemblies during this event.

The fuel assembly structural skeleton is comprised of the RCCA guide thimbles, in conjunction with the grid assemblies and the top and bottom nozzles. The top and bottom ends of the guide thimbles are fastened to the top and bottom nozzles, respectively. The guide thimbles carry axial loads imposed on the assembly. These loads, as well as the weight of the assembly, are distributed through the bottom nozzle to the lower core support plate. The grid assemblies, in turn, are fastened to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. The fuel rods are contained and supported within this skeletal framework. The grid assemblies consist of individual slotted straps which are assembled and interlocked in an 'egg crate' type arrangement. There are a number of grids that are connected to the guide tubes. The grids support the fuel rods in a suspended fashion, with gaps at the top and bottom of the fuel rods between the nozzles and the fuel rods. The gap is provided to allow expansion of the fuel rod without contacting either the top or bottom nozzle. Therefore, the fuel rods provide no mechanical structural function aside from the containment of the fuel pellets. The drive shaft is connected to the RCCA assembly, which is inserted into selected fuel assemblies. The PBNP reactor contains 33 RCCAs. Therefore, any loading due to the vessel head drop would be transmitted to the RCCA and its host fuel assembly through the drive shaft. The fuel assembly structure would absorb the load, and would have to become severely deformed before the fuel rods themselves could become loaded due to the gap that exists at each end of the fuel rods.

For plants that currently have Westinghouse-supplied drive shafts and Westinghouse-supplied fuel, Westinghouse has concluded that in the event of a reactor vessel head drop, skeletal damage to the 14x14 fuel assembly structure could occur, but that the fuel cladding integrity would be maintained (Westinghouse Nuclear Safety Advisory Letter 04-6 and WCAP 9198 Revision 1, "Reactor Vessel Head Drop Analysis"). This conclusion applies to PBNP Unit 2, because it has Westinghouse supplied drive shafts and the Westinghouse 14x14 VANTAGE 422+ fuel. In addition, at the end of the operation cycle for Unit 2, no fuel defects were present. The core reload for Unit 2 Cycle 28 replaced 36 thrice burned assemblies with fresh 422V+ fuel assemblies. Twenty of the fresh fuel assemblies have RCCAs; the remaining 13 RCCA locations contain once-burned fuel. With this core configuration, the load placed on the drive shafts of the RCCAs will primarily be transferred to fuel assemblies with fuel rods that have never been irradiated. Since there is no loss of clad integrity due to the postulated event, there is no release path introduced for the fission-product gases contained in the fuel cladding gap to escape.

In the event that the reactor vessel head drop results in the loss of normal vessel injection and cooling such that no water can be injected via normal paths, temporary line connections supplied from the redundant containment spray headers to the reactor vessel head will ensure that adequate coolant makeup is provided to keep the core covered. As long as the core is covered with water, core cooling will be maintained by the heat-up of the injected coolant and heat removal by nucleate boiling. At the low decay heat rate and temperature, a departure from nucleate boiling will not occur with the core covered. The coolant will remain no hotter than the 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. Note that previous studies performed for the Westinghouse two-loop design demonstrated that the fuel temperature remain at saturation conditions (WCAP-11916, "Loss of RHR Cooling While the RCS is Partially Filled"). Therefore, no subsequent damage to the fuel cladding could occur and no release of the fuel gap could ensue due to the loss of normal vessel injection. Since the containment is isolated by having the purge supply and exhaust fans off, the associated containment isolation valves closed, and the containment outer access doors closed prior to the movement of the reactor vessel head over the reactor vessel; and no fuel cladding damage that will result from this event; the reactor vessel head drop is bounded by the licensing basis fuel handling accident, which assumes the gap release of one-assembly to the environment (NRC safety evaluation dated April 2, 2004).

Operation of the RHR system post-accident would provide the suction to the containment spray system during recirculation of the coolant from the Unit 2 containment sump. Leakage from this system is generally collected and drained to the Primary Auxiliary Building (PAB) sump, which is processed through the waste disposal system. Similarly, leakage that may become an airborne source of radioactivity is collected by the PAB ventilation system and exhausted through the PAB exhaust stack to the environment. The PAB ventilation system is designed to limit offsite releases and support auxiliary building habitability during normal and accident conditions since it does have HEPA and charcoal filtration capabilities. The airborne source would be limited to the coolant concentrations at the time of the accident. The RCS has been open to atmosphere for a considerable amount of time such that the volatile radionuclides like iodine and noble gases are not present in any appreciable/measurable quantity. Since there are no fuel defects present and there is no postulated fuel damage, the activity present in the recirculated coolant is not a result of the postulated head drop but due to previous reactor operation. Therefore, the release of the activity in recirculated RCS coolant would be bounded by the allowable routine limits for effluents.

<u>Summary</u>

The reactor vessel head drop event does not result in a loss of fuel rod cladding integrity either through the impact load placed on the fuel assembly through the control drive shaft or through the loss of normal vessel injection due to the installation of TM 2005-008. Therefore, the postulated head drop for Unit 2 Refueling 27 is bounded by the licensing basis fuel handling accident, which demonstrates that the offsite consequences are well within 10 CFR 50.67 limits and the control room operator dose is less than the GDC-19 limit.

Reactivity Condition, k_{eff} (NUREG-0612 Criterion II)

NUREG-0612 Criterion II requests evaluation of damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95

The analysis of a head drop considers both a concentric impact with the head level and a concentric impact with a slight rotation of the head. These two cases result in the maximum impact force to the reactor vessel supports and nozzles. In both cases, the control rod drive shafts may be impacted. It has been shown that the result of impacting the drive shafts is buckling of the drive shafts with no change in core geometry. As such, no reactivity additions are anticipated as a result of a reactor vessel head drop, and it is only necessary to ensure $k_{eff} < 0.95$ as required by the refueling Technical Specifications.

Water Leakage and Inventory Makeup (NUREG-0612 Criterion III)

NUREG-0612 Criterion III requires evaluation of the ability to ensure that, "... damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel (Makeup water... should be from a borated water source)."

NMC's mitigating strategy for the bounding load drop scenario ensures that there is adequate makeup to keep the core covered.

No Loss of Required Safe Shutdown Functions (NUREG-0612 Criterion IV)

NUREG-0612 Criterion IV requires evaluation of damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

The bounding scenario assumes loss of normal decay heat removal. Mitigating actions taken with the use of the temporary modification to provide core cooling ensure that the safe shutdown function can be achieved.

Probabilistic Risk Analysis

An assessment of the risk of a reactor vessel head (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. The estimate considered the probability of dropping the RVH along with the Conditional Core Damage Probability if the head was to drop. For this estimate, it was assumed that the only core injection method available would be via hose connections from the containment spray system to the RVH.

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 attached piping such that normal injection methods (safety injection, residual heat removal and charging) are not available.

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.

For this assessment, a bounding drop probability of 5.6E-5 is assumed, but based upon the above discussion, it is believed to be conservative by a factor of six (6).

<u>Conditional Core Damage Probability</u>

The estimated Conditional Core Damage Probability is based upon the plant PRA model for the failure probability of both trains of containment spray. The model was adjusted to account for potential human errors that may occur due to the specific initiating event being postulated. The Conditional Core Damage Probability may consist of any of three failures: (1) Failure to initiate containment spray; (2) Failure to establish sump recirculation after draining the refueling water storage tank (RWST); (3) Equipment failure associated with the containment spray system, residual heat removal system and all support systems. The human error associated with containment sump recirculation was assumed to be bounded by the evaluation for a large loss-of-coolant accident (LOCA), which requires recirculation early in the event, and assumes a high stress level. Equipment failures were evaluated by solving the current plant PRA model. A Human Error Probability (HEP) to account for failure to initiate containment spray, when necessary, was estimated based upon specific procedures and training provided for this event.

The HEP estimate for the initiation of containment spray is based upon the manual action to start at least one train of containment spray prior to core damage. A combination of EPRI Cause-Based Decision Tree Method (CBDTM) and Technique for Human Error Rate Prediction (THERP) methods was used. The scenario evaluated starts with control room notification from the field of a dropped RVH followed by entry into the Abnormal Operating Procedure (AOP) for this event. Credit is taken for the initiation of containment spray upon verification of the event and the inability to manage core inventory through the appropriate shutdown LOCA procedures. The actions necessary to start containment spray are simple with all controls available from the control room. The analysis estimates a failure probability of 4.5E-4.

Considering the three basic failure modes discussed above, the overall Conditional Core Damage Probability was determined to be approximately 7.3E-3.

Core Damage Probability

The probability of a RVH drop times the Conditional Core Damage Probability provides the Core Damage Probability per RVH lift. Using the bounding RVH drop probability of 5.6E-5 and a Conditional Core Damage Probability of 7.3E-3, it is estimated that the Core Damage Probability is 4.1E-7 per lift. Results demonstrate that the upper boundary scenario for core damage probability is less than 1E-6 and the dose consequences are well within allowable limits.

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 head (RVH) drop event. Approval of this change to the licensing basis is requested solely for use on a one-time basis for Unit 2 RVH lifting activities upon completion of refueling activities associated with the spring 2005 refueling outage.

PBNP Unit 2 is currently in a refueling outage with the reactor vessel head removed. Replacement of the reactor vessel head is planned for May 2005. Consequently, exigent NRC approval of the proposed amendment is required 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, in accordance with the proposed amendments, presents no significant hazards. Our 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 one-time lifting of the PBNP Unit 2 reactor vessel head over a reactor vessel containing fuel assemblies. The proposed change does not significantly increase the accident initiators or precursors nor alter the design assumptions, conditions, or the manner in which the plant is operated and maintained. A reactor vessel head drop could initiate a loss of coolant accident under shutdown conditions, the consequences of which would be bounded by the licensing basis analysis for that accident. A RVH drop is of sufficiently low probability such that the probability of a loss of coolant accident is not significantly increased. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Further, 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 as analyzed in the fuel handling accident. The proposed change is consistent with safety analysis assumptions and resultant consequences. Therefore, it is concluded that 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 analysis supporting one-time lifting of the PBNP reactor vessel head over a reactor vessel containing fuel assemblies. The change does not impose any new or different requirements or eliminate any existing requirements. It does not change the design function or operation of the systems, structures or components involved. The drop of a reactor vessel head could be the initiator for a loss of coolant accident, which has been evaluated, in the existing licensing basis. An RVH drop would not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases, nor would it have been considered a design basis accident in the FSAR had it been previously identified. The changes do not alter assumptions made in the safety analysis. 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 reactor vessel head over a reactor vessel containing fuel assemblies. A plant specific risk-informed analysis was performed that suggests no significant reduction in margin to safety would occur. All the recommended margins regarding containment building polar crane loading and administrative controls 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. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

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 does not result in a significant hazards determination.

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 § 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 § 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 § 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, 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 requires a license amendment pursuant to 10 CFR 50.90. Because this analysis supports reactor vessel head 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 two commitments as follows:

1. NMC will establish administrative controls for the RVH lift as described in Enclosure 2.

2. NMC will continue assessment of the RVH drop event for future RVH lifts and submit the associated analyses for NRC review, as appropriate.

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 including Temporary Modification 2005-008,
- Developing and implementing one-time 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 screening and evaluation for the replacement RVH,
- Reassessing the RVH drop event (Westinghouse letter WEP-05-161).

PBNP Unit 2 is currently in a refueling outage. NMC recognizes that the condition applies to movement of the RVH during the outage. Therefore, NMC requests NRC approval of this amendment for Unit 2 on an exigent basis in accordance with 10 CFR 50.91(a)(6).

8.0 REFERENCES

1. Letter from Wisconsin Electric Power Company to USNRC, dated November 22, 1982, "Submittal of Outstanding Response Items, NUREG-0612– Control of Heavy Loads"

2. Letter from Nuclear Management Company, LLC to USNRC, dated April 15, 2005, "Response to Request for Additional Information, NUREG-0612, Control of Heavy Loads, Reactor Vessel Head Drop Analysis"

3. Letter from Nuclear Management Company, LLC to USNRC dated April 20, 2005, "Response to Request for Additional Information – Revision 1, NUREG-0612, Control of Heavy Loads, Reactor Vessel Head Drop Analysis"

4. Letter from Nuclear Management Company, LLC to USNRC dated April 29, 2005, "Request for Review of Heavy Load Analysis"

5. Letter from Nuclear Management Company, LLC to USNRC dated May 8, 2005, "Resolution of Safety-Related Questions Regarding Unit 2 Reactor Vessel Head Lift"

6. Letter from Westinghouse Electric Company to Nuclear Management Company, LLC dated May 11, 2005, "Assessment of Reactor Vessel Head Drop"

7. WCAP-11916, "Loss of RHR Cooling While the RCS is Partially Filled"

ENCLOSURE 2

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

1. Protected Equipment:
   - Both trains of residual heat removal, safety injection and containment spray
   - All three charging pumps
   - Two emergency diesel generators (Unit 2 A and B train)
   - Containment polar crane power supply

2. A dedicated operator will be staged in the vicinity of tagged out, but available, equipment with direct radio communications to the control room.

3. Unit 2 containment and the primary auxiliary building will be evacuated of personnel who are not essential to the reactor vessel head lift during the period of time the reactor vessel head is over the vessel.

4. A walkdown of containment Elevations 66', 21' (head lay down area), and 8' will be conducted prior to commencement of the lift to ensure there is no loose debris that could result in clogging of the containment sump.

5. The containment sump screen shall be installed.

6. A minimum borated water volume of 318,000 gallons shall be available as a suction source to required equipment.

7. 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" over the reactor vessel flange.

8. The maximum allowable lift height for the reactor vessel head shall not be exceeded.

9. A temporary modification will be installed that provides two redundant water supply connections to the reactor vessel head when the reactor vessel head is suspended greater than 24" over the reactor vessel flange (below 24" the temporary modification will be removed to provide a reactor vessel head vent path).

10. The temporary water supply connections to the reactor vessel head will be supplied from redundant spray headers.

11. Both personnel airlocks will be closed with at least one door and interlocks will be installed on the airlocks.

12. Both trains of containment spray will be available.

ENCLOSURE 3

WESTINGHOUSE REPORT
"ASSESSMENT OF REACTOR VESSEL HEAD DROP
POINT BEACH UNIT 1 AND UNIT 2"
(LTR-RCDA-05-428, REVISION 1, DATED MAY 13, 2005)

PROPRIETARY

WESTINGHOUSE AUTHORIZATION LETTER
AFFIDAVIT
PROPRIETARY INFORMATION NOTICE
COPYRIGHT NOTICE

(29 pages follow)

CAW-05-1996

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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. A. Gresham

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

Sworn to and subscribed
before me this 13th day
of May, 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 Manager, 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 in LTR-RCDA-05-428, Rev. 1, "Assessment of Reactor Vessel Head Drop - Point Beach Unit 1 and Unit 2" (Proprietary) dated May 13, 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 Units 1 and 2 is expected to be applicable for other licensee submittals 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) Confirm the accuracy of existing head drop analyses and adjust for the different weights of new replacement reactor vessel head and head assembly upgrade packages.

(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 LTR-RCDA-05-428, Rev. 1, "Assessment of Reactor Vessel Head Drop – Point Beach Unit 1 and Unit 2" (Proprietary)

Non-Proprietary copies of LTR-RCDA-05-428, Rev. 1 are not provided.

Also enclosed is Westinghouse authorization letter CAW-05-1996 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As LTR-RCDA-05-428, 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-05-1996 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.