NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

April 29, 2005

NRC 2005-0055
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
Request for Review of Heavy Load Analysis

Pursuant to 10 CFR 50.91(a)(6), Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed exigent license amendment for the Point Beach Nuclear Plant (PBNP). This proposed change is necessary to support a change to the PBNP Final Safety Analysis Report (FSAR) regarding control of heavy loads and to ensure that effective mitigation strategies are enacted and administratively controlled in the event of a heavy load handling event.

Enclosed for Commission review and approval is the PBNP evaluation for control of heavy loads. Enclosure 1 provides a description, justification, and significant hazards determination for the reactor vessel head (RVH) drop event analysis. Enclosure 2 contains the proposed FSAR revision regarding control of heavy loads, which includes an analysis of a postulated RVH drop event. The NMC evaluation of this event identifies numerous systems and components that must be available to mitigate the consequences of the postulated event. The proposed change provides additional assurances that these systems and components will be available during heavy load handling evolutions.

NMC requests exigent approval of this proposed license amendment for PBNP Unit 2 (Docket 50-301, License DPR-27) by May 10, 2005, with the approved amendment being implemented within seven days of approval. Exigent approval of the proposed license amendment for PBNP Unit 1 is not being requested.

A copy of this application, with attachments, is being provided to the designated representative of the State of Wisconsin.

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Summary of Commitments:

This letter contains three new commitments:

1. Administrative controls for reactor vessel head lifts will be established in the PBNP Technical Requirements Manual, as described in Table 1 of Enclosure 1.

2. Reactor vessel head lifting activities will not commence until greater than 65 hours have elapsed since reactor shutdown.

3. PBNP will not use the PaR (Programmed & Remote Systems, Inc.) device over a loaded core.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on April 29, 2005.

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

Enclosures

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

DESCRIPTION OF PROPOSED CHANGE

1.0 DESCRIPTION OF PROPOSED CHANGE

Nuclear Management Company, LLC (NMC) requests to amend Operating Licenses DPR-24 and DPR-27 for the Point Beach Nuclear Plant (PBNP) Units 1 and 2. The proposed amendment requests NRC approval to update the Final Safety Analysis Report (FSAR) to reflect inclusion of a reactor vessel head (RVH) drop event. Exigent review and approval is requested for the Unit 2 (DPR-27) Operating License Amendment. Exigent approval of the proposed license amendment for PBNP Unit 1 is not being requested.

2.0 PROPOSED CHANGE

NMC proposes revising the FSAR to incorporate the RVH drop event 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 (subsequently numbered as GL 80-113 and hereinafter referred to as GL 80-113), as supplemented by Generic Letter 81-07. The proposed FSAR appendix that incorporates the RVH drop analysis is provided as Enclosure 2.

The heavy load analysis is based upon a plant specific risk-informed evaluation that continues to demonstrate the low probability of occurrence of a RVH drop as originally evaluated. For the specific case of a RVH lift, this evaluation 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 PBNP's "defense in depth" strategy to mitigate the consequences of this low probability event should it occur. The required administrative controls for equipment in support of the heavy load analysis for a reactor vessel head lift will be established in the PBNP Technical Requirements Manual (TRM) during implementation of the proposed change. Other controls needed to fully implement this strategy will be established in appropriate owner controlled documents.

Controls will be in effect whenever the RVH is not fully resting on the reactor vessel and any part of the head is over a reactor vessel containing fuel assemblies. The head is considered to no longer be over a reactor vessel once the entire head has been moved past the railing along the refueling cavity and no part of the head is directly above any portion of the reactor vessel.

The administrative controls that address equipment requirements for potential event mitigation will include the requirements in Table 1 of this Enclosure. In addition to the equipment identified for inclusion into the Technical Requirements Manual (TRM), both trains of residual heat removal (RHR), including associated RHR heat exchangers are required to be operable in accordance with Technical Specification 3.9.5 or Technical Specification 3.4.8.

If one or more of the required administrative controls is not met, immediate action is required to suspend lifting of a RVH over a reactor vessel containing fuel assemblies. Performance of this action shall not preclude completion of movement of the RVH to a safe position.

The administrative controls contained in the proposed TRM will be established in appropriate owner controlled documents in accordance with existing plant processes.

3.0 BACKGROUND

On December 22, 1980, NRC issued as GL 80-113, which was supplemented on February 3, 1981, by Generic Letter 81-07 regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." These letters discussed a two-phase set of investigations and submittals by licensees. Phase I addressed the load handling equipment within the scope of NUREG-0612 and described 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 (WE), 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.

The November 22, 1982, letter (Reference 1) 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."

On March 27, 1984, the NRC issued a safety evaluation that addressed the NUREG-0612 Phase I actions taken and planned by WE. The safety evaluation permitted deferral of the annual inspections of the containment polar cranes if the applicable prior to use inspections were performed. 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 WE via a submittal to the NRC dated March 16, 1984, as modified in a September 25, 1984, submittal. In response, the NRC issued Facility Operating License Amendments 91 and 95 for Units 1 and 2, respectively, on April 8, 1985.

On April 6, 2005, while collecting documents associated with the new RVH to be provided to the NRC for inspection purposes, it was identified that the 10 CFR 50.59 screenings and evaluations performed for the new RVH did not recognize that the RVH drop event was within the PBNP licensing basis. Failure to recognize that the RVH drop analysis was within the PBNP licensing basis occurred because the FSAR had not been updated with the analysis results that had been previously reported to the Commission in 1982 (Reference 1). A corrective action program document (CAP063450) was initiated on April 7, 2005, the NRC senior resident inspector was informed, and a dialog on this matter commenced between NMC and NRC representatives.

Following several discussions between NMC and NRC representatives, a request for information was developed by NRC that was communicated to NMC. This request for information was formalized via letter from NRC to NMC that was received on April 15, 2005. In the interim, however, numerous communications between NMC and NRC representatives occurred and a significant amount of information was provided to the NRC via the resident inspectors.

The NRC request for additional information was primarily associated with heavy load handling activities that involved the existing RVH. These questions were responded to by NMC via letter to the NRC dated April 15, 2005 (Reference 2). This letter, and responses to questions posed by the NRC request for additional information, established one-time regulatory commitments associated with the lift of the Unit 2 existing RVH. During a telephone conversation conducted on April 16, 2005, NRC concurrence with NMC mitigative strategy and commitments associated with this one-time RVH lift was obtained. The existing Unit 2 RVH was successfully removed from the vessel and transported to its laydown area.

Reference (3) was submitted by NMC to reflect revision of two administrative controls, report successful completion of the RVH lift, and to provide a commitment that the replacement RVH would not be lifted over the reactor core containing fuel assemblies until the outstanding issues associated with the RVH drop event have been resolved.

In recognition of the consequences that could result from a heavy load handling event, NMC has identified the need to enhance the PBNP heavy load handling guidance to include effective mitigating strategies and methods to implement these strategies during specific heavy load handling evolutions.

A 10 CFR 50.59 evaluation of the control of heavy loads was performed by NMC in order to incorporate NUREG-0612 licensing basis correspondence into the FSAR. The evaluation concluded that incorporation of the analysis into the FSAR would create a possibility for an accident of a different type than any previously evaluated in the PBNP final safety analysis report. This conclusion was reached because the RVH drop analysis had not been previously reviewed by the Commission. Although this postulated event 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). Had the action been taken to incorporate the information into the PBNP FSAR in 1982, the corresponding 10 CFR 50.59 safety evaluation would have likely concluded that the analysis represented an unreviewed safety question that would have required review by the Commission in accordance with the requirements of the rule at that time.

4.0 TECHNICAL ANALYSIS

The heavy load drop evaluation discussed herein provides technical justification for creating a new appendix to the FSAR to address the control of heavy loads. The proposed appendix is an abridgement of requirements and commitments established pursuant to NUREG-0612. The scope of the proposed new appendix includes Phase I and Phase II submittals pursuant to the GL 80-113, and Generic Letter 81-07, however, only the heavy load drop event is being submitted for NRC review.

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 inadvertent dropping of a postulated heavy load produce doses that are well within 10 CFR 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 keff 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 Criteria I, II, and III were not fully developed at that time.

The drop of a RVH is expected to bound all other heavy loads that may need to be handled above the reactor vessel. Specific heavy loads that would generally be bounded because of their lower mass are the reactor upper internals, the PaR (Programmed & Remote Systems, Inc) device that is used to inspect the interior walls of the reactor vessel, and a reactor coolant pump (RCP) motor. Since the PaR device can be manipulated above a loaded core with no intervening internals, it was determined that if the PaR device is to be used above a core, the refueling boron concentration would need to be high enough to ensure that k_{eff} was maintained at 0.90 or less. 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 RVH and internals) would preclude contact with the fuel, or because administrative controls preclude handling the load over an exposed, and therefore, unprotected core.

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

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. Detailed information associated with PBNP conformance to Phase 1 guidance is contained in References (2) and (3).

Description of RVH Lifts

Maintenance procedures provide instructions to prepare for and to perform RVH removal, removal of the upper internals and subsequent replacement of the upper internals and RVH. The RVH lift rig is inspected and installed on the RVH and a pre-lift inspection is conducted. The crane operator then aligns the containment polar crane with the lift rig and engages the crane hook to the lift rig. A second individual is assigned to operate the disconnect switch, if required, while a third person is assigned to perform initial RVH lift verifications, and a fourth person performs the RVH visual inspections.

The initial lift raises the RVH just clear of the vessel flange and then continues up to approximately one foot above the flange to ensure the load is level and to perform needed visual and ultrasonic inspections.

The RVH lift continues to a height of approximately four feet above the vessel flange where additional inspections are performed to ensure there are no control rod drives moving with the RVH. Following this, the lift continues up to clear containment Elevation 66' and the RVH is transported to its support stand in its assigned laydown area in accordance with its approved safe load path. Based upon prior operating experience, the total period of time where the lift is over the reactor vessel is approximately one hour.

During RVH installation, the initial lift brings the RVH up just clear of its support stand and continues the lift up to approximately one foot (1') in order to ensure the load is level. Following this, the RVH lift is then continued up to sufficiently clear containment Elevation 66'. The RVH is then transported directly to the vessel in accordance with its approved safe load path. The RVH is then aligned with the reactor guide studs and lowered onto the vessel flange. As with RVH removal, the total period of time where the lift is over the reactor vessel is approximately one hour.

Potential Event Mitigation Contingencies

The heavy loads evaluation demonstrates that existing controls ensure that the probability of a head drop is extremely low. Additional controls allow appropriate action to be taken to mitigate and manage the adverse consequences should such an unlikely event occur. The following administrative controls were evaluated for use during handling of the reactor vessel head at PBNP. Of the controls evaluated, those listed in Table 1 of this enclosure will be established in the PBNP TRM. Appropriate procedures to implement the controls in the proposed TRM will be established in accordance with existing plant processes.

The following provides a discussion of the system and component-related elements to be addressed that are important to the mitigating strategies necessary to respond to the postulated load drop event. These requirements apply during the time that the RVH is being lifted over the reactor vessel with fuel assemblies in the vessel:

1. To ensure that redundant flow paths are available for injection of water, both trains of RHR including associated RHR heat exchangers, one train of containment spray, one train of safety injection, one charging pump and the standby emergency power sources aligned to the affected units safeguards buses will be available. Additionally, this equipment along with the polar crane power supply will be protected.

   "Protecting" the noted equipment provides a defense-in-depth approach to ensuring the availability of equipment critical to mitigating the consequences of a RVH drop. The available trains of RHR are required to be operable per PBNP Technical Specifications. Sodium hydroxide is not required to be available to the train of containment spray.

2. To ensure an adequate borated water volume exists to flood containment to provide lower vessel head cooling, a minimum volume shall be determined and made available as a suction source to the required equipment.

3. To ensure that any loose material that could result in containment sump blockage is not present, containment sump screens will be installed as required in the proposed TRM change. Additionally, a walk down of Elevation 8' of containment will be completed prior to the reactor vessel head lift to assure material is controlled.

   Containment spray is only used to flood containment to provide lower vessel head cooling should sump recirculation not be available. Due to the RCS being depressurized, temperature <140°F and a direct path to containment Sump B due to the RVH dropping, concerns for pipe whip or high pressure energy release are not applicable. Therefore, material contained on the upper elevations of containment would not be a concern. Material remaining on Elevation 8' will be controlled to prevent blockage of the containment Sump B screens should sump recirculation be required.

4. Containment access will be limited to those individuals involved in or supporting reactor vessel head lift activities. Limiting the number of personnel in containment during the RVH lift activities will facilitate the timely evacuation of containment if a RVH drop occurs.

5. To facilitate containment closure capability, containment penetrations required for containment closure will be closed as required in the proposed TRM. Maintaining all personnel airlock bulkhead doors closed would present an industrial safety hazard to personnel due to the differential pressure. Therefore, at least one personnel airlock must remain open during RVH lift activities. A dedicated individual will be stationed

at the open personnel airlock to shut the bulkhead door following the evacuation of personnel from containment.

The containment purge supply and exhaust valves are automatically closed on a containment ventilation isolation signal that would be generated by a high radiation condition in containment, or upon receipt of a manual containment isolation signal. In addition, procedural guidance instructs operators to manually initiate containment isolation. This will ensure that the containment will be closed to limit the release of fission products.

Mitigation Contingency Strategy Evaluation

The worst case scenario for dropping the RVH onto the reactor vessel is that severe damage will occur to piping attached to the reactor coolant system (RCS) and the ability to remove decay heat by normal means may be lost. This is because of the postulated failure of the vessel support columns, causing severe damage to safety injection lines and loop piping. The RCS level in the reactor vessel will drop to the RCS loop elevation. There is not a risk of core ejection due to the installation of the upper internals. Therefore, the reactor vessel remains intact and depressurized.

The mitigating strategy will attempt to restore core cooling using charging, residual heat removal (RHR), and safety injection system flow paths. If resultant damage is not severe (not worst case) then existing Abnormal Operating Procedures (AOPs) and Shutdown Emergency Procedures (SEPs) will mitigate the event by injection into the core and eventual sump recirculation. Assuming the worst-case scenario, the flooding of containment would be implemented via Severe Accident Management Guidelines (SAMGs).

The following assumptions were used to develop the mitigating strategy for this event:

- RVH remains on the vessel
- Core deluge lines severed (worst case)
- RCS loop injection lines from charging, residual heat removal, and safety injection systems severed (worst case)
- RCS loop nozzles remain attached
- RCS loss of coolant accident (LOCA) occurs
- Core exit thermocouple temperature indication is not available
- Possibility of no reactor vessel level indication system (RVLIS) indication after RVH drop

In the event of a RVH drop, operators will first assess damage using indications available to them in the control room and/or reports from personnel who witnessed the event. Indications available in the control room would include reactor vessel level instrumentation, containment sump level changes, and changes in containment radiation levels. If no immediate damage is assessed, operators would take action to ensure personnel safety and stabilize the plant using existing procedural guidance.

If entry conditions were met, the Operations staff would enter SEP-2, "Shutdown LOCA Analysis." Upon entry to SEP-2, the procedural flow path would 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 available plant systems. These include the charging, safety injection and residual heat removal systems and containment sump recirculation capability.

While implementing SEP-2.3, normal parameters used as an indication of adequate core cooling may not exist, such as core exit thermocouple and reactor vessel level. However, indirect indications of core cooling could be used. Erratic source range indications would provide an indication of lowering, or voiding, in the reactor vessel downcomer region. Additionally, ECA-1.1, "Loss of Containment Sump Recirculation," Figure 1, "Minimum Injection Flow vs. Time after Shutdown," is available to determine minimum injection needed to maintain core cooling.

ECA-1.1, "Loss of Containment Sump Recirculation," provides directions to restore containment sump recirculation capability, extending the availability of the refueling water storage tank (RWST) volume by adding makeup and reducing outflow, and to depressurize the RCS to minimize break flow. The procedure provides multiple methods for ensuring there is adequate cooling water provided. These methods include refilling the RWST by (1) using the reactor makeup water system; (2) from the fuel transfer canal by using the holdup tank recirculating pump; (3) from the opposite unit's RWST using the refueling water circulating pump, and (4) from the opposite unit's RWST using the boric acid storage tank to the safety injection pump suction.

If all attempts to establish core cooling using SEP-2.3 fail, then applicable Severe Accident Management Guidelines (SAMGs) will be implemented per Technical Support Center (TSC) direction. The control room could be directed by the TSC to implement SACRG-2, "Severe Accident Control Room Guideline for Transients after the TSC is Functional." The TSC would identify applicable Severe Accident Guidelines (SAGs) using the TSC Diagnostic Flow Chart (DFC). Applicable SAMG procedures include SACRG-2, SAG-3, "Inject into the RCS," and SAG-4, "Inject into Containment."

These SAMG guidelines direct the control room to follow TSC directions for component operation, establish alternate core cooling pathways, and, if needed, raise containment water level using borated water sources to establish external vessel cooling. This mitigative strategy consists of filling the containment sump with water, and, if necessary, continuing injection of water via containment spray, RHR system, safety injection system and the RWST gravity drain to ensure adequate core cooling is provided.

While implementing the above-mentioned procedures, replenishing RWST inventory is available in ECA-1.1, "Loss of Containment Sump Recirculation," Attachment A. If necessary, another potential water source to the RCS/containment would be the safety injection pump(s) available to refill the SI accumulators, which drain to the reactor vessel.

In addition to ensuring that redundant trains of cooling water are available and containment closure can be accomplished, the volume and diversity of water sources and the redundancy of trains form a vital facet of our mitigative strategy.

Reactor Vessel Head Drop Assessment

The PBNP reactor vessel head (RVH) drop analysis is summarized in Westinghouse analysis WEP-82-584, "Reactor Vessel Head Drop Analysis." Reference (1) summarized the results of this 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."

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.

Provided that the resultant impact forces at the vessel supports of the replacement head assembly drop are equivalent or less than those of the original head assembly drop, the postulated damage is expected to be the same or less. The maximum allowable lift height for the RVH will be specified in the TRM.

Dose Assessment (NUREG-0612 Criterion I)

NUREG-0612, Criterion I states: "Releases of radioactive material that may result from damage to spent fuel based on calculations involving inadvertent dropping of a postulated heavy load produce doses that are well within 10 CFR 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)."

NUREG-0612 does not have a specific expectation that the licensee must also demonstrate that the control room dose limits of General Design Criteria 19 (GDC 19) can be met due to the releases of radioactive material that may result from damage to spent fuel due to a postulated heavy load drop. However, a conclusion regarding the dose consequences to the control room operator is provided in addition to the offsite impacts.

For the purpose of evaluating Criterion I of NUREG-0612, Section 5.1.3(2), the radiological event scenario for a heavy load drop was assumed to be similar to the current design and licensing basis fuel handling accident. The dose estimation proposed in NUREG-0612 is based on Regulatory Guide 1.25; however, since the issuance of NUREG-0612, an alternative source term (AST) has been promulgated by 10 CFR 50.67. Regulatory Guide (RG) 1.183 provides the acceptable methodology for application of the AST and is the licensing basis for the PBNP fuel handling accident. (NRC Safety Evaluation Report dated April 2, 2004) Therefore, the analytical method for this postulated event is based upon the current licensing basis fuel handling accident source term to determine the release timing and fractional releases, as well as, the appropriate dose conversion factors.

A bounding assessment was performed to calculate the offsite radiological consequences due to a postulated fuel gap release resulting from a heavy load drop (e.g., reactor vessel head) on the reactor vessel for various times post-accident. The intent of the assessment was to determine under a worst case gap release (i.e., 100% of the fuel assemblies affected) the required containment initial access configuration necessary to demonstrate that mitigative actions could be taken to close containment in a timeframe commensurate with ensuring that the offsite doses would remain within a "small fraction" of the regulatory limit.

The PBNP Unit 2 containment has lower and upper personnel locks. Each personnel airlock contains two doors that are interlocked while in MODES 1, 2, 3 and 4, such that only one door can be opened at a time. During cold shutdown and refueling, MODES 5 and 6, the doors are no longer required to be interlocked so both doors can be opened for ease of access to containment.

The containment purge ventilation system can be operated with either one fan (12,500 cfm nominal rating) or two fans (25,000 cfm nominal rating) to provide personnel comfort. The containment purge system also has charcoal and HEPA filtration capabilities. When the purge system is running, the release point for the containment air is the purge vent stack. If the purge system is turned off, natural ventilation is the primary driving force for air release from containment.

Based on the configuration status of the containment access doors and purge ventilation system, the outflow of air from containment will vary. Therefore, two radiological consequence case scenarios were considered for the heavy load drop assessment: Full Release Flow and Nominal Release Flow.

Full Release Flow – 12,500 cfm

The Full Release Flow case scenario assumes that at the onset of the event all personnel airlocks are open (i.e., both doors in both personnel airlocks) and only one containment purge fan running. Once purge is isolated due to a high-radiation signal, natural convection is established and the release is from an open access hatch. Under this scenario, the outflow is assumed to be 12,500 cfm for the duration of this scenario.

Nominal Release Flow – 10 cfm

The Nominal Release Flow case scenario assumes that at the onset of the event only one door in one of the two personnel locks of the containment building is closed and the remaining doors are open. In addition, it also assumes that one purge fan is running until it is isolated due to a high-radiation signal. The closure of one door cuts off the airflow path from one personnel lock to the other in the containment building. In this configuration, the containment building atmosphere would be in a neutral condition and the airflow rate would be dependent on a pressure differential that may develop between the inside/outside containment environments. This pressure differential could be due to environmental conditions or to heat generation sources in the containment building. It is expected that a pressure differential that may be caused by heat generation sources in the containment building will be small and slowly occurring due to the mass of the materials in the containment building and their ability to absorb heat. Therefore, an initial flow of 12,500 cfm through the containment purge system is assumed until it is isolated, then a nominal flow of 10 cfm is assumed for the duration of the event. Credit for containment purge system filtration is applied to the portion of the release from the containment purge stack.

Source Term

The heavy load drop radiological consequence assessment makes the following adjustments to the licensing basis fuel handling accident scenario to bound this evaluation.

1. All fuel assemblies are affected with a gap activity makeup the same as the licensing basis fuel handling accident.

2. The maximum core radial peaking factor of 1.8 is not applied to the source term. The peaking factor is applied to events that do not involve the entire core to account for differences in power level across the core such that a limiting release is predicted. Since all assemblies are assumed to be damaged and the total core inventory is used to estimate the release, the maximum core radial peaking factor of 1.8 does not apply.

3. No credit for scrubbing of iodine as it evolves out of the reactor coolant is taken.

4. A portion of the containment net free volume is credited for dilution of the activity released from the fuel. See assumptions below.

5. The containment purge ventilation filter system is credited for the release through the purge stack prior to isolation for the Nominal Release Flow. See the assumptions below.

Dose Determination

The concentration in the containment building atmosphere was determined by subtracting the amount of activity exiting the containment building through an open personnel airlock from the amount of activity entering the containment building atmosphere due to the damaged fuel assemblies. The activity enters and exits the containment building atmosphere at constant rates. The rate of activity entering the containment building atmosphere from the damaged fuel assemblies is based on the fuel handling accident two-hour release duration. The amount of activity exiting the containment building is based on the airflow through the open personnel lock and the concentration of activity in the containment building atmosphere. The released activity is then multiplied by the atmospheric dispersion factor to determine the concentration at the offsite location. Decay and plume deposition are ignored. For internal dose, the concentration of each radionuclide in the source term is multiplied by the breathing rate and the appropriate CEDE (Committed Effective Dose Equivalent) conversion factor. The external dose is simply the concentration multiplied by the EDE (Effective Dose Equivalent) conversion factor. Total dose (i.e., the Total Effective Dose Equivalent, or TEDE) is the sum of the CEDE and EDE doses. The doses are determined at one-minute intervals.

Assumptions

1. Containment Purge isolates within 13 seconds of radioactivity entering the sampling point for the purge radiation monitoring system.

   Basis: The setpoint of the containment ventilation radiation monitors, RE-212 or RE-305, would be exceeded once the activity reaches the detector, since the setpoint is based on meeting 10 CFR 20 release rate requirements. Since the entire core is assumed to be damaged at the onset of the event, the gap release would create airborne concentrations at the start of the event that will exceed the monitor setpoint. Additionally, in 1988 an estimate of the time required for radioactivity to migrate from the refueling cavity to a personnel hatch was derived from calculations and actual measurements. The results of this activity estimated that the total time it would take for gases to reach the purge radiation monitor (1 minute, 25 seconds) and to trip the containment purge system is 1 minute, 39 seconds. The time to travel from the refueling cavity pool surface to the containment equipment hatch is 6 minutes, 45 seconds, and 8 minutes, 39 seconds to the Elevation 66' personnel airlock. The filtration of the activity released via this pathway prior to isolation is credited for the nominal release flow case only. The charcoal filter is assumed to be 90% efficient for iodine in all chemical forms.

2. It is assumed that the activity released from the fuel is instantaneously mixed in the containment atmosphere, such that the containment atmosphere is a homogenous mixture at all times. However, only one-half the volume of the containment is credited for dilution.

   Basis: Assuming that the released activity is dispersed instantaneously is a simplification of the distribution of the radioactivity released to the containment building atmosphere. Diffusion time of the activity released is ignored and air that could be trapped in stagnant areas of the containment equipment cubicles is discounted.

3. It is assumed that the heavy load drop does not occur until the reactor has been shutdown for 65 hours.

   Basis: The licensing basis fuel handling accident assumes that the reactor has been shutdown for 65 hours. This source term is used to determine the radiological consequences following a heavy load drop.

Acceptance Criteria

NUREG-0612 provides that for a postulated heavy load drop the radiological consequences are to be a small fraction of the 10 CFR 100 limits, where a small fraction is determined to represent 25% of the limit. Since 10 CFR 50.67 serves as the basis for the source term, the acceptance criteria for this analysis provided in RG 1.183 is 6.3 rem TEDE for the worst two-hour duration of the event for the exclusion area boundary (EAB) and the low population zone (LPZ). The offsite limit provided in 10 CFR 50.67 is 25 rem TEDE.

Conclusion

The table below provides conservative approximate times at which the dose would reach the dose acceptance criteria (6.3 rem TEDE) and the 10 CFR 50.67 limit (25 rem TEDE) at the EAB and the LPZ post-heavy load drop on the reactor vessel.

It should be noted that the results of this bounding assessment are very conservative due to the assumptions made regarding the magnitude of the source term (all fuel assemblies damaged), and the atmospheric conditions (95th percentile).

Scenario	Acceptance Criteria 6.3 rem TEDE		10 CFR 50.67 Limit 25 rem TEDE	
	EAB	LPZ	EAB	LPZ
Nominal Release Flow (10 cfm)	60 min	>60 min	> 60 min	> 60 min
Full Release Flow (12,500 cfm)	1 min	7 min	3 min	15 min

Although the activity released to the environment for the heavy load drop accident is significantly larger than that assumed for the fuel handling accident, the dose to the control room operator can be maintained below the GDC-19 limit of 5 rem TEDE. Based upon a comparison of the nominal flow release case for the heavy load drop to the licensing basis fuel handling accident, the amount of activity released to the environment at 1-hour, post-heavy load drop is approximately 10 times higher than the fuel handling accident. The dose to the control room operator due to this increase in activity can be mitigated by the ingestion of potassium iodide (KI), which provides a factor of 10 reduction in the internal dose due to the radioisotopes of iodine. The use of KI is currently integrated in the Emergency Plan procedures. Since the iodine internal dose (CEDE) is the primary contributor to the TEDE dose, reducing its contribution provides assurance that the operator's dose is limited.

Based on the above results, at least one personnel airlock will be closed during RVH lifts and containment closure will be accomplished within 60 minutes of a reactor vessel head drop to ensure adequate protection is provided for public health and safety.

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

NUREG-0612, Criterion II states: "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 keff is larger than 0.95."

The analyses of a head drop consider 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 and that the control rod and fuel elements remain unaffected (i.e., no change in core geometry). As such, no reactivity additions are anticipated because of a reactor vessel head drop, and it is not necessary to ensure a k_{eff} other than as required by the Technical Specifications.

A drop of the upper internals is considered bounded by a drop of the reactor vessel head for the purposes of kinetic energy transfer to the reactor vessel. Due to the close fit between the upper internals package and the core barrel, it is not possible to drop the upper internals onto the core without the internals being essentially level. The close fit would also tend to cause a "dashpot" effect to limit the velocity of the drop as it approached the top of the core. The kinetic energy of the drop would be absorbed by a combination of the fuel assembly hold down springs (a cushioning effect transferring some impact force to the vessel via the lower core plate and core barrel load path) and the internals impacting the support ledge in the core barrel. Therefore, the geometry of the core would not be affected, and the load on the fuel assemblies would be limited by the deflection of the hold down springs. Therefore, it is not necessary to ensure a k_{eff} other than as required by the Technical Specifications for an upper internals lift.

Existing heavy load handling controls preclude the lifting of an RCP motor over an exposed core. This precludes damage to irradiated fuel assemblies or alteration of core geometry from a postulated drop of an RCP motor. Therefore, it is not necessary to ensure a k_{eff} other than as required by the Technical Specifications for a RCP motor lift.

A drop of the PaR device over a loaded core could result in damage to the fuel. Therefore, PBNP will not use the PaR device over a loaded core.

Conclusion

Criterion II is met for the reactor coolant pump motor lift and the RVH lift in accordance with the plant's design features.

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

NUREG-0612 Criterion III requires evaluation of the ability to ensure that, "...damage following inadvertent 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)."

Westinghouse letter WEP-82-584, dated November 15, 1982, establishes the postulated sequence and magnitude of damage that could be expected to occur due to a RVH drop. The limiting case postulated drop results in damage such that the RCS loop piping and SI injection piping cannot be relied upon for decay heat removal, the reactor vessel support and support structure is damaged and the reactor vessel drops approximately 10.8 inches. The analysis does not explicitly establish any additional concrete failure that would result in the vessel dropping further than the stated 10.8 inches.

To evaluate Criterion III further, two different cases were considered:

<u>Case 1</u>. The vessel drop is limited to the stated 10.8 inches, and the only damaged vessel penetrations are for the RCS piping and SI injection. Under these conditions, due to the relative elevations of these vessel penetrations being above the core, the core is not expected to be uncovered due to leakage, and Criterion III would be met.

<u>Case 2</u>. This case assumes that the ability to provide makeup water to the core exists regardless of the damage to the reactor vessel. It is assumed that additional penetrations are damaged, such that leakage exists. This could be due to other damage not explicitly identified in the 1982 analysis, but postulated as a bounding condition. The specific penetrations to be considered are the 36 penetrations on the bottom of the vessel for bottom-mounted instrumentation (BMI). Due to the flexible nature of the BMI and the distance below the vessel that is available for downward motion (beyond the 10.8 inches) it is unlikely that any of the BMI would fail. However, a conservative bounding condition has been assumed that all 36 BMI do fail. The BMI have a nozzle inner diameter of 0.39 inches each and the combined postulated failure of all 36 BMI's could result in an upper bounded leak rate of approximately 300 gpm if all failed based on a static head of standing water as transmitted to the bottom of the reactor vessel nozzles.

This postulated bounding leak rate is well within the capabilities of the makeup sources available in our mitigating strategy to preclude core uncovery, thereby satisfying Criterion III. This is considered a bounding condition for leakage in that makeup capabilities can withstand and assumes the ability to makeup to the core. If makeup capability to the core does not exist, the mitigation strategies imparted via Severe Accident Management Guidelines (SAMG) guidelines would take precedence.

<u>Conclusion</u>

Criterion III can be satisfied for the bounding condition of severance of all 36 BMI.

Probabilistic Risk Analysis

An assessment of the risk of a reactor vessel head (RVH) drop event was performed in 1982. That assessment concluded that the upper bounding probability of a RVH drop event was approximately 5E-5 per lift. A new assessment was performed using current probabilistic risk assessment (PRA) methods and plant specific input data for the frequency of lifts and failures. The methodology used and results of this assessment were submitted to the NRC via References (2) and (3).

Conclusion

As documented in References (2) and (3), the probability of an RVH drop during a crane lift above the reactor vessel at PBNP is bounded by the upper value of 5.6E-5 per lift provided in NUREG-1774.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

NMC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   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. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an 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. The proposed change is consistent with safety analysis assumptions and resultant consequences.

   Therefore, this change does not significantly increase the probability of occurrence of an accident previously evaluated.

2. Does the proposed change 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 lifting of the PBNP reactor vessel head over a reactor vessel containing fuel assemblies. The changes do not impose any new or different requirements or eliminate any existing requirements. The safe handling of heavy loads was previously evaluated in NRC Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612," dated June 28, 1985. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve 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. 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.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the above, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.82(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements

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 GL 80-113, which was supplemented on February 3, 1981 (Generic Letter 81-07) regarding NUREG-0612. PBNP responses to these generic communications were not incorporated into the FSAR as required by 10 CFR 50.71(e) as previously discussed.

NMC concludes that the necessary update to the PBNP FSAR with the heavy load analysis requires a license amendment pursuant to 10 CFR 50.90. Because no reactor vessel head lift over a reactor vessel containing fuel assemblies may occur prior to NRC approval of this license amendment, exigent approval is required to prevent unnecessary 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 new commitments as follows:

1. Administrative controls for reactor vessel head lifts will be established in the PBNP Technical Requirements Manual, as described in Table 1 of Enclosure 1.

2. Reactor vessel head lifting activities will not commence until greater than 65 hours have elapsed since reactor shutdown.

3. PBNP will not use the PaR (Programmed & Remote Systems, Inc.) device over a loaded core.

5.4 Impact on Previous Submittals

NMC has reviewed the proposed change and has determined there is no impact on any previous license amendment request submittals awaiting NRC approval.

5.5 Schedule Requirements

NMC requests approval of this license amendment by May 10, 2005.

6.0 ENVIRONMENTAL EVALUATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative 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

7.1 Bases for Requesting Exigent Approval

NMC considers that the 1982 failure to incorporate the RVH drop analysis into the 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. As a direct result of this condition, the 10 CFR 50.59 evaluations completed in February 2005 concluded that prior NRC approval was not required prior to implementation.

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, given the results of the RVH drop analysis and the fact that the PBNP containment polar cranes are not single failure-proof. Numerous additional corrective action program documents were initiated as NMC continued to investigate issues associated with this issue. In addition to developing and implementing administrative controls to address the immediate need to lift the existing RVH, NMC initiated corrective actions to resolve each of the identified issues documented in the corrective action program (CAP).

One of the actions needed to resolve the issues surrounding control of heavy loads was revision to the FSAR. Enclosure 2 submits that proposed revision. The 10 CFR 50.59 for this proposed revision for those updates indicated that NRC approval is required prior to implementation.

PBNP Unit 2 is currently in a refueling outage. NMC recognizes that the condition applies to movement of the RVH and replacement RVH during the outage. NMC submitted one-time administrative controls to support lifting of the Unit 2 RVH to continue critical path activities on April 15, 2005 (Reference 2). NMC subsequently provided an update on the status of commitments in Reference (3) and received NRC concurrence of providing a reasonable assurance of protecting the public health and safety during a teleconference held on April 16, 2005. NMC is requesting NRC approval for this proposed exigent amendment to support restart from the outage. The current schedule shows that the RVH may be placed on the reactor vessel as early as 2100 hours on May 9, 2005, with plant restart activities to continue immediately thereafter.

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 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.

The actions taken above have culminated in the development of the proposed revision to the FSAR (Enclosure 2) that incorporates PBNPs commitments and compliance with the guidance contained in NUREG-0612 and in our commitment to implement a new TRM to ensure long-term equipment controls are in place that implement our "defense in depth strategy" for mitigating the consequences of a potential RVH drop event.

All of the above actions were methodically taken to resolve the single error that occurred in 1982. The corrective actions that have been taken to resolve this issue have resulted in organizational learning.

NMC recognized on April 7, 2005, that a reactor vessel head drop analysis was conducted in 1982 but was not inserted into the FSAR. Furthermore, NMC understands the necessity to expeditiously update the FSAR to reflect the strategies necessary to effectively mitigate the consequences of a heavy load drop. The proposed FSAR change will introduce a new accident into the FSAR that requires NRC prior approval. NMC believes this situation places the PBNP Unit 2 refueling outage in the position of requiring exigent approval of the FSAR change in order to:

- Approve the FSAR change, introducing the heavy load handling controls necessary to effectively mitigate the consequences of a load handling event; therefore permitting,
- The ability to successfully complete a 10 CFR 50.59 evaluation of the replacement RVH modification without prior NRC approval; therefore permitting,
- The site to remove the administrative restriction on replacing the Unit 2 reactor vessel head; thereby permitting,
- A safe and timely return to power operation for the Unit 2 reactor.

Therefore, NMC requests NRC approval of this amendment for Unit 2 on an exigent basis in accordance with 10 CFR 50.91(a)(6). While this is a dual-unit submittal, exigent NRC review and approval of the corresponding Unit 1 license amendment is not requested.

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"

ENCLOSURE 1
REQUEST FOR REVIEW OF HEAVY LOAD LIFT ANALYSIS

Table 1
Technical Requirements Manual
Equipment Controls During Reactor Vessel Head Lift Above Fueled Reactor Vessel

<table>
<tr><td>1. All containment penetrations required for containment closure shall be closed, with the following exceptions:

a. One containment personnel airlock may be open provided one door in that airlock shall be capable of being closed and:

b. The airlock door is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/airlock doors or devices to keep the door open/supported do not violate this provision.

c. Personnel are designated each shift with the responsibility for expeditious closure of at least one door on the personnel airlock.

d. Each Containment Purge and Exhaust System penetration shall be either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

e. Other containment penetration flow paths may be open provided they are capable of being closed by an OPERABLE automatic isolation valve.</td></tr>
<tr><td>2. One charging pump with a flow path to the Reactor Coolant System shall be available.</td></tr>
<tr><td>3. One safety injection train shall be available.</td></tr>
<tr><td>4. One containment spray train shall be available.</td></tr>
<tr><td>5. A minimum borated water volume shall be determined and available as a suction source to the required equipment if the lower cavity is filled for refueling activities. If the lower cavity is empty and a flow path to containment sump B from the lower cavity does not exist, then an additional water volume shall be determined and required*.</td></tr>
<tr><td>6. Technical Specification LCO 3.7.9, "CREFS," shall be met.</td></tr>
<tr><td>7. Technical Specification LCO 3.3.5, "CREFS Actuation Instrumentation," shall be met.</td></tr>
<tr><td>8. Containment sump B screen shall be installed.</td></tr>
<tr><td>9. One standby emergency power source capable of supplying each 4.16 kV/480 V, Class 1E safeguards bus on the associated unit shall be OPERABLE.</td></tr>
<tr><td>10. No more than one containment supply and exhaust fan is running.</td></tr>
<tr><td>11. The maximum allowable lift height for the reactor vessel head shall not be exceeded*.</td></tr>
</table>

*Specific values will be incorporated into the Technical Requirements Manual prior to RVH lift.