January 10, 2006

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

SUBJECT: POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING EVENT NOTIFICATION 42129 (TAC NOS. MC9035 AND MC9036)

Dear Mr. Koehl:

On November 8, 2005, Nuclear Management Company, LLC (NMC), notified the U.S. Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors" (Event Notification 42129), that the design basis for long-term cooling at the Point Beach Nuclear Plant, Units 1 and 2, was not correctly modeled. Your notification stated that, "These errors in the modeling fidelity potentially impact the analytical basis for demonstrating compliance with the acceptance criteria of 10 CFR 50.46(b)(5), Long-term cooling." Your notification further stated that operability analyses were performed as immediate actions and that the operability analyses demonstrated that adequate net positive suction head would be available to the emergency core cooling system pumps to ensure long-term cooling.

The NRC staff is reviewing NMC's actions to establish that the requirements of 10 CFR 50.46(b)(5) continue to be met and finds that additional information is needed as shown in the enclosed request. This information is requested for the NRC staff to make a regulatory determination under 10 CFR 50.46(a)(2). The items in the request were discussed with Mr. Jim McCarthy, Site Director of Operations, and other members of your staff, on January 9, 2006, and a target date of February 6, 2006, for your response was established. If you have any questions, please contact me at (301) 415-2296.

Sincerely,

/RA by Harold Chernoff for/

Carl F. Lyon, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure: As stated

cc w/encl: See next page

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POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REQUEST FOR ADDITIONAL INFORMATION

EVENT NOTIFICATION 42129

DOCKET NOS. 50-266 AND 50-301

On November 8, 2005, Nuclear Management Company, LLC (NMC) notified the Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors" (Event Notification 42129), that the design basis for long-term cooling at the Point Beach Nuclear Plant, Units 1 and 2 (PBNP), was not correctly modeled. NMC's notification stated that, "These errors in the modeling fidelity potentially impact the analytical basis for demonstrating compliance with the acceptance criteria of 10 CFR 50.46(b)(5), Long-term cooling."

The NRC staff is reviewing NMC's actions to establish that the requirements of 10 CFR 50.46(b)(5) continue to be met. The NRC staff's review includes the potential blockage of the containment sump and its effect on the ability to sustain long-term cooling, the potential impact of the SI-850 valves to operate so as to sustain long-term cooling, and the potential impact of leakage from the recirculation line, particularly regarding dose to operators. The NRC staff has determined that responses to the following questions are needed to proceed with this review.

For each of the questions below, please ensure that your responses describe your assumptions, methods, and conclusions in sufficient detail to support the NRC staff's independent review. If technical reports are referenced, you should provide a copy of the reports and the technical basis for the applicability of the reports to your facility.

1. General

   A. Provide a discussion of actions taken to demonstrate the ability to establish and maintain long-term cooling in accordance with 10 CFR 50.46(b)(5).

   B. Have you completed a 10 CFR 50.59, "Changes, tests, and experiments," evaluation of compensatory measures (e.g., ECCS flow reduction) taken as part of your Operability Recommendations (OPRs)? If so, provide a copy of those evaluations. If not, please explain why?

   C. Provide a detailed discussion including planned actions and schedule for resolution of any non-conformances with the current licensing basis or degraded conditions.

2. Zone of Influence

   A. What is the zone of influence? How was this determined? What is the basis for this answer?

Enclosure

3. Potential Blockage of the Sump/Long-term Cooling

A. Containment Coatings

1. How much (percentage, area, and volume) of the coatings will fail? Include the location of the failed coatings, the type of coating, and qualification level of the coatings. What is the basis for this answer?

2. What are the physical characteristics of the failed coatings (particle size, thickness, and specific gravity)? What is the basis for this answer?

3. Will the failed coatings be transported, including during the blow down phase of the event, to the sump? What is the basis for this answer?

4. How much (percentage, volume, particle size) of the coatings will be transported, including during the blow down phase of the event? What is the basis for this answer?

5. How much of the degraded qualified and unqualified coatings are on the containment floor (both pre-existing and event generated) in the zone of influence around the sump, and how much of those will be transported to the sump? What is the basis for this answer?

6. What percentage of the sump screen will be blocked by failed coatings or by coatings in combination with other material? What is the basis for this answer?

B. Containment Insulation

1. How much (percentage, volume, type and size) of the insulation will fail, including during the blow down phase of the event? What is the basis for this answer?

2. What are the physical characteristics of the failed insulation (particle size, thickness, and specific gravity)? What is the basis for this answer?

3. Will the failed insulation be transported, including during the blow down phase of the event, to the sump? What is the basis for this answer?

4. How much (percentage, volume, particle size) of the insulation will be transported, including during the blow down phase of the event? What is the basis for this answer?

5. What percentage of the sump screen will be blocked by failed insulation or by insulation in combination with other material? What is the basis for this answer?

C. Containment Debris

1. How much (volume, type and size) containment debris will be transported to the sump? What will happen during the blowdown phase? What is the basis for this answer?

2. What are the physical characteristics of the debris (size, shape, thickness, and specific gravity)? What is the basis for this answer?

3. What percentage of the sump screen will be blocked by debris or by debris in combination with other material? What is the basis for this answer?

D. Sump Blockage

1. What are the safety functions of the emergency core cooling system (ECCS) sump? What is the basis for this answer?

2. What percentage of the sump screen will be blocked by coatings, insulation, and debris? What is the basis for this answer?

3. What percentage of the sump screen is required to be unblocked (or, what head loss can be sustained) to fulfill its safety functions? What is the basis for this answer?

4. Is there a reasonable expectation that the sump will fulfill its safety function? What are the major uncertainties and the sensitivity of the answer to those uncertainties? What is the basis for this answer?

E. Affects on Downstream Components

1. What types, particle sizes and quantity of materials are expected to pass through the sump screens? What is the basis for this answer?

2. What ECCS equipment/components have tight clearances that could potentially be affected by foreign materials that pass through the sump screens (e.g., pump seals, flow orifices, throttle valve trim, etc.)? What is the basis for this answer?

4. <u>SI-850 Valves</u>

A. What are the safety functions of the valves (e.g., to open/stay open, to shut/stay shut, to maintain leak tightness) and what is the basis for this determination? What are the ECCS pump minimum and maximum recirculation flows and net positive suction head (NPSH) requirements? What is the basis for this answer?

B. Have the valves been adequately tested to demonstrate that they will perform each of their safety functions identified above? Explain and identify what testing has been performed? What is the frequency of this testing and how do the test acceptance criteria demonstrate/relate to the valve safety function? What is the

basis for this answer?

C. Is there a reasonable expectation that the valves will perform their safety functions for the duration of the events, as defined in the safety analyses? What is the basis for this answer?

D. What are the consequences if the valves fail closed? If the valves fail closed, can they be re-opened? If these valves can drift shut, what amount of closure will cause the open indication in the control room to be lost and what will be the effect on recirculation flow/NPSH/pump operation with these valves in this partially closed position? Is the equipment that provides control room position indication qualified in accordance with 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"? If they can be re-opened what are the consequences of the time period the valves are not fully open? What is the justification for the time period assumed? Can the valves be opened with pumps in operation? What is the basis for this answer?

E. What is the radiation exposure to the operator if local manual action is necessary? What is the basis for this answer?

F. Will flashing occur in the piping below the valves when they are opened to perform their safety function during an event, including the long term? Consider containment overpressure, ECCS flow, and the number of ECCS trains in operation. If containment overpressure is needed, has it been analytically shown that the minimum overpressure assumed in the analysis will be present for the limiting combination of conditions (e.g., including inadvertent operation of secured equipment that could reduce containment pressure), including the long term? What is the basis for this answer?

G. If flashing occurs, what are the potential consequences? What is the basis for this answer?

5. <u>ECCS Leakage From the Recirculation Line (Flange/Body-Bonnet/Packing/Weld)</u>

A. What is a technically defensible failure (leakage rate) to consider and when and where are these leaks postulated to occur? What is the basis for this answer?

B. What compensatory measures are available to detect and isolate this leakage? If non-safety related equipment is relied on to support detection and isolation, explain why this is appropriate. What is the basis for this answer?

C. How long will detection and isolation of a passive leak take? What is the basis for this answer?

D. What are the consequences of leakage with regard to control room habitability for the limiting passive leak and where and when does this leak occur and what activity level is assumed during this leakage? What is the basis for this answer?

E. What are the consequences of leakage with regard to offsite dose for the limiting passive leak and where and when does this leak occur and what activity level is assumed during this leakage? What is the basis for this answer?

F. Are the consequential radiation exposures within calculated results and regulatory limits? What is the basis for this answer?

G. What are the consequences of passive leakage and isolation capabilities with respect to ECCS functions (e.g., preservation of containment sump inventory to support post-loss-of-coolant accident recirculation)? What is the basis for this answer?

H. Are the SI-850 valves credited with isolating a passive leak? If so, is this a safety-related function? If not, explain. What is the basis for this answer?

I. If the SI-850 valves are credited with isolating a passive leak, explain how much this valve will continue to leak after closure and how this leak rate was determined. If this leak rate has not been measured, explain what a limiting leak rate would be and your basis for this leak rate. What is the basis for this answer?

J. Was the continued leakage past the shut SI-850 valve considered in calculation of control room dose, offsite dose or preservation of containment sump inventory? If not, explain. What is the basis for this answer?

6. <u>At a minimum, provide the following documents with your response</u>

A. Gibbs & Hill, Inc., “Evaluation of Paint and Insulation Debris Effects on Containment Emergency Sump Performance,” for Unit 1, forwarded by letter to WEPC [Wisconsin Electric Power Company], dated May 18, 1989.

B. Gibbs & Hill, Inc., “Evaluation of Paint and Insulation Debris Effects on Containment Emergency Sump Performance,” for Unit 2, forwarded by letter to WEPC, dated August 1, 1990.

C. Sargent and Lundy Calculation M-09334-345-RH-1, “Containment Sump Blockage Due to Failure of Unqualified/Undocumented Coatings (Unit 1), Rev. 0,” approved June 4, 1998, and Rev. 1, issued January 21, 1999.

D. Sargent and Lundy Calculation M-09334-431-RH-1, “Containment Sump Blockage Due to Failure of Unqualified/Undocumented Coatings (Unit 2), Rev. 0,” issued January 1, 1999.

E. Operability Recommendation OPR-161, current revision.

F. Operability Recommendation OPR-162, current revision.

G. Point Beach Nuclear Plant Engineering Evaluation 2005-0024, “Evaluation of Containment Sump Screen Debris Buildup Based on EPRI [Electric Power Research Institute] Technical Report and Current Degraded Epoxy Inventories, Rev. 1,” dated November 9, 2005.

H. Point Beach Nuclear Plant CAP 068442, Generic Letter 1998-04 Commitments.

I. EPRI Technical Report 1011753, “Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings.”

J. Operability Recommendation OPR-164, current revision.

K. Operability Recommendation OPR-165, current revision.

Point Beach Nuclear Plant, Units 1 and 2

cc:

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