NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0090

September 18, 2003

Mr. J. L. Caldwell, Regional Administrator
U. S. Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

DOCKETS 50-266 AND 50-301
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
SUBMITTAL OF ADDITIONAL INFORMATION CONCERNING
AUXILIARY FEEDWATER ORIFICE REGULATORY CONFERENCE

On June 6, 2003, a regulatory conference was conducted between representatives of the Nuclear Management Company, LLC (NMC) and members of your Staff to discuss the Auxiliary Feedwater Orifice Issue at Point Beach Nuclear Plant (PBNP). During the presentation, several questions were raised regarding information presented or discussed during the conference. In a letter to the NRC dated June 27, 2003, NMC provided information to address the questions posed by the NRC during the presentation. Most of these questions were related to the fire probabilistic risk assessment (PRA) NMC was scheduled to complete by the end of August 2003.

Attached and enclosed with this letter is information regarding the current Unit 2 PRA results, including fire events. The Unit 2 results bound the Unit 1 results, because the Unit 2 turbine-driven auxiliary feedwater pump (TDAFWP) recirculation line orifice was installed much longer than the Unit 2 TDAFWP recirculation line orifice. An independent review of these results is scheduled for October 2003.

Internal Events and Seismic

In June, we provided the NRC with a summary of our preliminary determination of the increase in core damage probability due to internal and seismic events. Since then, corrections were made to a few failure probabilities and to the system success criteria for one initiator. With work completed to-date, the change in core damage probability (CDP) for Unit 2 is 7.7E-05 for internal events and 9E-06 for seismic events. Change in core damage probabilities by initiator are provided in Attachment A.

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Fire Events

A great deal of time was spent evaluating the impact that this issue had on fire events. Fire compartments were first ranked for their potential contribution to risk given the potential for failure of Auxiliary Feedwater. A detailed analysis of the most risk significant contributing compartments has been prepared. This analysis included scenario development and detailed fire models using state-of-the-art software and techniques.

The current core damage probabilities from the detailed fire models for the most risk significant compartments were summed with the estimated values determined for the remaining compartments. The current total change in core damage probability results from fire events for Unit 2 due to this issue is between 1.1E-04 and 2.2E-04. A summary of current change in core damage probabilities by compartment and a more detailed description of the fire analysis is contained in Attachment B.

If you have any questions, please contact Gordon P. Arent at 920/755-6518.

A. J. Cayia
Site Vice President

RDS/kmd
Enclosure

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Attachment: A. Change in Core Damage Probabilities By Initiator
B. Change in Core Damage Probabilities By Compartment

cc: (with enclosure)
S. Burgess, Senior Reactor Analyst, NRC Region III

cc: (w/o enclosure)
T. Vegel, PBNP Branch Chief, NRC Region III
Mr. M. Kunowski, Project Engineer, NRC Region III
NRC Resident Inspector - Point Beach Nuclear Plant
PSCW

ATTACHMENT A

<u>Change in Core Damage Probabilities By Initiator</u>

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Summary of Current Delta Core Damage Probabilities for Internal Events and Seismic for Unit 2

Loss of Service Water	2.5E-05
Dual Unit Loss of Off-site Power	1.6E-05
Loss of Instrument Air	4.9E-07
Loss of DC Bus D02	1.5E-05
Transient with Loss of Heat Sink	6.3E-06
Transient with Heat Sink	2.7E-07
Single Unit LOOP	1.3E-05
Total for Internal Events	**7.7E-05**
Seismic Event	9.1E-06

Significant Changes Since June

1. The contribution of Loss of Service Water events increased significantly due to eliminating the option to use low pressure injection to the steam generators using fire water. Cooling provided by this method is insufficient to prevent the reactor coolant system (RCS) from going solid, forcing open a Pressurizer safety valve and passing water. This has a high likelihood of failing the valve in the open position, resulting in an opening from the RCS to containment. Even though RCS makeup from Safety Injection is available, containment cooling and residual heat removal (RHR) cooling are not available due to a lack of service water. This eliminates the ability to use containment sump recirculation, which is required once the refueling water storage tank (RWST) inventory is depleted.

2. An adjustment was made to the human error probability (HEP) for placing shutdown cooling in service during a loss of DC Bus D02 event, because the value used originally was for cases with two trains available. In the loss of D02 event, only one train is available, and recovery of a human error is less likely because an opportunity to correct the error by placing the second train in service is eliminated.

3. A reduction in the failure probability for shutdown cooling and containment sump recirculation was made possible when one of the largest contributing cutsets was eliminated. The cutset, a flow diversion through a test valve that was inadvertently left in the open position was found to not be credible because a second locked-closed valve in the same test return line would prevent diversion. Correcting this error resulted in a small reduction of the change in core damage probability values for all initiating events except loss of service water where shutdown cooling or recirculation are not possible.

4. The core damage probability contribution due to seismic events increased because the impact of additional stress from dealing with the aftermath of an earthquake were factored into the human error probabilities. The method of including this effect was the same as was used in the IPEEE for seismic events and resulted in the HEPs being increased by a little less than a factor of two.

ATTACHMENT B

Change in Core Damage Probabilities By Compartment

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Description of AFW Recirculation Orifice Fire Risk Analyses

A screening process was performed that identified potential significant core damage probability increases considering

- Fire Frequency
- Fire Severity Factors
- Automatic Suppression Capability
- Manual Suppression Capability
- Extent of Equipment Damage (including loss of AFW)

Fire frequencies and severity factors were developed using latest EPRI information. Automatic suppression capability was based upon plant specific designs considering generic industry failure probabilities. For the initial screening, manual fire suppression probabilities and core damage probabilities were estimated.

The screening resulted in 9 fire compartments that had potential to significantly increase the core damage probability over the 1 year period being considered. The remaining compartments were determined to have a negligible impact on CDP. Following is a prioritized list of fire compartments from most significant to least significant.

- 26' Elevation Central Primary Auxiliary Building (PAB) (Fire Zone 187)
- Vital Switchgear Room (Fire Zone 305)
- 8' Elevation PAB MCC Room 2B32 (Fire Zone 166)
- Cable Spreading Room (Fire Zone 318)
- 8' Elevation PAB Component Cooling Water Pump Room (Fire Zone 142)
- Aux Feed Water Pump Room (Fire Zone 304)
- 46' Elevation PAB CCW Heat Exchanger Room (Fire Zone 237)
- Instrument Air Compressor Room (Fire Zone 310)
- 13KV Switchgear Rooms (Fire Zones 675, 676, 677)

Detailed evaluations, which include state of the art fire modeling techniques (Fire Dynamic Simulator), have been prepared for 4 of the nine compartments.

- 26' Elevation Central PAB (Fire Zone 187)
- Vital Switchgear Room (Fire Zone 305)
- Cable Spreading Room (Fire Zone 318)
- Instrument Air Compressor Room (Fire Zone 310)

The increase in CDP for the remaining 5 compartments was determined by using a detailed fire frequency analysis combined with results from the internal events analysis.

Fire Dynamics Simulator (FDS) is a computational fluid dynamics model of fire-driven flow developed by the National Institute of Standards and Technology (NIST). In FDS, each room or building of interest is divided into small rectangular control volumes or computational cells. The model then computes the density, velocity, temperature, pressure and species concentration of the gas in each cell, based upon the conservation laws of mass, momentum, and energy to model the movement of the fire gases. FDS is designed as a best estimate tool that is not intended to include excessive conservatism.

Attachment B
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Following is an estimate of the increase in Unit 2 core damage probability over the 1 year period being considered with work completed to date:

Fire Compartment	Increase in CDP
26' Elevation Central PAB (Fire Zone 187)	5E-5 to 9E-5
Vital Switchgear Room (Fire Zone 305)	2E-5
8' Elevation PAB MCC Room 2B32 (Fire Zone 166)	4E-6 to 2E-5
Cable Spreading Room (Fire Zone 318)	1E-5 to 4E-5
8' Elevation PAB Component Cooling Water Pump Room (Fire Zone 142)	3E-6 to 2E-5
Aux Feed Water Pump Room (Fire Zone 304)	5E-6 to 1E-5
46' Elevation PAB CCW Heat Exchanger Room (Fire Zone 237)	2E-6 to 8E-6
Instrument Air Compressor Room (Fire Zone 310)	7E-6
13 KV Switchgear Rooms (Fire Zones 675, 676, 677)	5E-6
Total	1.1E-4 to 2.2E-4