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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

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

POINT BEACH NUCLEAR PLANT
DOCKETS 50-266 AND 50-301
RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION REGARDING
RELIEF REQUEST 3, RISK-INFORMED INSERVICE INSPECTION PROGRAM

On July 1, 2002, Point Beach Nuclear Plant (PBNP) updated the Inservice Inspection (ISI) Program to the 1998 Edition of ASME Section XI with all addenda through 2000. This edition and addenda were approved for use via a safety evaluation report (SER) issued by the Nuclear Regulatory Commission (NRC) dated November 6, 2001.

On July 3, 2002, PBNP submitted Relief Request 3, a request for relief to allow a Risk-Informed Inservice Inspection Program as an alternative to the requirements of ASME Section XI categories B-F, B-J, C-F-1, and C-F-2 examination methods and selection criterion.

During a conference call held with PBNP staff on January 16, 2002, NRC staff discussed additional information needed to support Relief Request 3.

Attachment 1 of this letter provides the NMC's response to the staff's questions.

This letter contains no new commitments and no revision to existing commitments.

A. J. Cayia
Site Vice President

LAS/kmd

Attachment

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

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RELIEF REQUEST 3

RISK-INFORMED INSERVICE INSPECTION PROGRAM

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

The following information is provided in response to the Nuclear Regulatory Commission staff's request for additional information, regarding Point Beach Nuclear Plant (PBNP) Relief Request 3, as discussed during a telephone conference on January 16, 2003.

The NRC staff's questions are restated below, with the NMC response following.

NRC Question 1:

The submittal states, "The original results provided for the RI-ISI analysis are based on the 1996 PRA Update. The last PRA update was done in the summer of 2001, to prepare for the PRA Certification effort. To the degree available, those results are incorporated into this study."

Please elaborate on the last sentence quoted above.

Response:

Only results from the updated 2001 version of the PBNP PRA model were used for the Risk-Informed Inservice Inspection Program (RI-ISI) analysis. This model is Revision 3.00, dated 10/12/2001. Earlier versions of the PRA model were provided to our contractors for their use in preliminary studies, but all final conclusions were based on the October 2001 model. This is the same model that was reviewed in draft form by the Westinghouse Owner's Group (WOG) PRA Peer Review Team in June 2001.

NRC Question 2:

The staff review of the original IPE and the WOG review of the latest version of the PRA have identified shortcomings in the treatment of pre-accident human errors. Furthermore, according to your submittal, the WOG reviewers also critiqued the common cause failure analysis performed in the latest version of the PRA. Please elaborate on your claim that the RI-ISI consequence evaluation results would not be impacted once these modeling deficiencies are eliminated.

Response:

The Staff Evaluation Report of the Individual Plant Examination (IPE) submittal noted that pre-initiator instrument miscalibration events were not included in the PRA model. The report also pointed out that valves in support systems were not systematically evaluated for restoration errors as was done for front-line systems. The WOG PRA Peer Review Team also identified the lack of instrument miscalibration events in the model and had a concern that the restoration error probabilities in the model were based on screening values rather than a detailed evaluation of each specific restoration error opportunity.

PBNP agrees that miscalibration events have the potential to be important to the reliability of some systems used for accident response in general. PNBP will be systematically reviewing systems for these potential pre-initiator errors when this Peer Review item is addressed. The Peer Review Team stated in their report that instrument calibration errors that could be important relate to Auxiliary Feedwater (AFW) flow, Condensate Storage Tank (CST) level, containment sump level, and Refueling Water Storage Tank (RWST) level. For LOCAs, AFW is only used over the long-term for small breaks. Redundant and diverse indications exist for the operators for AFW flow (steam generator level) and for CST level (multiple level instruments and a low suction pressure trip for the AFW pumps).

Containment sump level and RWST level are important for all LOCA events, but these two parameters also have redundant instrumentation and serve as backup indicators for each other (as RWST level lowers, containment sump level rises). Decision criteria in the Emergency Operations Procedures (EOP) to switch to containment sump recirculation is based on either parameter meeting set criteria. Other instrumentation calibration errors, such as temperature, level, and pressure of the reactor coolant system and containment could be present. However, because redundant and diverse indications are available to the operators and because the emergency procedures use multiple parameters for major decision points and also provide recovery paths, the lack of instrument miscalibration errors in the PRA model used for the Risk-Informed ISI Analysis will not affect the consequence evaluation results to any significant degree.

Valve restoration errors are now included in the Point Beach PRA model for both standby front-line systems and for important support systems, such as Main Steam supply to the turbine driven AFW pumps, valves for the standby Component Cooling Water heat exchangers, Instrument Air valves, Fire Protection valves, and numerous Service Water valves. The WOG PRA Peer Review Team issue was primarily a concern that the values used for the restoration errors were too high such that they may be appearing in some dominant cutsets when they in fact should not. Since the review, PBNP has examined a few of the restoration error probabilities using a standardized process. The results of this preliminary look indicate that it may be possible to lower the restoration error probabilities by 20% to 60%. PBNP also looked at the importance of restoration errors to the LOCA core damage results. Restoration errors account for approximately 10% of the Large LOCA core damage frequency results, approximately 15% of the Medium LOCA results, and less than 10% of the Small LOCA results. Replacing the current bounding valve restoration error probabilities with best estimate values will result in less than a 10% reduction in the LOCA contribution to total core damage frequency and will not significantly affect the consequence evaluation results.

Regarding the common cause failure analysis, the primary global issue raised by the Peer Review Team was that the method in which choosing of common cause groups and Multiple Greek Letter (MGL) factors were chosen was not well documented. The Peer Review report summary for the Data Analysis element concludes that the MGL common cause methodology used by PBNP is "generally consistent with that documented in NUREG/CR-4780."

The Peer Review Team did have specific issues with the common cause failure analysis for Component Cooling Water pumps, Service Water pumps, Instrument Air and Service Air compressors, and the two generations of Emergency Diesel Generators. The Component Cooling Water and Service Water systems play a major role in all sizes of LOCA by providing cooling to the Residual Heat Removal heat exchangers for containment sump recirculation cooling. However, component failures in these systems are dominated by the human error probabilities (HEPs) for switching from injection mode to high or low head recirculation. These HEPs have values of 1.25E-02 and 2.45E-02, respectively. Common cause failures in the Service Water and Component Cooling Water systems have probabilities at least two orders of magnitude lower than these dominating HEPs and will have very little impact on the LOCA core damage results even if they are increased significantly after further analysis.

The loss of instrument air due to a common cause failure of the compressors will impact some of the mitigating systems for LOCA events. Auxiliary Feedwater, Steam Generator Relief Valves, and Pressurizer Power Operated Relief Valves all rely on instrument air and are used in the response to Small LOCAs and, to a lesser degree, Medium LOCAs. However, as discussed above, the low common mode failure probabilities are dominated by at least two orders of magnitude by the HEPs in these core damage sequences. Even a significant increase in the compressor common cause failure probability will not appreciably affect the LOCA results.

Finally, with regard to the Emergency Diesel Generators (EDGs), the WOG PRA Peer Review Team primarily questioned the lack of a detailed justification for not including a common cause factor for all four of our EDGs. The diesel units are a generation apart. Two EDGs were installed when the plant was constructed, and two were added 25 years after plant start-up. The WOG PRA Review Team agreed that there was a reasonable case for not having a common cause failure for all four diesels. For LOCA events, the EDGs are not used unless a concurrent loss of offsite power takes place. Since it is extremely unlikely that a loss of offsite power would occur as a consequence of a LOCA and the probability of a concurrent (within 24 hours) random loss of offsite power is also very low (1.4E-03), EDG failures have a very small impact on LOCA results.

In conclusion, the pre-initiator event and common cause deficiencies identified by the Staff Review Report and by the WOG PRA Peer Review will have no significant impact on the consequence evaluation results presented in the Risk-Informed ISI submittal.

NRC Question 3:

The risk profile of the PBNP has changed significantly relative to the IPE. In the IPE model, the LOCA events were the dominant contributors to CDF. This is not the case for the latest PRA model, which identified SGTR events as dominant contributors. Please identify design/operation modifications that you credited in the PRA model that are responsible for reducing the importance of LOCA sequences.

Response:

The difference in Medium and Large LOCA significance between the Revision 0 PRA model developed for the IPE and the Revision 3.00 model is primarily due to a significant reduction in the initiator frequencies. The reduction in the importance of the Small LOCA is due to PRA model improvements as described below.

Table 3.1 below shows the LOCA initiator frequencies for the three break sizes from the current PRA model and the IPE PRA model. The Medium LOCA frequency dropped by an order of magnitude, and the Large LOCA frequency dropped by two orders of magnitude in the Revision 3.00 PRA model versus the IPE model. The Small LOCA frequency actually went up slightly from the IPE value. This is because more events, such as a stuck open relief or safety valve and Reactor Coolant Pump seal LOCA events, were lumped with the pipe failures into the most recent Small LOCA frequency. The source for all of the updated LOCA frequencies used in the Revision 3.00 PRA model is NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995.

Table 3.1
Comparison of LOCA Initiating Event Frequencies

LOCA Size	Initiator Frequency Rev 3.00	Initiator Frequency IPE
Small	3.2E-03/yr	3.0E-03/yr
Medium	1.1E-04/yr	1.0E-03/yr
Large	5.0E-06/yr	5.0E-04/yr

In Table 3.2 three sets of LOCA core damage frequency (CDF) results are shown. The purpose of this table is to compare the CDF results from the different versions of the PRA model to determine if the reduction in importance of LOCAs in the Revision 3.00 model is due primarily to use of lower initiating event frequencies. The first column has CDF results for the three LOCA sizes from the Rev 3.00 PRA model used for the Risk-Informed ISI analysis. The right-hand column has the LOCA results from the IPE version of the PRA model. The middle column has CDF values for each of the three LOCA sizes that result from using the IPE LOCA initiator frequencies in the Revision 3.00 fault trees. If the results in this middle column are close to those in the right column, then the change is primarily due to the new initiating event frequencies. If the results are still substantially different, then some other model changes are responsible for the reduction in LOCA significance.

Table 3.2
Comparison of LOCA Core Damage Frequencies

LOCA Size	CDF Rev 3.00	CDF Rev 3.00/IPE*	CDF IPE
Small	3.2E-07/yr	3.0E-07/yr	2.0E-06/yr
Medium	1.8E-06/yr	1.6E-05/yr	1.1E-05/yr
Large	1.4E-07/yr	1.4E-05/yr	2.6E-05/yr

* Used Rev 3.00 fault trees and data with IPE LOCA initiating event frequencies

Comparing the middle column CDF results to the IPE LOCA CDF results leads to the conclusion that most of the difference between the Revision 3.00 and IPE results for Medium and Large LOCA events is not due to system changes, but is due to the reduction in LOCA initiator frequencies. The remainder of the difference is most likely due to availability and reliability data changes.

For the Small LOCA, using the IPE initiator frequency in the Revision 3.00 model still shows an order of magnitude reduction in CDF as compared to the IPE result. That difference has been traced to an improvement in the PRA fault tree model for High Pressure Safety Injection (SI) that was implemented in Revision 3.00. In the IPE PRA model for SI, a simplifying assumption was made that the SI train associated with the Reactor Coolant System (RCS) loop containing the break was failed. In the plant, the SI system is constructed with a restriction in the injection line from SI to each RCS loop that will limit the diversion of injection flow out of a break in the RCS loop. Upstream of the flow restriction is a crosstie between the two loops of SI that will allow either train to inject to either RCS loop. This crosstie along with the downstream flow restriction allows successful injection from either SI train regardless of which RCS loop contains the break. Including this feature in the PRA model resulted in a significant reduction of the failure probability of the SI function for a Small LOCA because redundant trains are now properly credited. This accounts for the reduction in significance of the Small LOCA in Revision 3.00 as compared to the IPE model.