NMC
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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

February 27, 2004

NRC 2004-0022
GL 96-06

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
Response to Request for Additional Information Regarding Generic Letter 96-06

Reference: 1) Letter from Nuclear Management Company, LLC, to NRC, "Supplement to Generic Letter 96-06 Response," dated March 27, 2003 (NRC 2003-0025).

The NRC staff issued Generic Letter (GL) 96-06 on September 30, 1996. Wisconsin Electric Power Company (WEPCO), then licensee for the Point Beach Nuclear Plant (PBNP), provided its assessment of the waterhammer and two-phase flow issues for PBNP in letters dated January 28, June 25, and December 18, 1997, and related submittals dated September 9, September 30, and October 30, 1996. Responses to NRC requests for additional information were provided on September 4, 1998, and October 12, 2000. With these submittals, the GL 96-06 two-phase flow issues were fully addressed.

Actions to fully address the waterhammer issues were deferred pending completion of the Electric Power Research Institute (EPRI) project and its review and approval by the NRC. EPRI Report TR-113594, Resolution of Generic Letter 96-06 Waterhammer Issues, was issued in December 2000, and NRC accepted the report on April 3, 2002. The waterhammer issues were initially evaluated for PBNP using Fauske & Associates (FAI) and their in-house thermal-hydraulic transient modeling tool, TREMOLO. On July 30, 2002, Nuclear Management Company, LLC, (NMC) submitted updated information regarding actions to address the resolution of GL 96-06 waterhammer issues at PBNP.

On August 14, 2002, the NRC requested additional information regarding the July 30, 2002, submittal and the use of the TREMOLO methodology. During a conference call held August 20, 2002, the NRC staff, NMC personnel, and FAI discussed the additional information requested by the NRC to support their review. During the conference call, NMC proposed to provide sample cases and additional basis for the rationale that the FAI analyses for PBNP bound the EPRI methodology.

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On September 10, 2002, NRC staff agreed to review the comparison between the TREMELO methodology and the EPRI methodology as proposed by NMC.

On March 27, 2003, NMC provided an FAI Calculation Note generated to calculate the waterhammer loads for the PBNP containment fan coolers using the EPRI methodology and compared those results against the results previously generated using TREMOLO. The FAI Calculation Note demonstrated that the forcing functions achieved by the EPRI approach were bounded by the TREMOLO-produced forcing functions. By letter dated November 3, 2003, NMC provided additional information in response to questions from NRC staff.

As discussed during a conference call between NRC staff and NMC personnel on January 30, 2004, the NRC has requested additional information to support their review of the GL 96-06 issue at PBNP. Enclosure 1 of this letter contains the NMC's response to the staff's questions.

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

Gary D. Van Middlesworth

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Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
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PSCW

ENCLOSURE 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 96-06

NRC Question 1:

The October 12, 2000, response to RAI Item 6, indicates that: "Several plant modifications have been implemented (and more are in progress) to enhance the reliability of automatic isolation of non-essential SW loads during limiting accidents..." However, the July 30, 2002, response to the same item does not include any discussion of these modifications. Please provide a complete response to RAI Item 6 for both waterhammer and for two-phase flow, and indicate to what extent all planned modifications have been completed along with schedules for completing any remaining modifications (as applicable).

> **Original NRC Question – RAI Item 6:**
>
> Describe in detail any plant modification or procedure changes that have been made or are planned to be made to resolve the waterhammer and two-phase flow issues.

NMC Response:

All modifications and procedure changes necessary to address the waterhammer and two-phase flow concerns of GL 96-06 have been completed.

Waterhammer

The dynamic loads resulting from water column closure and steam void collapse was initially evaluated for PBNP using Fauske & Associates (FAI) and their in-house thermal-hydraulic transient modeling tool, TREMELO. The details of those analyses are presented in our previous responses to GL 96-06 and subsequent responses to the staff's questions.

Using the results of those early analyses, several piping support modifications were implemented on Unit 1.

The analyses, their resultant piping support modifications, and the installation status of those modifications is tabulated below.

Unit 1 Inside Containment			Unit 2 Inside Containment		
Analysis	**Mod#**	**Installed**	**Analysis**	**Mod#**	**Installed**
WE-100121 Rev 1	96-064A	Yes	WE-200090 Rev 1	96-064C	Yes
WE-100123 Rev 1	96-064A	Yes	WE-200091 Rev 1	96-064C	Yes
WE-100125 Rev 1	96-064A	Yes	WE-200092 Rev 1	96-064C	Yes
WE-100126 Rev 1	96-064A	Yes	WE-200093 Rev 1	96-064C	Yes
WE-100129 Rev 1	96-064A	Yes	WE-200094 Rev 1	96-064C	Yes
WE-100131 Rev 1	96-064A	Yes	WE-200095 Rev 1	96-064C	Yes
WE-100132 Rev 1	96-064A	Yes	WE-200096 Rev 1	96-064C	Yes
			WE-200097 Rev 1	96-064C	Yes

Unit 1 Outside Containment			Unit 2 Outside Containment		
Analysis	**Mod#**	**Installed**	**Analysis**	**Mod#**	**Installed**
WE-300035 Rev 1	96-064B	Yes	WE-300044 Rev 1	96-064D	Yes

In aggregate, the above modifications installed 5 new supports, modified 67 existing supports, and removed 11 existing supports. The majority of these changes were for SW return piping immediately adjacent to the Containment Fan Coil units (CFCs). Changes more distant from the coolers were confined to the return piping, and were for purposes of accommodating thermal growth due to higher return temperatures than those used in the original system design.

Subsequent to the above modifications being installed, each of the four CFCs in each containment was replaced due to aging degradation. The replacements were performed under modification packages MR-98-024*J, K, X, & Y. In conjunction with the replacement, the service water (SW) supply and return piping to each CFC was substantially modified.

The original piping consisted of relatively long runs of small diameter (2") carbon steel piping from a valve manifold located some distance from the CFCs. The replacement design ran the larger diameter supply piping as close to the CFC units as possible before short branch lines distributed the flow to the individual coils. This change in configuration eliminated many of the supports modified or installed by the previous efforts. The same methodology (i.e., generation of design loads by use of TREMOLO and subsequent design to meet Code allowable stresses) was used in designing the replacement piping and supports.

Two-Phase Flow:

The PBNP SW system supports several non-essential loads that were designed to automatically isolate upon receipt of an accident signal to ensure that sufficient SW is available to all safety related accident loads. As originally designed, these isolations occurred only if the actuating logic detected less than four SW pumps operating after the automatic start signal to all the pumps had been processed (i.e., logic, power failure, or pump unavailability had rendered at least three of the six pumps non-functional). Also, each non-essential load was isolated by only a single valve actuated from either the "A" or "B" safeguards actuation train.

The combination of a "4 out of 6" logic and the single failure vulnerabilities introduced by having only a single automatic isolation valve on each non-essential load created a very complex set of failure modes and effects, particularly when the two-phase flow concerns of GL 96-06 were being considered.

To increase the SW pressure at the higher elevation CFCs and alleviate two-phase flow concerns, modifications were completed to provide redundant automatic isolation of non-essential SW loads that must isolate to ensure adequate performance of the CFCs (i.e. outlet enthalpy of each of the cooling coils remains less than saturation). The loads now automatically isolated with redundant valves (one each actuated from Train A and Train B) are the following:

- Spent Fuel Pool cooling heat exchangers,
- Radioactive Waste handling system,
- Water Treatment system,
- Primary Auxiliary Building non-safety auxiliary loads (local area coolers, etc.)

One set of non-essential (but important) loads was intentionally omitted from this scheme: the turbine hall on both units. The cooling provided by these loads is essential for personnel safety and commercial investment protection as it provides cooling to the turbine lubricating oil, hydrogen coolers, and hydrogen seal oil system on the main generator. Loss of cooling to these components during turbine coast down could pose a significant fire hazard. Hydraulic evaluations of the SW system have demonstrated that these non-essential loads can be supported without compromising nuclear safety related functions, even under the most limiting conditions postulated.

As a result of the completed modifications, receipt of any Safety Injection signal (i.e., Train A or Train B signal on either unit) will cause automatic isolation of non-essential SW loads using redundant and independent isolation valves independent of the number of operating pumps. This simplifies the system design, provides consistency in plant accident response, and eliminates the potential to divert SW due to the limiting failure of a single train of Safety Injection to actuate. By ensuring SW is not diverted, the two-phase flow concern of GL 96-06 was also addressed.

Finally, the replacement of the CFCs (as discussed under Water Hammer above) also minimized the frictional head losses of SW supplied to the CFCs by modifying the supply piping. The recovered pressure significantly increased the margin to two-phase conditions at the outlet of the CFCs.

Procedure Changes:

The water hammer phenomena discussed in GL 96-06 are of very short duration, and occur at the beginning of a transient when Operators cannot reasonably be expected to provide mitigating intervention. Therefore, the approach taken by Point Beach was to modify the system and components as necessary to accommodate the transients without losing functionality. There was no attempt to credit operator action for mitigation of water hammers. As a result, emergency operating procedure changes were limited to verifying that the automatic actions performed by the system had been completed, and ensuring that the system configuration is maintained per the design.

Prevention of two-phase flow is dependent upon maintaining the SW system within the analyzed operating envelope. This is a concern both during the injection phase of an accident, and again later when establishing containment sump recirculation. During recirculation, some of the available SW is diverted to the Component Cooling (CC) heat exchangers in support of Residual Heat Removal (RHR) operation. This reduces the pressure in the SW headers, and consequently reduces the margin to boiling in the CFCs. Accordingly, a step was added to the appropriate procedures (EOP-1.3 and 1.4) to isolate additional SW demands (turbine hall loads) if necessary prior to establishing containment sump recirculation.

The following procedures have been modified throughout the course of the modification implementation to ensure that they reflect the SW hydraulic analyses of record at any given time:

- OI-70, Operating Instructions – Service Water System Operation
- OI-71, Operating Instructions – Component Cooling Water System, later superceded by 1/2-SOP-CC-001, Component Cooling System
- PBF-2033, Unit 2 Turbine Building Hall Shift Log
- PC-43 Part 3, Service Water System Strainers and Flushing
- IT-72, Service Water Valves
- IT-7C, P-32C Service Water Pump Inservice Test

Various lesser maintenance and testing procedures were also revised to ensure that the automatic design functions are maintained and tested to ensure continued operability, and that these supporting procedures reflect the new plant configuration (e.g., the new automatic isolation valves are tested as part of the station IST program, and testing of the previous "4 out of 6" pump logic was eliminated, etc.).

NRC Question 2

Because the NRC staff is relying on the EPRI methodology as a means to judge the acceptability of the waterhammer analysis that was performed for the Point Beach units, a risk assessment similar to the one that was performed by EPRI (EPRI letter dated February 1, 2002) is required. The EPRI letter is included in the EPRI reports, in an Appendix that contains EPRI correspondence.

NMC Response:

The NRC accepted the EPRI Report in a safety evaluation dated April 3, 2002. This safety evaluation stated that the methods proposed in the EPRI Reports may be considered to not significantly increase the risk of unacceptable plant performance nor lead to an unacceptable risk to the plant. Therefore, the methods of the EPRI report may be safely implemented without compromising the integrity or safety of the piping for plant application.

NMC provided sample cases and additional basis to the NRC, for the rationale that the FAI analyses (TREMOLO methodology) for PBNP bounded the EPRI methodology. On September 10, 2002, NRC staff agreed to review the comparison between the TREMOLO methodology and the EPRI methodology as proposed by NMC. On March 27, 2003, NMC provided an FAI Calculation Note generated to calculate the waterhammer loads for the PBNP containment fan coolers using the EPRI methodology and compared those results against the results previously generated using TREMOLO. The FAI Calculation Note demonstrated that the forcing functions achieved by the EPRI approach were bounded by the TREMOLO-produced forcing functions.

Therefore, the FAI analyses for PBNP produces more conservative results than the EPRI methodology. Consequently, the risk assessment performed by EPRI in their February 1, 2002 submittal applies to PBNP as well.

The relevant portions of that risk assessment are reproduced below.

Risk Consideration

In order to assess the risk to Point Beach, a review of the "progression" of events that could lead to an unacceptable condition was performed similar to that discussed in the NRC safety evaluation dated April 3, 2002. For the purposes of this evaluation, the "unacceptable condition" following a LOOP/LOCA event will be defined as a breach of the service water system pressure boundary. The events are as follows:

1. **Occurrence of a LOCA or MSLB**
   The probabilities of Occurrence of LOCA and MSLB events are provided in NUREG/CR-5750. From that document, the mean frequency of occurrence of a large LOCA is 5E-06/year, a medium LOCA is 4E-05/year, and a MSLB is

1E-03/year. The LOCA probabilities are represented in NUREG/CR-5750 as "reasonable but conservative" estimates of the frequency of occurrence.

2. **Occurrence of a LOOP following a LOCA or MSLB**
   Studies provided in NUREG/CR-6538 and subsequent NRC work indicate that the dependent probability of a LOOP event following a LOCA event is approximately 1.4E-02/demand.

3. **Occurrence of a Simultaneous LOCA/LOOP Event**
   The required design basis consideration is for the simultaneous occurrence of a LOCA or MSLB and a LOOP. The frequency of the combined event depends upon the probability of the LOCA and the MSLB and the dependent probability of the LOOP given that the LOCA has occurred. Using the values defined in each of the NUREGs referenced above gives a probability of the combined event on the order of 1.5E-05/year. For our purposes here, the value of probability of the design basis event (LOCA or MSLB occurring simultaneously with a LOOP) will be taken as 1E-05/year. With best estimate probabilities, this event likelihood of occurrence could be expected to be even lower.

4. **Void Formation**
   If PBNP has a LOCA/LOOP event, a void will form with certainty.

5. **Pump Restart**
   The pumps will restart with certainty and the velocity of the fluid in the pipe, immediately prior to closing the void, will be defined by the pressure in the void, the piping geometry, and the pump characteristics. This uncushioned closure velocity can be reliably calculated. This velocity will not be higher than the rate at which the pumps, once restarted, can pump water. The calculation of the water velocity prior to closure is a plant specific analysis that can be conservatively performed. In the case of Point Beach, these calculations have been performed and the results confirm this assertion by EPRI.

6. **Column Closure**
   The water columns will refill the void and the velocity at closure cannot be larger than the largest calculated differential velocity for the upstream and downstream water columns.

7. **Maximum Waterhammer Pressure**
   An upper bound on the water hammer pressure can be calculated by the Joukowski relationship with the uncushioned closure velocity that corresponds to the pipe in which the closure will occur. The waterhammer pressure cannot be larger. With a probability of one, the waterhammer pressure will be equal to or less than the Joukowski pressure. The actual waterhammer pressure that will occur is stochastic and will have a wide variation. This variation is due to variations in the void distribution in the system immediately prior to final closure. This variation appears in

all the integral system level experiments. The variation in the test data has been reviewed and, in the velocity range of interest, it varies from 50% to 100% of the maximum (for example, in the Configuration 2a tests, at a velocity of approximately 25 feet per second, the maximum pressure measured from the test was approximately 400 psig, the minimum pressure was approximately 200 psig, the Joukowski pressure for this velocity of closure is 775 psig -- see Figure 10-9 in the Technical Basis Report). The variation in the test data that has been seen as part of the EPRI project is typical of many other waterhammer tests that have been previously performed and it indicates that it is unlikely that the Joukowski pressure will be attained, given the scatter in the results of measured waterhammers compared to those predicted. It is assumed in the EPRI reports that the largest (Joukowski) pressure is attained for the calculated cushioned velocity, although it is very likely that a pressure less than the maximum seen in a test, will be experienced.

8. **Cushioned Waterhammer**
   With the cushioning that is predicted to occur due to gas and steam, the cushioned velocity will be on the order of approximately 30% to 40% lower than the maximum velocity. (According to the User's Manual Appendix, this depends on many parameters, including the amount of gas and steam.) The waterhammer that is predicted, then, will be on the order of 30% to 40% less than the pressure calculated by Joukowski, as the relationship between pressure and velocity is linear. If the cushioning did not occur, the waterhammer pressure and the stresses in the piping would be equivalent to the uncushioned waterhammer that would not have the 30% to 40% adjustment. There are two ways to consider the impact of this potentially higher stress:

   - The first is to consider actual plant performance. The occurrence of the waterhammer following a LOOP event - either simulated in a test or real - is known to have occurred many times in the industry. The waterhammer following a LOOP-only event is not cushioned by gas and steam in the void. The total number of occurrences of LOOP-only events is estimated to be on the order of at least several hundred, based on a review of the available plant data. These occurrences have all been in open loop plants and are more severe than a waterhammer that would occur following a LOOP/LOCA event. Without any cushioning, the LOOP waterhammer is more severe than that following a LOOP/LOCA. No piping failures have occurred in any of these events. This would indicate that the probability of failure for a more severe waterhammer (an uncushioned waterhammer) is of the order of or lower.

The other method is to take the ASME Code limits and to calculate the probability of failure if the code limits were to be exceeded by approximately 40%. For the purpose of this evaluation, it will be assumed that the piping system is designed so that all the ASME code stresses in the piping were at the faulted condition limit when the cushioned waterhammer occurred (i.e., the EPRI methodology is used and that the pipe was designed up to the code acceptable limit for that load). To determine probability of failure, an assumed stress distribution is used around a stress that is 40% larger than the faulted allowable ($2.4 \cdot S_h$) and compared to the actual tested material strengths for A106-Gr B piping. Based on the actual margins available in the ASME code (see NUREG/CR-2137), the probability of the stress exceeding the strength can be shown to be on the order of 1E-04 or less.

For the purpose of continuing the "event progression", a probability of failure in the pipe if the cushioned waterhammer were exceeded will be taken to be on the order of 1E-02. It is probably much less likely.

9. **Likelihood of an Unacceptable Event**
   Given the low probability (1E-05/year) of the initiating events and the low probability (1E-02) of piping failure, the use of the methodology in the User's Manual and the Technical Basis Report will lead to a likelihood of an unacceptable event that is on the order of 1E-07. Again, for the purposes of this evaluation, the "unacceptable event" following a LOOP/LOCA event is taken as a breach of the service water system pressure boundary. The probability of 1E-07 for this event is below the threshold for significant risk to the plant. Use of the methods in the User's Manual, therefore, will not compromise the safety of the plant for the systems within the bound provided in the User's Manual and Technical Basis Report. The methodology should be accepted as recommended in the report.

The most important consideration in the behavior of the waterhammer following pump restart is that there is an upper bound on the waterhammer pressure that can be attained – the waterhammer without cushioning -- and that the waterhammer without cushioning has occurred many times in simulated LOOP events. The methods proposed in the EPRI Technical Basis Report use the physics of gas compression to calculate a reduced closure velocity and waterhammer magnitude. The change in risk introduced by the use of these methods is not significant and the methods do not lead to an unacceptable plant risk following a LOOP/LOCA event. Hence, from the risk-informed perspective, the methods proposed in the submitted EPRI Technical Basis Report and User's Manual are adequate for plant-specific application for resolution of the GL 96-06 issues.

The methods provided in NUREG 5220 were considered acceptable for conservatively analyzing waterhammer events. The NUREG uses the Joukowski relationship with the uncushioned velocity. NUREG-5220 acknowledges that the calculated results using their methods could be 2 - 10 times higher than reality. All appear to agree that if a void forms following a LOOP/LOCA event, it will contain non-condensable gas and some steam; some cushioning will occur. The emphasis of the work that was performed by EPRI was to define the amount of cushioning that would be expected. The cushioning calculated will provide, in general, between zero and approximately 40% velocity reduction and subsequent pressure reduction. The risk discussion above concludes that, given the low probability of the events, the limited energy available from this event, and the low probability of pipe failure, the methodology proposed in the User's Manual is reasonable for those cases that fit within the parameters of the User's Manual and Technical Basis Report.