NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

April 29, 2005

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10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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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 the Point Beach Nuclear Plant
License Renewal Application
(TAC Nos. MC2099 and MC2100)

By letter dated February 25, 2004, Nuclear Management Company, LLC (NMC), submitted the Point Beach Nuclear Plant (PBNP) Units 1 and 2 License Renewal Application (LRA). On March 31, 2005, the Nuclear Regulatory Commission (NRC) requested additional information regarding Aging Management of Auxiliary Systems (Section 3.3 of the LRA), and Aging Management of Containments, Structures and Component Supports (Section 3.5 of the LRA). The enclosure to this letter contains NMC's response to the staff's questions.

Should you have any questions concerning this submittal, please contact Mr. James E. Knorr at (920) 755-6863.

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

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

Dennis L. Koehl

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

Enclosure

A093

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

ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) regarding the Point Beach Nuclear Plant (PBNP) License Renewal Application (LRA).

The NRC staff's questions are restated below with the Nuclear Management Company (NMC) response following.

Aging Management of Auxiliary Systems

NRC Question RAI 3.3-7:

In LRA Table 3.3.2-2 (Page 3-221), the applicant proposed to manage cracking due to IGA/IGSCC of stainless steel in heat exchanges exposed to primary treated water with T>480°F (internal) using the Water Chemistry Control Program. This line item cites Note 35, which states "Component/material/environment is not addressed in the corresponding NUREG-1801 Chapter, but the component/material/environment is addressed in another NUREG-1801 Chapter." The AMR line item references AMR line item 3.1.1-36 which provides the following discussion:

> "Crack initiation and growth due to SCC and flaw growth are identified as aging effects requiring management for the reactor vessel nozzle safe ends, CRD housing, and RCS components. Aging management programs credited for managing these effects are the Water Chemistry Control Program and ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program."

The Note implies that ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program should have also been applied to LRA Table 3.3.2-2. The staff requests the applicant to explain this discrepancy or make a commitment to revise the line item in LRA Table 3.3.2-2 to include the Inservice Inspection Program.

NMC Response:

The heat exchangers represented by the line item in Table 3.3.2-2 referenced above are the sample heat exchangers. These heat exchangers are small helicoil-type heat exchangers, for which there is no practical way to inspect the tubes. Since NUREG-1801 did not specifically address sample heat exchangers, a line item from NUREG-1801 that had the same material, environment and aging effects was selected. Line item 3.1.1-36 was selected as it included Primary Water for an Environment. The discussion section of line item 3.1.1-36 includes the paragraph quoted in NRC Question RAI 3.3-7 but also includes the following paragraph:

> "The ISI Program inspects critical component locations as determined by the ASME Code approved by the NRC. Thus, it may be noted that some components (manway inserts, thermal sleeves, SG divider plate) are not inspected by the ISI Program, and therefore ISI is not credited for these select components. Plant specific operating experience has not identified these aging effects to-date, and therefore the Water Chemistry Control Program alone has proven to be effective in managing these aging effects."

The referenced sample heat exchangers are not required to be inspected by the ISI Program, and therefore only the Water Chemistry Control Program was selected to manage the aging. Note that this is consistent with a recent safety evaluation report (dated March 2005), where the applicant (also a Westinghouse PWR) credited only the Water Chemistry Control Program to manage cracking in the stainless tubing of a sample heat exchanger.

Aging Management of Containments, Structures and Component Supports

NRC Question RAI 3.5-14:

(1) In LRA Table 3.5.1, Item 3.5.1-33, the applicant stated that the Bolting Integrity Program includes the use of Inservice Inspections to evaluate and monitor crack initiation and growth due to SSC, if present, in high strength low-alloy steel bolts used in NSSS component support.

In LRA Tables 3.5.2-1 through 3.5.2-14, the applicant does not address Group B1.1, high strength low-alloy bolts.

In LRA Section B2.1.4, the applicant indicated that high strength component support bolting is used in pinned connections associated with steam, reactor coolant pumps and reactor vessel supports and is loaded only in shear with no preload stress.

No preload stress indicates that PBNP NSSS components bolting were installed with snug-tight only, without using any torque or turn-of nuts method during installation. The staff requests the applicant to provide justification for the statement of no preload stress for NSSS components bolting. The applicant is requested to clarify if there are any Group B1.1, high strength low-alloy bolts, not used for pinned connection only.

Additionally, the applicant's Bolting Integrity Program does not support the discussion in LRA Table 3.5.1, Item 3.5.1-33. The staff requests the applicant to identify the aging management for the Group B1.1, high strength low-alloy bolts.

NMC Response:

The LRA Table 3.5.1, line item 3.5.1-33 pertains to high strength low-alloy ASME Class 1 component bolting. This component type is found only in LRA Table 3.5.2-10, "Structures and Component Supports-Component Supports Commodity Group-Summary of Aging Management Evaluation."

The aging effect/mechanism associated with line item 3.5.1-33 is "crack initiation and growth due to SCC." This has been evaluated as not being an applicable aging effect requiring management at PBNP. The aging effects requiring management for Group B1.1 bolting include: line item 3.5.1-31 - loss of material due to boric acid wastage and line item 3.5.1-32 - loss of material due to general corrosion. These aging effects are depicted in LRA Table 3.5.2-10, which addresses Group B1.1 bolting. The intent of the Discussion column for line item 3.5.1-33 was that *"if present"*, cracking may be detected during ISI inspections which would evaluate any noted non-conformance.

Cracking of high strength structural bolting materials has been attributed to stress corrosion cracking (SCC), line item 3.5.1-33. The materials of concern are high nickel maraging steels or low-alloy quenched and tempered (LAQT) steels, (i.e., yield strength S_y >150 ksi). The bolting material specification in question at PBNP is ASTM A490, a LAQT steel. Three parameters are required for SCC to occur: (1) an environment with contaminants, (2) a susceptible material, and (3) high sustained tensile stresses.

A listing of reported failures of structural bolting used in Class 1 components supports is provided in Table 4-1 of EPRI NP-5769, Volume 1, "Degradation and Failure of Bolting in Nuclear Power Plants," April 1988. Common features of the failures by SCC were that high strength or overly hard materials were used in moist environments under high sustained tensile stresses. The majority of the bolting failures were for ultra high strength materials. Problems with A490 material, a medium strength bolt, has always had failure contributors, such as improper heat treatment or moist environment.

Contaminants such as sulfates, fluorides, or chlorides can provide the necessary environment for SCC. The containment environment experienced by the bolts is considered benign with respect to SCC, i.e., the bolts are located in a dry environment high up above any source of leakage and, therefore, not exposed to an aggressive or aqueous environment. In the past, SCC in the presence of bolt lubricants has been a problem. NMC controls the selection of lubricants to preclude lubricant-induced SCC.

Structural bolt preload (or tension) is in accordance with the AISC specification. Structural bolt preload has been employed with NSSS component bolting at PBNP. These components are age-managed in accordance with LRA Table 3.5.1, line item 3.5.1-32 (the ASME Section XI, Subsection IWF Inservice Inspection Program). The major RCS component supports at PBNP are configured with a pinned leg arrangement. The pin, 4 inches in diameter, is made of A490 bolt material but is not a true bolt and was not preloaded as a normal bolted connection would have been.

In summary, SCC is not an aging effect for high strength low-alloy component Group B1.1 bolting used in the supports of NSSS components at PBNP. A review of industry failure databases and NRC generic communications support the fact that a combination of material selection, control of contaminants, and proper torquing has been effective in eliminating the potential for SCC of bolting materials. This conclusion is based on the totality of the information presented, including plant operating experience.

Nonetheless, the Bolting Integrity Program includes the use of ISI to evaluate and monitor crack initiation and growth due to SCC, if present, in Group B1.1 high strength low-alloy steel bolts used in NSSS component supports at PBNP (Table 3.5.1, line item 3.5.1-33).