August 19, 2005

NRC 2005-0107
10 CFR 2.390

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

Request for Withholding of Proprietary Information from Public Disclosure

Reference: Letter from NMC to NRC dated October 28, 2004 (NRC 2004-0113)

In the reference, Nuclear Management Company, LLC (NMC), submitted a copy of WCAP-14000, Revision 2, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: Point Beach Units 1 & 2", dated September 2004 (Proprietary) and a copy of WCAP-15950, Revision 2, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: Point Beach Units 1 & 2," dated September 2004 (Non-Proprietary). These documents were provided in support of a request for Nuclear Regulatory Commission (NRC) review and approval of relaxation from certain requirements of NRC Order EA-03-009, for the Point Beach Nuclear Plant, Unit 1.

These two documents were subsequently found to contain a non-proprietary listing of references and an explanation of three types of proprietary classification categories that were not appropriately indicated as non-proprietary. Consequently, Westinghouse Electric Company ("Westinghouse") issued errata pages to correct these discrepancies. This letter submits the errata pages as page-for-page replacements for the documents provided in the reference. As such, the NRC may release for public disclosure, the revised list of references and the explanation of the three types of proprietary classification categories, which are submitted herein for incorporation into WCAP-15950.

Enclosed with this letter are errata pages for WCAP-14000, Revision 2, and for WCAP-15950, Revision 2. Replacement instructions for the errata pages are included in the enclosures.

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

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 19, 2005.

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

Enclosures (2)

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

ENCLOSURE 1

ERRATA PAGES
WCAP-14000, "STRUCTURAL INTEGRITY EVALUATION OF REACTOR VESSEL UPPER HEAD PENETRATIONS TO SUPPORT CONTINUED OPERATION: POINT BEACH UNITS 1 & 2" REVISION 2

Page Replacement Instructions

Please remove and discard the indicated pages in WCAP-14000, Revision 2, and replace them with the enclosed pages as instructed below.

Remove and Discard	Insert
September 2004, Page 8-1	Errata, August 2005, Page 8-1
September 2004, Page 8-2	Errata, August 2005, Page 8-2

8 REFERENCES

1. Scott, P. M., "An Analysis of Primary Water Stress Corrosion Cracking in PWR Steam Generators," in Proceedings, Specialists Meeting on Operating Experience With Steam Generators, Brussels Belgium, Sept. 1991, pages 5, 6.

2. McIlree, A. R., Rebak, R. B., Smialowska, S., "Relationship of Stress Intensity to Crack Growth Rate of Alloy 600 in Primary Water," Proceedings International Symposium Fontevraud II, Vol, 1, p. 258-267, September 10-14, 1990.

3. Cassagne, T., Gelpi, A., "Measurements of Crack Propagation Rates on Alloy 600 Tubes in PWR Primary Water," in Proceedings of the 5th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors," August 25-29, 1991, Monterey, California.

4A. *Crack Growth and Microstructural Characterization of Alloy 600 PWR Vessel Head Penetration Materials,* EPRI, Palo Alto, CA. 1997. TR-109136.

4B. Vaillant, F. and C. Amzallag. "Crack Growth Rates of Alloy 600 in Primary Water," Presentation to the EPRI-MRP Crack Growth Rate (CGR) Review Team, Lake Tahoe, NV, August 10, 2001.

4C. Vaillant, F. and S. Le Hong. *Crack Growth Rate Measurements in Primary Water of Pressure Vessel Penetrations in Alloy 600 and Weld Metal 182,* EDF, April 1997. HT-44/96/024/A.

4D. Framatome laboratory data provided by C. Amzallag (EDF) to MRP Crack Growth Rate Review Team, October 4, 2001 (Proprietary to EDF).

4E. Cassagne, T., D. Caron, J. Daret, and Y. Lefevre. "Stress Corrosion Crack Growth Rate Measurements in Alloys 600 and 182 in Primary Water Loops Under Constant Load," *Ninth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors* (Newport Beach, CA, August 1-5, 1999), Edited by F. P. Ford, S. M. Bruemmer, and G. S. Was, The Minerals, Metals & Materials Society (TMS), Warrendale, PA, 1999.

4F. Studsvik laboratory data provided by Anders Jenssen (Studsvik) to MRP Crack Growth Rate Review Team, October 3, 2001 (Proprietary to Studsvik).

4G. Bamford, W. H., "D. C. Cook Unit 2 Upper Head Penetration Crack Growth Determined from Inspection Data," Westinghouse Electric Report LTR-SMT-01-72, November 2001.

4H. Materials Reliability Program (MRP) Recommended Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick Wall Alloy 600 Material," EPRI MRP Report 55, May 30, 2002.

5A. Newman, J. C. and Raju, I. S., "Stress Intensity Factor Influence Coefficients for Internal and External Surface Cracks in Cylindrical Vessels," in Aspects of Fracture Mechanics in Pressure Vessels and Piping, PVP Vol. 58, ASME, 1982, pp. 37-48.

5B. Hiser, Allen, "Deterministic and Probabilistic Assessments," presentation at NRC/Industry/ACRS meeting, November 8, 2001.

6A. Fleming, Mark A., "Stress Analysis for Head Penetrations of a 2 Loop Westinghouse RV Constructed by B&W," Dominion Engineering Calculation No. C-7744-00-1, June 2002.

6B. Fleming, Mark A., "Farley CRDM and Head Vent Stress Analysis," Dominion Engineering Calculation No. C-7744-02-1, July 2002.

7. USNRC Letter, W. T. Russell to W. Raisin, NUMARC, "Safety Evaluation for Potential Reactor Vessel Head Adapter Tube Cracking," November 19, 1993.

8. USNRC Letter, A. G. Hansen to R. E. Link, "Acceptance Criteria for Control Rod Drive Mechanism Penetrations at Point Beach Nuclear Plant, Unit 1," March 9, 1994.

9. Westinghouse Letter, LTR-EMT-02-258, K. B. Neubert, "Point Beach Units 1 and 2 – Upper Head Mean Fluid Temperature for Flaw Handbook Calculation," September 16, 2002.

ENCLOSURE 2

ERRATA PAGES
WCAP-15950, "STRUCTURAL INTEGRITY EVALUATION OF REACTOR VESSEL UPPER HEAD PENETRATIONS TO SUPPORT CONTINUED OPERATION: POINT BEACH UNITS 1 & 2" REVISION 2

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September 2004, Page 8-1	Errata, August 2005, Page 8-1
September 2004, Page 8-2	Errata, August 2005, Page 8-2

1 INTRODUCTION

In September of 1991, a leak was discovered in the reactor vessel control rod drive (CRDM) head penetration region of an operating plant. This has led to the question of whether such a case could occur at Point Beach Units 1 and 2. The geometry of interest is shown in Figure 1-1. Throughout this report, the penetration rows have been identified by their angle of intersection with the head. For each penetration of each unit, the angle is identified in Table 1-1 for Units 1 and 2.

The issue resulted from cracking, which occurred in the outermost penetrations of a number of operating plants, as discussed in Section 2. This outermost CRDM location, as well as a number of intermediate CRDM penetrations, the head vent, and the center penetration were chosen for fracture mechanics analyses to support continued safe operation of Point Beach Units 1 and 2 if such cracking were to be found. The dimensions of the CRDM penetrations are all identical, with 4.0 inch outside surface (OD) and wall thickness of 0.625 inches. For the head vent, the OD is 1.014 inches, and the wall thickness is 0.122 inches.

The basis of the analyses was a detailed three dimensional elastic-plastic finite element analysis of several penetration locations, as described in detail in Section 5. Results were obtained at a number of locations in each penetration, and used in the fracture analysis.

The fracture analyses were carried out using reference crack growth rates recommended by the EPRI Materials Reliability Project, which are consistent with service experience. The results are presented in the form of flaw evaluation charts for both surface flaws and through wall flaws, to determine the allowable time of safe operation if indications are found. All the times calculated in this handbook are effective full power years (EFPY).

Note that there are several locations in this report where proprietary information has been identified and bracketed. For each of the bracketed locations, the reason for the proprietary classification is given, using a standardized system. The proprietary brackets are labeled with three different letters, to provide this information, and the explanation for each letter is given below:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by Westinghouse's competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

c. The information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

e. The information reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

8 REFERENCES

1. Scott, P. M., "An Analysis of Primary Water Stress Corrosion Cracking in PWR Steam Generators," in Proceedings, Specialists Meeting on Operating Experience With Steam Generators, Brussels Belgium, Sept. 1991, pages 5, 6.

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