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

March 9, 2007

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U.S. Nuclear Regulatory Commission
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Point Beach Nuclear Plant, Unit 1
Docket 50-266
Renewed License No. DPR-24

License Amendment Request 248
Technical Specification 5.5.8, Steam Generator Program
Response to Request for Additional Information

References: (1) NMC to NRC Letter Dated July 11, 2006, License Amendment Request 248, Technical Specification 5.5.8, Steam Generator Program, (ML062050338)
(2) NMC to NRC Letter Dated January 19, 2007, Supplement 1 to License Amendment Request 248, Technical Specification 5.5.8, Steam Generator Program (ML070220084)
(3) NRC to NMC Letter Dated February 27, 2007, Request for Additional Information, License Amendment Request, Steam Generator Program (ML07052461)

By References (1) and (2) above Nuclear Management Company, LLC (NMC) submitted a proposed one-time amendment to the Technical Specifications (TSs) for Point Beach Nuclear Plant, Unit 1. The proposed changes would revise TS 5.5.8, "Steam Generator (SG) Program," to change the repair criteria for the portion of the tubes within the hot-leg region of the tubesheet for a single operating cycle following Refueling Outage 30. The proposed amendment defines a distance downward into the hot leg tubesheet, below which flaws may remain in service regardless of size. As a result, tube inspection within the hot leg region would be required only within 17 inches of the top of the tubesheet.

Reference (3) transmitted a request to NMC for additional information regarding the proposed amendment. NMC's response to this request for additional information is contained in the enclosure to this letter.

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

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 9, 2007.

Dennis L. Koehl

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

Enclosure

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

ENCLOSURE
REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 248
TS 5.5.8, STEAM GENERATOR PROGRAM

Background

By References (1) and (2), Nuclear Management Company, LLC (NMC) submitted a proposed amendment to the Technical Specifications (TSs) for the Point Beach Nuclear Plant (PBNP), Unit 1. The proposed changes would revise TS 5.5.8, "Steam Generator (SG) Program," to change the repair criteria for the portion of the tubes within the hot-leg region of the tubesheet for a single operating cycle following Refueling Outage 30. The proposed amendment defines a distance downward into the hot-leg tubesheet, below which flaws may remain in service regardless of size. As a result, tube inspection within the hot-leg region would be required only within 17 inches of the top of the tubesheet.

Reference (3) transmitted a request for additional information to NMC that provides the results of studies performed by Argonne National Laboratory (ANL) relating to the leakage behavior of steam generator tube-to-tubesheet joints under postulated severe accident conditions. As part of this work ANL benchmarked its finite element model of the tube-to-tubesheet joint against pullout and leakage tests carried out by Westinghouse on tube-to-collar joint specimens for the Callaway Plant.

Reference (3) states that ANL observed that the distribution of contact pressure (between the tube and tubesheet) caused by differential thermal expansion effects and by the roll expansion process (calculated indirectly by Westinghouse based on the test results), is much different than that calculated by ANL. ANL states that this may be due to an incorrect choice of thermal expansion coefficients for the tubesheet collar specimens used in the Westinghouse tests. The test specimen collars were made from cold-worked 1018 steel instead of the actual tubesheet material, which is forged SA-508 steel. Although the yield strengths of these two materials are comparable, their coefficients of thermal expansion are somewhat different. ANL believes the ANL results differ from the Westinghouse results because Westinghouse assumed thermal expansion properties of SA-508 rather than 1018 steel when analyzing their test results.

The Request for Additional Information states that the Westinghouse tests for Callaway are similar to those performed in support of the proposed amendment for PBNP Unit 1. The July 11, 2006, NMC letter, enclosed Westinghouse report LTR-CDME-05-201-P, Revision 1, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Point Beach Unit 1." The NRC staff noted that the pullout and leakage tests described in that report also utilized collar specimens fabricated from 1018 steel and not the SA-508 steel from which the PBNP Unit 1 tubesheets were actually fabricated.

NRC Request 1

"Provide any comments on ANL's observation regarding the possibility of an incorrect choice of thermal expansion coefficient for the tubesheet collar specimens used in the Westinghouse tests in terms of how it may apply to Point Beach 1. If you should disagree with the ANL observations pertaining to Callaway or if you believe the ANL observations are not relevant to Point Beach 1, provide an explanation."

NMC Response

Reference (1) transmitted Westinghouse report LTR-CDME-05-201-P, Revision 1, "Limited Inspection of the Steam Generator Tube Portion within the Tubesheet at Point Beach Unit 1." This report does not use tube pullout test results to establish the minimum length of engagement (H*) distance as it is conservatively assumed that there is zero residual strength in the joint. However, the submittal does use results from previous leakage tests performed by the vendor to address the relationship between flow resistance and contact pressure. The application of a lower value for coefficient of thermal expansion (CTE) yields a more conservative value used for flow resistance versus contact pressure. However, the conclusion that the leakage rate during a postulated steam line break (SLB) event would be bounded by twice the leak rate that is present during normal operating conditions does not change. Additional detail is provided below.

The possible impact of the use of a different CTE for the tube pullout test specimen is two-fold: 1) the effective residual contact pressure due to the hydraulic expansion process is semi-empirically determined based on tube pull test results and typically would be affected as the effect of temperature is analytically subtracted from the pullout test results; and 2) the correlation between crevice loss coefficient as a function of contact pressure is established using elevated temperature leakage testing.

The first possible impact is not applicable to PBNP. The structural analysis of the tube-to-tubesheet joint for PBNP conservatively assumed no residual contact pressure due to the hydraulic expansion process [Page 13 of Enclosure 5 to Reference (1)]. The H* distances defined in Tables A.2-1 and A.2-2 of LTR-CDME-05-201-P are conservatively established assuming no residual contact pressure in the steam generator tube joints as a result of the hydraulic expansion process. The H* distance is based solely on interference loads resulting from internal pressure in the tube that is transmitted from the inner diameter to the outer diameter, thermal expansion of the tube relative to the tubesheet, and bowing of the tubesheet that results in dilation of the tubesheet holes above the neutral axis of the TS.

The second impact is applicable to PBNP Unit 1. Reference (1) used results from leakage tests that establish the correlation between crevice loss coefficient as a function of contact pressure. This was used to discuss the increase in flow resistance that occurs during a postulated steam line break event relative to the normal operating condition at an elevation of 17 inches below the top of the tubesheet near the center of the tubesheet (See the response to Request 2 below).

NRC Request 2

"If there is a problem with the assumed values of thermal expansion coefficient, describe the problem and provide any necessary corrections to Westinghouse report LTR-CDME-05-201-P, Revision 1. (To support timely completion of the NRC staff's review, the items of highest priority to the staff are any revisions to Figures 5-10, Tables A.2-1 and A.2-2, and Figures A.1-2 and A.1-3.)"

NMC Response

No changes are required in the figures or tables attached to LTR-CDME-05-201-P because these attachments were calculated using the correct coefficient of thermal expansion for SA-508. However, LTR-CDME-05-201-P states on Page 16 of 52 that the flow resistance at a depth of 17 inches from the top of the tubesheet would be expected to increase by about 60% due to the difference in contact pressure between normal operating and steam line break conditions. While the revised analysis has not been completed for PBNP, based on a comparative study with another plant of similar design as PBNP, using the ANL CTE at 600°F (ANL letter from S. Majumdar to distribution, "Analysis of Callaway Pullout and Leakage Tests," dated December 29, 2006), the vendor estimated that the 60% increase in resistance would, at most, be reduced to approximately a 20% increase in flow resistance. The reduction in resistance occurs due to a needed adjustment to the contact pressures calculated for the elevated temperature leak rate testing. The results were then used to establish the relationship between leak loss coefficient and contact pressure. The vendor concluded that the leak rate for indications below a depth of 17 inches from the top of the tubesheet during a postulated steam line break event would be bounded by twice the leak rate that is present during normal operating conditions for PBNP Unit 1. Since the driving head at SLB conditions is approximately twice that at normal operating conditions, and the resistance to flow at SLB conditions is greater than at normal operating conditions, the SLB leak rate must be less than twice the normal operating condition leak rate.

Based on the above, the conclusions of the LTR-CDME-05-201-P continue to apply.

NRC Request 3

"Provide any revisions or corrections to LTR-CDME-05-201-P, Revision 1, to reflect any new Westinghouse crevice pressure test data and analyses that may be relevant to tube repair criteria and inspection in the tubesheet region at Point Beach 1. (To support timely completion of the staff's review, the items of highest priority to the staff are any revisions to Figures 5-10, Tables A.2-1 and A.2-2, and Figures A.1-2 and A.1-3.)"

NMC Response

The vendor made a comparison of the PBNP Unit 1 data with the analysis for the H. B. Robinson plant (Docket No. 50-261), which also has Model 44F steam generators. That comparison determines the effect of the new Westinghouse crevice pressure data and the analysis that is provided in LTR-CDME-05-201-P for PBNP Unit 1. The H. B. Robinson analysis was provided to the NRC (ML062890083). It was shown for H. B. Robinson that the effect of using crevice pressure ratios of 0.3686 and 0.6977 for SLB and normal operating conditions, and assuming that the divider plate is "non-functional," results in a increase in H* distance from 8.34 inches to 11.81 inches for the limiting condition of a postulated SLB event. Note that the original limiting H* inspection depth of 12.1 inches for PBNP (LTR-CDME-05-201-P) remains conservative with respect to the revised analyses for the comparable plant that are based on the updated analysis which include the effect of new crevice pressure data and assume a "non-functional" divider plate. In this context, "non-functional" refers only to the ability of the divider plate to restrict vertical displacements of the tubesheet.

This result is expected to be similar for the PBNP Unit 1 steam generators if tube pull results were used to establish the limiting distance of 12.1 inches included in LTR-CDME-05-201-P. In addition, the vendor concluded for H. B. Robinson that the leak rate for indications below the depth of 17 inches from the top of the tubesheet would be bounded by twice the leak rate that is present during normal operating conditions. This is also expected to be the case for the PBNP Unit 1 steam generators when the updated analysis is completed prior to submittal of the permanent B* license amendment request.