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10 CFR 50.90

RA-08-062
June 30, 2008

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Oyster Creek Nuclear Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Response to Request for Additional Information Concerning Technical Specification Change Request No. 348 – Relocation of Pressure and Temperature (P-T) Curves to the Pressure and Temperature Limits Report (PTLR)

Reference: Letter from P. B. Cowan (AmerGen Energy Company, LLC) to U.S. Nuclear Regulatory Commission, "Technical Specification Change Request No. 348 – Relocation of Pressure and Temperature (P-T) Curves to the Pressure and Temperature Limits Report (PTLR)," dated March 10, 2008

In the referenced letter AmerGen Energy Company, LLC (AmerGen) requested a change to the Technical Specifications. The proposed change modifies Technical Specifications (TS) Section 1.0 ("Definitions"), Limiting Conditions for Operation Section 3.3 ("Reactor Coolant"), Surveillance Requirement 4.3 ("Reactor Coolant"), and 6.0 ("Administrative Controls") to delete reference to the pressure and temperature curves, and include reference to the Pressure and Temperature Limits Report (PTLR).

As a result of additional discussions with the U.S. Nuclear Regulatory Commission (U.S. NRC) staff on June 19, 2008, additional information has been requested. Enclosure 1 contains our response to this request.

We note that license renewal commitment No. 46 requires the submittal of Pressure-Temperature curves for 50 Effective Full Power Years (EFPYs) for U.S. NRC review and approval. These curves are contained in Enclosure 2. In accordance with the commitment, we request your review and approval of the 50 EFPY curves by June 30, 2009. The requested date for review and approval of the reference submittal remains as October 15, 2008, as stated in the reference submittal.

If any additional information is needed, please contact Tom Loomis at (610) 765-5510.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th of June, 2008.

Respectfully,

A008

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Enclosures: (1) Response to Request for Additional Information
(2) "Revised P-T Curves Based on New Fluence" (File No. OC-05Q-313), Revision 3
(3) "Feedwater Nozzle Green's Functions" (File No. OC-05Q-307), Revision 0

cc: S. J. Collins, Administrator, USNRC Region I
G. E. Miller, USNRC Project Manager, Oyster Creek
M. S. Ferdas, USNRC Senior Resident Inspector, Oyster Creek
Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection

ENCLOSURE 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Response to Request for Additional Information Regarding Technical Specification Change Request No. 348 – Relocation of Pressure and Temperature (P-T) Curves to the Pressure and Temperature Limits Report (PTLR)

QUESTION:

1) To verify that Oyster Creek reactor pressure vessel (RPV) pressure-temperature (P-T) limits in the proposed Pressure-Temperature Limits Report (PTLR) are results of correctly implementing the methodology of SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," the staff requests you:

   a. Provide an evaluation showing that you have performed analysis for the bottom head region and the non-beltline region in accordance with SIR-05-044-A and identify the portion of the proposed P-T limits (Figures 1 to 3 of the proposed PTLR are sufficient) that are limited by materials in these regions,

RESPONSE:

Refer to Section 4.1.2 (Curve A) and 4.2.2 (Curve B) contained in Enclosure 2 ("Revised P-T Curves Based on New Fluence" (File No. OC-05Q-313), Revision 3) for the evaluation of the bottom head region. Refer to Section 4.1.3 (Curve A) and 4.2.3 (Curve B) of Enclosure 2 for the evaluation of the non-beltline (feedwater nozzle/upper vessel) region.

QUESTION:

   b. Provide an evaluation for the small-diameter, drill-hole type instrument nozzle (e.g., water level nozzle) if it exists in your RPV beltline,

RESPONSE:

The Oyster Creek Nuclear Generating Station (OCNGS) Reactor Pressure Vessel (RPV) small-diameter, drill-hole type instrument nozzles (identified as N13B through N17B) have been evaluated with respect to P-T limits. A review of the location and fluence level of the N13B through N17B instrument nozzles (1 inch diameter) closest to the beltline area of the RPV was performed. The RAMA fluence evaluation for the OCNGS RPV identifies that the maximum elevation in the RPV which is exposed to a fluence equal to or greater than 1.0×10^{17} n/cm^2 at 50 EFPY fluence is 372.20 inches above vessel zero (inside surface of the bottom head). The N13B through N17B instrument nozzles are located at 422 inches or higher above vessel zero. Therefore, they are at least four feet above the elevation in the RPV that is considered to be the beltline region. The beltline region is defined by a fluence level of 1.0×10^{17} n/cm^2, which is the limit where fluence effects are considered insignificant consistent with Section III.A of 10CFR50 Appendix H. Therefore, the OCNGS instrument nozzles lie outside of the beltline, and are covered by the feedwater nozzle/upper vessel (non-beltline) region. The upper vessel non-beltline region is controlled by the limiting RT_{NDT} of the CRD return nozzle, combined with the limiting stresses of the feedwater nozzles, which collectively bound the instrument nozzle properties (also refer to the discussion in Section 3.2.2 of Enclosure 2).

QUESTION:

c. Identify among the three methodologies (Page 2-13 of SIR-05-044-A) the one that you used to calculate thermal stress intensity factors for shell regions,

RESPONSE:

As indicated in Section 4.2.1 of Enclosure 2, the "Section XI Nonmandatory Appendix G Method" specified in SIR-05-044-A was used to calculate thermal stress intensity factors for shell regions.

QUESTION:

d. Provide the temperature instrument uncertainty, the pressure instrument uncertainty, and the pressure head to account for the column of water in the RPV (Page 2-25 of SIR-05-044-A) so that the staff can assess the difference between the staff's calculated P-T limits and your proposed P-T limits, and;

RESPONSE:

Refer to Section 3.2.3 and pages 8, 10, 12, 14, 16, and 19 of Enclosure 2 for the pressure instrument uncertainty (0 psig), temperature instrument uncertainty (0°F), and pressure head to account for the column of water in the RPV (20.1 psig).

QUESTION:

e. Provide Reference 6.16 (a reference listed in the proposed PTLR), "Revised Calculation of P-T Limit Curves for the Oyster Creek Generating Station," dated February 26, 2008, to supplement the above requested specific information.

RESPONSE:

Reference 6.16 is provided in Enclosure 2.

QUESTION:

2) Page 7 of the PTLR mentioned that, "With respect to operating conditions, stress distributions were developed for a thermal shock of 450°F, which represents the maximum thermal shock for the feedwater nozzle during normal operating conditions." The only guideline in SIR-05-044-A regarding the thermal stress intensity factor calculation for the feedwater nozzle is that the stress distribution should be extracted from a finite element model using the limiting normal/upset transient. Provide information regarding whether your feedwater nozzle analysis is plant-specific and how the 450°F thermal shock transient was selected and determined to be bounding.

RESPONSE:

As discussed briefly in Section 3.2.2 of Enclosure 2, the stress intensity factor calculation is based on the stress results for the limiting normal/upset transient for the feedwater nozzle using a plant-specific finite element model of the Oyster Creek feedwater nozzle (Enclosure 3 – "Feedwater Nozzle Green's Function," OC-05Q-307, Revision 0 (referred to as Reference 6.7 in the PTLR)). Although Enclosure 3 was developed to establish fatigue monitoring inputs, the limiting nozzle corner hoop stresses were extracted and used to determine polynomial fits of the pressure and thermal hoop stresses for use with the SIR-05-044-A methodology, as described in Section 4.1.3 and 4.2.3 of Enclosure 2. The thermal transient evaluated in Enclosure 3 is equivalent to the limiting normal/upset design transient for a BWR feedwater nozzle, which is a Turbine Roll event that represents initiation of feedwater flow into the reactor pressure vessel (RPV). This event is assumed to occur immediately after RPV heatup to rated temperature and pressure, where cold (unheated = 100°F) feedwater is injected into the hot (550°F) RPV. Because the transient is an injection event, the transient is assumed to be an instantaneous (shock) event for the nozzle. All other normal/upset events occur either at slower rates or from less bounding temperatures (because of the presence of heated feedwater or lower RPV temperatures), thereby making the Turbine Roll event the most severe event for the feedwater nozzle.

QUESTION:

3) Pages 7 and 8 provided some additional information regarding the finite element modeling of the feedwater nozzle for evaluating P-T limits for non-beltline materials that were not in SIR-05-044-A. Please justify the following:

   a. Use of a conversion factor of 3.2 times the cylinder radius to model the sphere (upper head), and

RESPONSE:

The conversion factor of 3.2 does not apply to the upper head but rather is a factor to correct for modeling the feedwater nozzle using an axisymmetric model (refer to Section 4.1 of Enclosure 3). An axisymmetric model causes the cylindrical reactor pressure vessel to be modeled as spherical, which therefore needs correction to properly estimate pressure hoop stresses in the nozzle-to-vessel intersection due to the geometric discontinuity caused by two intersecting cylinders. Increasing the modeled radius of the RPV wall by a multiplier of 3.2 will increase pressure hoop stress by a factor of 3.2. The value of 3.2 was selected consistent with the current licensing basis (CLB) stress report for the feedwater nozzle (Reference 1 of Enclosure 3: MPR Report No. MPR-783, "Oyster Creek Nuclear Generating Station Evaluation of Low Flow Feedwater Control System," August 1983), which selected the value of 3.2 based on a paper, J. B. Truitt and P. D. Raju, "Three Dimensional Versus Axisymmetric Finite Element Analysis of Cylindrical Vessel Inlet Nozzle Subject to Internal Pressure – A Comparative Study," ASME Transactions, Paper No. 78-PVP-6. The value of 3.2 is a conservative multiplier, as more recent studies of BWR nozzles using three-dimensional finite element models (Reference: BWRVIP-108: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and

Nozzle Blend Radii, EPRI, Palo Alto, CA 2002, 1003557) demonstrate this factor to be an average value of approximately 2.6 (see Table 1).

Table 1: Pressure Hoop Stress Multiplier to Correct Axisymmetric to 3-D

Nozzle	Vessel Inside Radius R, in	Vessel Thickness t, in	Pressure Hoop Stress PR_m/t, psi*	3-D Model Stress at Corner, psi	Ratio	BWRVIP-108 Reference
Core Spray	106.7188	7.125	15,478	40,594	2.623	Table 4-1 and Figure 4-22
Main Steam	106.7188	7.125	15,478	40,207	2.598	Table 4-1 and Figure 4-24
Recirc Inlet	110.00	5.69	19,832	49,966	2.519	Table 4-1 and Figure 4-26
Recirc Outlet	113.20	7.00	16,671	42,501	2.549	Table 4-1 and Figure 4-28
				Avg:	2.572	

$R_m = R + t/2$
P= 1000 psi (Page 4-3 of BWRVIP-108)

QUESTION:

b. Use of material properties at 325°F to bound the 100°F condition.

RESPONSE:

Refer to the discussion in Section 3.0 of Enclosure 3. Use of 325°F material properties were used to bound the 100°F condition where the peak stresses occurred during the evaluated transient. This is because the product of Young's modulus and the coefficient of thermal expansion (Eα), which is the most influential parameter for thermal stress analysis, is larger for greater temperatures. In addition, the instantaneous coefficient of thermal expansion was used in the stress evaluation instead of the mean coefficient of thermal expansion for added conservatism.