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From: "Hamer, Mike" <mhamer@entergy.com>
To: "Jonathan Rowley" <JGR@nrc.gov>
Date: Thu, Nov 9, 2006 5:03 PM
Subject: VYNPS License Renewal Application Am. 20

Jonathan,

Attached is VYNPS Letter BVY 06-098 to provide responses for RAIs pertaining to Rx Vessel Neutron Embrittlement and TLAA in BWRVIP Documents, received on 10/10/06. Please contact me if you have any questions.

<<BVY 06-098 VYNPS RAI Responses - Rx Vessel Neutron Embrittlement and TLAA in BWRVIP Documents.pdf>>

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November 9, 2006
Docket No. 50-271
BVY 06-098
TAC No. MC 9668

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

Reference: 1. Letter, Entergy to USNRC, "Vermont Yankee Nuclear Power Station, License No. DPR-28, License Renewal Application," BVY 06-009, dated January 25, 2006.

2. Letter, USNRC to VYNPS, "Requests for Additional Information for the Review of Vermont Yankee Nuclear Power Station License Renewal Application", NVY 06-147, dated October 10, 2006.

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
License Renewal Application, Amendment 20

On January 25, 2006, Entergy Nuclear Operations, Inc. and Entergy Nuclear Vermont Yankee, LLC (Entergy) submitted the License Renewal Application (LRA) for the Vermont Yankee Nuclear Power Station (VYNPS) as indicated by Reference 1. Attachment 1 provides responses to the NRC requests for additional information provided in Reference 2.

This submittal does not contain additional new regulatory commitments. Should you have any questions concerning this letter, please contact Mr. Dave Mannai at (802) 451-3304.

I declare under penalty of perjury that the foregoing is true and correct, executed on November 9, 2006.

Sincerely,

for

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment 1
cc: See next page

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Attachment 1

Vermont Yankee Nuclear Power Station

License Renewal Application Supplement

Amendment 20

RAI Responses for Sections 4.2 and 4.7

RAI 4.2.2-1
RAI 4.2.2-2
RAI 4.2.2-3
RAI 4.2.3-1
RAI 4.2.4-1
RAI 4.2.4-2
RAI 4.2.5-1
RAI 4.2.5-2
RAI 4.2.5-3
RAI 4.2.6-1
RAI 4.7.2.1-1
RAI 4.7.2.2-1

RAI 4.2.2-1

Section 4.2.2 of the VYNPS License Renewal Application (LRA), "Pressure/Temperature Limits," discusses the bases for the current pressure-temperature (P-T) limit curves (neutron fluence and adjusted reference temperature (ART) values at the 1/4T and 3/4T locations), which are valid to 32 effective full power years (EFPY), and the projected neutron fluence and ART values for the extended period of operation (54 EFPY). The P-T limit curve bases at 32 EFPY and 54 EFPY are summarized in LRA Table 4.2-1. It is unclear to the staff why the projected neutron fluence and ART values for 54 EFPY are substantially less than the corresponding values for 32 EFPY.

Sections 4.2.2 of the VYNPS LRA states that "conservative values" were used for determining the 32 EFPY P-T limits. Please discuss the conservative assumptions that resulted in the 32 EFPY neutron fluence and ART values, and, based on these assumptions, explain why the corresponding 54 EFPY values are less.

RAI 4.2.2-1 Response

VYNPS had P-T curves prepared based on a 1/4T fluence of 1.24 $\times 10^{18}$ n/cm^2 (E>1MeV). When GE (Ref. 1) projected the 32 EFPY vessel surface fluence to be 2.99 $\times 10^{17}$ n/cm^2 (E>1MeV) [the corresponding 1/4T fluence is only 2.2 $\times 10^{17}$ n/cm^2], VYNPS opted to use the previously calculated curves and avoid the time and expense of generating new curves based on a lower fluence. The conservative existing curves provided acceptable operating margin.

> Ref 1 VYNPS letter BVY 03-66 (J. K. Thayer to NRC, Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements, Additional Information – Neutron Flux Evaluation, 24July2003), including attached report GE-NE-0000-007-2342-R1-NP, Entergy Northeast Vermont Yankee Neutron Flux Evaluation, Revision 1, July 2003.

The 54 EFPY fluence value presented in Section 4.2.1 of the LRA [3.98 $\times 10^{17}$ n/cm^2 (E>1MeV)] was obtained by extrapolating the 51.6 EFPY value calculated in GE-NE-0014-0292-01. The fluence for 54 EFPY is higher than the fluence projected for 32 EFPY, but remains below the value on which the P-T curves were based.

RAI 4.2.2-2

Please discuss whether the 32 EFPY and 54 EFPY P-T limit curve bases summarized in LRA Table 4.2-1 take into consideration the VYNPS extended power uprate (EPU) conditions.

RAI 4.2.2-2 Response

In LRA Table 4.2-1, the highest projected 54 EFPY fluence of 5.39$\times 10^{17}$ n/cm^2 (E>1MeV) includes consideration of the extended power uprate. The highest value in the table for 32 EFPY is 1.24$\times 10^{18}$ n/cm^2 (E>1MeV). As discussed in the response to RAI 4.2.2-1, this 32 EFPY value resulted from an earlier calculation which yielded a significantly higher fluence value than more recent calculations. The 32 EFPY value in Table 4.2-1 is greater than the expected fluence at 54 EFPY including consideration of the extended power uprate Therefore, the fluence values in Table 4.2-1 for both the 32 EFPY and the 54 EFPY P-T limit curve bases bound the expected fluence including consideration of extended power uprate.

RAI 4.2.2-3

The P-T limit curves for the extended period of operation do not need to be submitted as part of the applicant's LRA for this time-limited aging analysis (TLAA). In accordance with 10 CFR Part 50, Appendix G, the applicant will need to submit new P-T limit curves for staff review and approval prior to the expiration of the unit's P-T limit curves for 32 EFPY. Section 4.2.2 of VYNPS LRA states that the P-T limit curve bases for 54 EFPY are bounded by the bases for the current (32 EFPY) P-T limit curves, and, as such, the TLAA for the P-T limits remains valid in accordance with 10 CFR 54.21(c)(1)(i). Please state when VYNPS intends to submit P-T limit curves for NRC approval for the extended licensed period of operation (54 EFPY).

RAI 4.2.2-3 Response

VYNPS will submit a Technical Specification amendment requesting extension of the P-T curves prior to the expiration of the approved P-T curves currently in Technical Specifications.

RAI 4.2.3-1

Table 4.2-3 of the LRA shows a value of "4.49E+16" for both the surveillance plate fluence and the surveillance weld fluence. This value is stated as being in exponential units of 10^{19} n/cm2 for both surveillance materials. Please state the correct fluence values for these surveillance materials, consistent with appropriate exponential units.

RAI 4.2.3-1 Response

The fluence value received by the surveillance specimens in Table 4.2-3 is 4.49×10^{16} n/cm^2 (E>1MeV), which is equivalent to 0.00449×10^{19} n/cm^2 (E>1MeV). The appropriate values for the table are shown as follows (bold information added, strike-outs deleted).

Table 4.2-3
Equivalent Margins Analysis

VYNPS Plate Material USE at 54 EFPY

Surveillance Plate % Cu	0.11%
Surveillance Plate Fluence (10^{19} n/cm^2)	**0.00449**~~4.49E+16~~
Surveillance Plate Measured Decrease	8.03%
RG 1.99 Predicted Decrease	5.55%
Ratio of Measured to Predicted	1.448
Beltline Plate % Cu	0.14%
1/4 T fluence (10^{19} n/cm^2)	0.0398
RG 1.99 Predicted Decrease	10.7%
Adjusted % Decrease	15.5%
Limiting % Decrease	23.5%
Plate Acceptable	Yes

The above decrease is less than the 23.5% decrease in the bounding equivalent margin analysis, so the analysis conclusions apply to the vessel plates.

VYNPS Weld Material USE at 54 EFPY

Surveillance Weld % Cu	0.03%
Surveillance Weld Fluence (10^{19} n/cm^2)	**0.00449**~~4.49E+16~~
Surveillance Weld Measured Decrease	4.80%
RG 1.99 Predicted Decrease	4.77%
Ratio of Measured to Predicted	1.005
Beltline Weld % Cu	0.10%
1/4 T fluence (10^{19} n/cm^2)	0.0398
RG 1.99 Predicted Decrease	11.19%
Adjusted % Decrease	11.24%
Limiting % Decrease	39.0%
Weld Acceptable	Yes

The above decrease is less than the 39% decrease in the bounding equivalent margins analysis, so the analysis conclusions apply to the vessel welds.

RAI 4.2.4-1

Please indicate why Table 4.2-4 of the VYNPS LRA lists the same 1/4T neutron fluence value (0.0398 x 1019 n/cm2) for all VYNPS reactor vessel (RV) beltline materials.

RAI 4.2.4-1 Response

VYNPS conservatively used the peak vessel fluence for all vessel plates and welds. This is consistent with the data currently in RVID2 as it also uses the same fluence for all locations. This approach is simpler than calculating the peak fluence for each plate and weld individually. While this is a conservative approach, the VYNPS vessel fluence is among the lowest in the BWR fleet and even this conservative approach yields adequate margin.

RAI 4.2.4-2

Table 4.2-4 of the VYNPS LRA lists initial reference nil ductility transition temperature (RT_{NDT}) values that are less conservative than those values currently established in the NRC staff's Reactor Vessel Integrity Database (RVID). Footnote 1 to Table 4.2-4 states that "the initial RT_{NDT} values supersede RVID2, as agreed to by the NRC in their SER [safety evaluation report] (Reference 4.2-9)." Reference 4.2-9 points to the NRC staff's SER on Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05, dated July 28, 1998. Section 4.2.4 of the VYNPS LRA states that initial RT_{NDT} values and standard deviations were taken from VYNPS NEDC-33090P, "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate," Table 3-2a. This document was submitted in support of Technical Specification Proposed Change No. 263 for the VYNPS EPU, dated September 10, 2003. Please explain this discrepancy.

It is unclear to the staff whether the initial RT_{NDT} values listed in Table 4.2.4 are currently authorized for use in determining the ART values for the RV beltline materials. Please provide information that points to where the NRC staff authorized the use of the specific initial RT_{NDT} values listed in Table 4.2-4 for determining the ART values for the VYNPS RV beltline materials.

RAI 4.2.4-2 Response

The referral to Reference 4.2-9 under Table 4.2-4 of the LRA is an error. The correct Reference is given as Reference 1 below.

As noted in the text of the LRA, these values of $RT_{NDT}(U)$ come from table 3-2a of NEDC-33090P, submitted with a proposed Technical Specification change on 10 September 2003 (Ref. 2) in support of the extended power uprate. The NRC SER for the extended power uprate had not been issued at the time the license renewal application was submitted. It was issued on March 2, 2006 (Ref. 1).

The SER for power uprate did not specifically state that the new RT_{NDT} (U) values could be used in place of the values then in RVID2; however, the NRC approved the Technical Specification change, implicitly approving the new RT_{NDT} values. The SER states "The staff confirmed, using the methodology of RG 1.99, Revision 2, that the limiting beltline material was the reactor vessel plate 1-14 with an ART of 58 °F at the 1/4T location . . ." NEDC-33090P calculated 57.22 °F for this location based on the revised $RT_{NDT}(U)$. The NRC accepted the new RT_{NDT} (U) values to do their confirmatory calculation.

NOTE: There is a typographical error in LRA Table 4.2-4. The initial RT_{NDT} for plate 1-15 should be -10 °F instead of +10 °F. The rest of the table is calculated using -10 °F and all other values are correct.

References:

1 NRC letter, R. B. Ennis to M. Kansler, Vermont Yankee Nuclear Power Station – Issuance of Amendment RE: Extended Power Uprate (TAC No. MC0761), 02 March 2006

2 VYNPS letter BVY 03-80, J.K Thayer to USNRC, Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate, 10 September 2003

RAI 4.2.5-1

BWRVIP-74, Section A.4.5, "Circumferential Weld Inspection Relief," states that in order to obtain relief from circumferential weld examination requirements, each licensee must submit a plant-specific relief request. In that submittal, licensees must demonstrate that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the staff's SER on BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," dated July 28, 1998, and (2) a licensee has implemented operator training and established procedures that limit the frequency of cold overpressure events to the frequency specified in the staff's July 28, 1998 SER. The LRA addressed condition 1 for this TLAA. However, the LRA did not address condition 2. Please address condition 2 as it relates to the proposed extended period of operation.

RAI 4.2.5-1 Response

The following discussion of reactor operator training was provided in the request for relief dated September 25, 2003. This training remains in effect and will continue throughout the period of extended operation.

"Licensed Operator Training provides another method to control reactor water level, temperature, and pressure, in addition to the design and procedural barriers discussed above. Simulator training for start-up and shut down scenarios provides an opportunity to perform the operator actions to maintain reactor pressure-temperature limits.

Procedural controls for reactor temperature, water level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling RPV water level within specified limits, as well as responding to abnormal RPV water level conditions outside the established limits. Additionally, control room operators receive training on compliance with the Technical Specification pressure-temperature limits curves and the basis for these limitations and curves. Plant-specific procedures have been developed to provide guidance to the operators regarding compliance with the Technical Specification requirements on pressure-temperature limits."

RAI 4.2.5-2

In accordance with 10 CFR 50.55a, the applicant would have to reapply for relief from American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code) RV circumferential shell weld examination requirements upon, or prior to, entering the first inspection interval in the extended period of operation. Please state if and when VYNPS intends to apply for relief from the ASME Code RV circumferential weld examination requirements for the extended licensed period of operation.

RAI 4.2.5-2 Response

VYNPS does plan to request relief from the RV circumferential shell weld examination requirements for the period of extended operation. In accordance with 10 CFR 50.55a, VYNPS must apply for relief prior to the end of each 10-year ISI interval.

RAI 4.2.5-3

In the July 28, 1998 SER on BWRVIP-05, the NRC staff concluded that examination of the RV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the RV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds.

RAI 4.2.5-3 Response

Reactor vessel plate 1-15 had one indication in the 1996 inspection. The indication is located in the plate below weld EF which joins plates 1-12 and 1-15. The sub-surface indication is outside the core region and is circumferential in orientation. A flaw evaluation was performed using the methodology in Appendix "A" of Section XI of the ASME Code and met the criteria in IWB-3600. The acceptability of the indication for continued service was approved by NRC in their letter of 11 October 1996.

> NRC Letter dated 11 October 1996, C. C. Harbuck to D. A. Reid, "Evaluation of Flaw Indication found during reactor pressure vessel inspections at Vermont Yankee Nuclear Power Station (TAC No. M96670)

The flaw was re-inspected in 2004 and no flaw growth was observed. VYNPS plans to re-inspect this weld each 10 year interval, thereby verifying there is no crack growth.

RAI 4.2.6-1

Section 4.2.6 of the VYNPS LRA states that the mean RT_{NDT} value for the limiting RV axial shell weld at the end of the extended period of operation (54 EFPY) is significantly less than the NRC limiting plant-specific mean RT_{NDT} value established in Table 1 of the staff's SER on BWRVIP-74, and therefore, the VYNPS axial weld failure probability is well below the acceptable limit of 5 x 10^{-6} per reactor-year. However, the limiting axial weld failure probability calculated by the NRC is based on the assumption that "essentially 100 percent" (i.e., greater than 90 percent) examination coverage of all RV axial welds can be achieved in accordance with ASME Code, Section XI requirements.

State whether the inservice inspection (ISI) examinations achieve "essentially 100 percent" (i.e., greater than 90 percent) overall examination coverage for the RV axial welds for the duration of the current licensed operating period. If they do not, reference the NRC staff's SER granting relief for limited scope axial weld examination coverage. If less than 90 percent overall examination coverage is achieved for the RV axial welds, revise this TLAA to account for the effects of the limited scope examination coverage.

RAI 4.2.6-1 Response

Various obstructions within the reactor vessel at VYNPS limit the inspection of the reactor vessel welds. The augmented reactor vessel examination performed by VYNPS in 1996 (third ISI interval) inspected approximately 65% of all the axial welds in the vessel (63% of all welds in the vessel). The NRC accepted 65% as an adequate inspection (Ref. 1) and stated "...based on the cumulative volumetric examination coverage obtained, the staff concluded that any significant patterns of degradation, if present, would have been detected and that the examinations performed provide an acceptable level of quality and safety . . . " (Ref. 1)

Examination of the welds in Refuel Outage 24 (2004 – 4th ISI interval) resulted in significantly increased coverage of the inspected welds due to improvements in inspection equipment and methodology. The results of RFO-24 inspections are presented below. Note that for the axial welds inspected, 90.1% of the length of the welds was scanned and an estimated 88% of the volume of those welds was inspected compared to only 65% inspected in 1996. However; the installation of shroud repair tie rods between the 1996 and 2004 inspections blocked the F1 and F2 axial welds in 2004. Because of this interference, welds F1 and F2 were not scheduled and are not considered in the percentage of coverage achieved. When all axial welds, including F1 and F2 are considered, VYNPS inspected approximately 68% of the axial welds.

Axial Weld	1996 % overall coverage	Total Length	2004 Length Scanned	2004 % Coverage (Volume)	2004 % Coverage (Length)
D1	99%	131.60	131.60	94.51%	100.00%
D2	96%	131.60	131.60	94.51%	100.00%
E1	51%	131.60	104.16	78.58%	79.15%
E2	53%	131.60	104.04	78.75%	79.06%
F1	65%	131.6	0	0	0
F2	65%	131.6	0	0	0
G1	45%	131.60	120.00	90.38%	91.19%
G2	45%	131.60	120.00	88.31%	91.19%
Summary	65%	789.60	711.40	88%	67.57%

VYNPS started the fourth Inservice Inspection 10-year interval in the fall of 2003. VYNPS has no 4th interval relief request for non-inspection of the vessel circumferential welds. In accordance with the requirements of ASME Section XI, VYNPS must inspect the circumferential welds or submit a relief request prior to the end of the 4th interval. Appropriate technical justification must accompany any such relief request.

References:

1 NRC letter (SER), W. M. Dean to G. A. Maret, Augmented Examination of the Reactor Pressure Vessel Shell Welds at Vermont Yankee Nuclear Power Station (TAC No. M99389), 18 February 1999

RAI 4.7.2.1-1

Section 4.7.2.1 of the VYNPS LRA is titled "BWRVIP-05, Reactor Vessel Axial Welds." However, this section addresses the elimination of RV circumferential welds from examination for the period of extended operation, as discussed in Section 4.2.5 of the VYNPS LRA. Please address whether there is any additional TLAA for the RV axial welds, other than what is addressed in Section 4.2.6.

RAI 4.7.2.1-1 Response

The only TLAA associated with the axial welds is the RT_{NDT} projections that show the bases for the circumferential weld inspection relief will still be met at 54 EFPY. This is the TLAA addressed in Section 4.2.6.

RAI 4.7.2.2-1

Section 4.7.2.2 of the VYNPS LRA addresses the recommendations of BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," pertaining to the TLAA for the RV core plate hold-down bolts. The relevant degradation mechanisms for this TLAA include the loss of preload and cracking of the core plate rim hold-down bolts. Section 4.7.2.2 of the VYNPS LRA indicated that BWRVIP-25 calculated the loss of preload for these bolts for the original 40-year licensed operating period and that Appendix B to BWRVIP-25, "BWR Core Plate Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," projected this calculation to 60 years, demonstrating that the VYNPS core plate rim hold-down bolts would experience only a 5 to 19 percent loss of preload for the extended period of operation.

In order for the staff to understand the data and analyses that were used to determine that the loss of preload due to stress relaxation at the end of the period of extended operation would be less than 20 percent, the staff requests that the applicant provide additional information demonstrating that BWRVIP-25 and BWRVIP-25, Appendix B are applicable to VYNPS, based on the following:

a. configuration and geometry of the VYNPS core plate rim hold-down bolts,
b. the temperature of the core plate rim hold-down bolts during normal operation, taking into consideration EPU conditions, and

c. projected bolt neutron fluence at the end of the period of extended operation, taking into consideration EPU conditions.

Please include the actual values for bolt temperature and projected bolt neutron fluence in the above discussion, and explain how it was determined that the effects of temperature and neutron fluence at the end of the period of extended operation would result in less than a 20 percent loss of bolt preload. Provide a detailed description of the methodology and data used at VYNPS to perform the above analyses, and include the basis for the stress relaxation curves.

The staff requests the applicant demonstrate that the axial and bending stresses for the hold-down bolts with the mean and highest loading will not exceed the allowable stresses for primary membrane and primary membrane plus bending of ASME Code, Section III, as a result of a 20 percent reduction in the specified bolt pre-load. Clearly state the assumptions on which this analysis is based, taking into consideration the fact that the approach recommended in Appendix A of BWRVIP-25 is based on an elastic finite element analysis of the core plate and hold-down bolts.

<u>RAI 4.7.2.2-1 Response</u>

1) The following additional information listed below is provided to demonstrate that BWRVIP-25 and BWRVIP-25, Appendix B are applicable to VYNPS;

   a. The configuration and geometry of the VY core plate and core plate rim hold-down bolts was reviewed and confirmed during the preparation of BWRVIP-25. As stated in Section 2.1 of BWRVIP-25, the result of <20% loss of preload is applicable to all but two BWRs (Millstone and Pilgrim) that have different core plate configurations,

   b. The feedwater temperature nominally increased about 15 degrees for the power uprate; however, the recirculation steam flow in the vessel decreased as a percentage of total flow resulting in less "hot" steam mixing with the "cold" feedwater. The net result was that core inlet enthalpy actually went down by 2 points. Since this results in a conservative effect on the core plate rim-hold down bolts loss of preload, the existing analysis remains limiting.

   c. VYNPS did not calculate the neutron fluence at the core plate hold down bolts for the period of extended operation. VYNPS did calculate the peak 1/4T fluence for the vessel, at 54 EFPY after the power uprate, as 3.98×10^{17} n/cm4 (e>1MeV) (LRA Section 4.1).

      Section B.4 of BWRVIP-25, Appendix B re-iterates that the loss of preload calculation is applicable to all but 2 BWRs, and is therefore applicable to VYNPS. Review of all BWR vessel fluences in Table 2-5 of BWRVIP-78 shows that prior to the power uprate VYNPS had the lowest end-of-license vessel fluence of any BWR (34th out of 34). After the power uprate and license renewal, VYNPS end-of-license fluence will still be among the lowest of the 34 BWRs, and still over an order of magnitude below the highest fluence. The fluence used by BWRVIP-25 that bounds all the BWRs, also bounds VYNPS.

2) The on-site audit of the license renewal application TLAAs requested the site-specific calculation for core plate hold-down bolts discussed in Appendix A of BWRVIP-25. VYNPS responded that there was no site-specific calculation (VY LR database question 108). VYNPS has addressed this issue via VYNPS license renewal commitment #29: "VYNPS will either install core plate wedges or complete a plant-specific analysis to determine acceptance criteria for continued inspection of core plate hold down bolting in accordance

with BWRVIP-25." If the calculation is performed, it will demonstrate that the axial and bending stresses for the hold-down bolts with the mean and highest loading will not exceed the ASME Code, Section III allowable stresses for primary membrane and primary membrane plus bending, as a result of a 20 percent reduction in the specified bolt pre-load.