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March 15, 2012

MEMORANDUM TO: George A. Wilson, Jr. Chief
Plant Licensing Branch LPL 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Martin C. Murphy, Chief ***/RA/***
Mechanical and Civil Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: SAFETY EVALUATION REGARDING VERMONT YANKEE NUCLEAR POWER STATION CORE PLATE HOLD DOWN BOLT INSPECTION PLAN AND ANALYSIS (TAC ME6248)

By letter dated March 18, 2011, Entergy Nuclear Operations (Entergy) submitted a plant-specific analysis report of the core plate hold down bolts (ML110840068). In Amendment 11 of the license renewal application (LRA), Entergy committed to either install core plate wedges or complete a plant-specific analysis to determine the acceptance criteria for continued inspection of the core plate hold down bolts in accordance with boiling water reactor (BWR) Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) and submit the inspection plan and analysis to the NRC two years prior to the period of extended operation (PEO). By letter dated December 30, 2011, Entergy updated the commitment to indicate the inspection plan and analysis would be provided one year prior to the PEO.

The Mechanical and Civil Engineering Branch, and the Vessels and Internals Integrity Branch completed the review of the applicable portions of the subject request related to the core plate hold down bolt inspection plan and analysis (ML110840069), and the responses to requests for additional information (ML120100126, ML11353A407, ML12037A066, and ML120100126) The safety evaluation input for the core plate hold down bolt inspection plan and analysis is provided as stated in the enclosure.

Docket Nos.: 50-271

Enclosure:
As stated

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<u>DISTRIBUTION</u>:
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SAFETY EVALUATION INPUT BY THE EMCB & EVIB
VERMONT YANKEE NUCLEAR POWER STATION (VYNPS)
CORE PLATE HOLD DOWN BOLT INSPECTION PLAN AND ANALYSIS
ENTERGY NUCLEAR OPERATIONS, INC. (ENTERGY)
DOCKET NO. 50-271
TAC NO. ME6248

1.0 INTRODUCTION

1.1 Application

By letter dated March 18, 2011, Entergy Nuclear Operations (Entergy) submitted a plant-specific analysis report of the core plate hold down bolts (ML110840068) (Ref. 5). Vermont Yankee is a BWR type 4 with Mark I containment design. In Amendment 11 of the license renewal application (LRA), Entergy committed to either install core plate wedges or complete a plant-specific analysis to determine the acceptance criteria for continued inspection of the core plate hold down bolts in accordance with Boiling Water Reactor (BWR) Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) (Ref. 1, 2, 3, and 4) and submit the inspection plan and analysis to the NRC two years prior to the period of extended operation (PEO). By letter dated December 30, 2011, Entergy updated the commitment to indicate the inspection plan and analysis would be provided one year prior to the PEO.

The Mechanical and Civil Engineering Branch (EMCB), and the Vessels and Internals Integrity Branch (EVIB) completed the review of the applicable portions of the subject request related to the core plate hold down bolt inspection plan and analysis (ML110840068, and ML110840069) (Ref. 5 and 6), and the responses to requests for additional information (ML120100126, ML11353A407, ML12037A066, and ML120100126) This safety evaluation input is based on review of the core plate hold down bolt inspection plan and analysis (Ref. 5) submittal by Entergy, Vermont Yankee core plate hold down bolt stress analysis report (Ref. 6) prepared by GE Hitachi (GEH) Nuclear Energy document NEDC-33618P, Rev. 0, and the responses to the Requests for additional information (RAIs) (Ref. 8, 12, and 14).

1.2 Core Plate Assembly

The core plate assembly, located inside the BWR reactor pressure vessel, consists of a perforated stainless steel plate reinforced by stiffener beams and supported on the perimeter by a circular rim. Stiffener beams are welded to the core plate to carry the pressure loads from design basis loss of coolant accident (LOCA) events. The pressure loading from LOCA causes compressive stresses in the lower edges of the stiffener beams. Cross ties or stabilizer beams are added between the stiffener beams to prevent flange buckling by providing lateral support.

ENCLOSURE

The core plate rim is bolted to a ledge on the core shroud by stainless steel studs which prevent vertical movement. The rim hold down bolts attach the core plate to the core shroud. The stabilizer beams or rods also provide support for in-core housing monitors. Core plate assembly provides lateral support for the fuel bundles, control rod guide tubes, and in-core instrumentation during seismic events and provides vertical support for the peripheral fuel assemblies. The core plate is positioned on the shroud ledge by four vertical aligner pins. The seismic and other dynamic loads are shared between the friction load of the shroud to rim bolt connection, and the shear resistance of the aligner pins. During seismic events the core plate provides lateral support for the core to prevent misalignment that could affect the insertion of the control rods. For plants such as VYNPS that do not have wedges and studs between core plate rim and the shroud, the core plate may shift more than 0.75 inch if sufficient hold down bolt failures are assumed, According to BWRVIP-25 (Ref. 1), control rod insertion testing has demonstrated that a core plate horizontal misalignment of 0.75 inch would not significantly increase the scram time, and a displacement of 1.0 inch would inhibit insertion. The critical number of intact hold down bolts required to prevent lateral displacement during a seismic event is plant unique, and can be determined from a plant specific analysis. Even if hold down bolt failures resulted in significant core plate movement preventing the insertion of control rods, the plant could still be brought to a safe shutdown condition using the standby liquid control (SLC) system. Core plates experience tensile stresses and have stress concentrations due to threaded regions. GEH has also determined that core plate bolt stress relaxation occurs due to thermal and irradiation effects.

2.0 REGULATORY EVALUATION

Title 10 Part 54 of the Code of Federal Regulations 10 CFR 54.21(a)(3) requires that for each component within the scope of license renewal as defined in 10 CFR 54.4 and subject to aging management review according to the criteria of 10 CFR 54.21(a)(1)(typically described as long-lived, passive components), applicants for license renewal must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

10CFR54.21(c)(1) requires an evaluation of time-limited aging analyses (TLAAs), as defined in 10 CFR 54.3, which states that [TLAAs], for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

(6) Are contained or incorporated by reference in the CLB.

10 CFR 54.21(1)(c) requires for each TLAA that the applicant shall demonstrate that—

- (i) The analyses remain valid for the period of extended operation;
- (ii) The analyses have been projected to the end of the period of extended operation; or
- (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The initial version of "BWR [Boiling Water Reactor] Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) (Reference 1) was approved by the NRC staff for providing acceptable guidance for the inspection and evaluation of core plate components (including the core plate rim hold-down bolts also referred to as the core plate hold-down bolts, or simply core plate bolts) for the current operating period (plants in their initial 40 years of operation) by letter dated December 19, 1999 (Reference 2). By letter dated July 17, 1997 (Reference 3), the BWRVIP submitted "Appendix B, BWR Core Plate Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)." The NRC staff transmitted its safety evaluation for referencing BWRVIP-25 in license renewal applications, as modified by Reference 3, via letter dated December 7, 2001 (Reference 4). Reference 4 concluded that BWRVIP-25 provided an acceptable basis for managing aging of the core plate bolt components, provided that applicants for license renewal meet the limitations and conditions and the plant-specific action items of the enclosed SE. Plant-specific Applicant Action Items 4 and 5 are most relevant. Applicant Action Item 4 of the SE (Reference 7) stated that due to the susceptibility of the rim hold-down bolts to stress relaxation, applicants referencing the BWRVIP-25 report for license renewal should identify and evaluate the projected stress relaxation as a potential TLAA issue. Applicant Action Item 5 stated, that until such time as an expanded technical basis for not inspecting the rim hold-down bolts is approved by the staff, applicants referencing the BWRVIP-25 report for license renewal should continue to perform inspections of the rim hold-down bolts.
Since VYPNS did not have a plant-specific stress relaxation TLAA analysis for the core plate bolts, Entergy provided Commitment No. 29 in Amendment 11 to the VYNPS License Renewal Application to either install core plate wedges or complete a plant-specific analysis to determine acceptance for continued inspection of core plate bolts in accordance with BWRVIP-25.

3.0 TECHNICAL EVALUATION

3.1 Licensee Evaluation

By letter dated March 18, 2011 (Reference 5), the licensee submitted its plant-specific analysis of the core plate bolts intended to fulfill the requirements of the commitment described above. The analysis report (Reference 6) was included as Attachment 1 to Reference 5. The licensee described the core bolt stress analysis, load cases, load combinations and results from the plant specific analysis. The licensee described the method of evaluation of stress relaxation of the core plate bolts in Section 5.0 of Reference 6. The licensee's evaluation is based on proprietary data generated by General Electric-Hitachi (GEH). Figure 5-1 of Reference 6 shows a mean design curve fit to the plotted data, designated the GEH design curve. The licensee also presented in Figure 5-2 of Reference 6 data from BWRVIP-99, "BWRVIP Vessel and Internals Project Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components," for Type 304/316/348 wedge loaded double cantilever beam specimens (DCBs) in a BWR environment. The data are for higher fluence levels (4-6 $x10^{20}$ n/cm^2) than those experienced by the core plate bolts. Figure 5-3 of the Reference 6 shows some additional test reactor data compared to the mean design curve determined using GEH data only. This figure shows the GEH design curve is conservative compared to the test reactor data.

The licensee provided the results of their evaluation of the potential for stress relaxation of the core plate bolts in Section 6.7 of Reference 6. The licensee provided the percentage of preload relaxation due to the peak neutron fluence predicted for the core plate bolts. The licensee indicated that the fluence was a conservative estimate based on a flux evaluation performed in support of the extended power uprate (EPU) for VYNPS in 2003.

3.2 Staff Evaluation

3.2.1 Loss of Preload of Core Plate Bolts (EVIB)

The staff used BWRVIP-25 as guidance for our review of the licensee's evaluation of stress relaxation of the core plate hold-down bolts. Appendix B to BWRVIP-25 provides an evaluation of the potential loss of preload in BWR core plate bolts that is intended to be bounding for all BWRs. Additionally, in the "Safety Evaluation Report (SER) related to the License Renewal of Vermont Yankee Nuclear Power Station," (NUREG-1907, Reference 7), the staff noted that [VYNPS] did not calculate a plant-specific value of the neutron fluence at the core plate bolts. However, in NUREG-1907, the staff concluded the core plate bolt fluence should remain bounded by the fluence used for BWRVIP-25, based on VYNPS maximum EOL RV neutron fluence being lower than that of most BWR's. However, because the staff has not previously approved a calculated or estimated plant-specific value for the core plate bolt neutron fluence, in RAI 1, the staff requested the applicant provide the details of the flux evaluation that was used to determine projected total fast neutron fluence for the core plate bolts for a 60-year plant life.

In its response to RAI 1 by letter dated December 9, 2011 (Reference 8), the licensee provided a discussion of the flux evaluation. The licensee indicated that the flux evaluation was based on a best-estimate flux evaluation performed in 2003 in support of an extended power uprate (EPU). Results from the EPU flux evaluation were used to estimate the flux and fluence for the core plate bolts at VYNPS. In the EPU flux evaluations, best estimate fast flux values were determined at the RV inside surface, core shroud inside surface, and surveillance capsule. To determine the flux at the bolt location, the licensee first determined the core midplane flux corresponding to the radial location of the bolt. The licensee then divided the bolt into twenty evenly spaced axial sections. A synthesized flux was determined for each section by multiplying the core midplane flux at the radius of the bolts ($3.09x10^{11}$ n/cm^2-s, E> 1 MeV) by the axial flux factor (defined as the ratio of the flux at a particular axial location to the core midplane flux), times a safety factor of 1.5 intended to account for uncertainties associated with flux calculation for regions beyond the core beltline. The licensee then averaged the synthesized fluxes for the 20 bolt sections to obtain the average flux for the bolt over the axial length of $7.09x10^9$ n/cm^2-s (E > 1MeV). For time periods prior to the implementation of the EPU in 2003, the licensee's analysis ratioed the flux based on the previous power levels in megawatts thermal (MWt) to the post-EPU flux. VYNPS operated at two different thermal power levels including the previous thermal power and a transitional power level for the cycle prior to full EPU implementation. The licensee thereby obtained peak and average fluxes corresponding to each power level at which VYNPS has operated.

To determine the EOL fluences for the core plate hold down bolts, the licensee then multiplied the EFPY for each power level by the flux for that power level (peak and average) to determine the peak and average fluences for the bolts. A peak 60-year fluence of $5.2x10^{19}$ n/cm^2 for the bolt was thus obtained. The staff checked the licensee's calculation and obtained the same result.

The staff finds the response to RAI 1 acceptable because it provides an adequate description of how the core plate hold-down bolt flux was extrapolated, and includes appropriate conservatisms to ensure the flux used to project the loss of preload is bounding. Specifically, 1) the peak azimuthal flux at the radius of the bolts was used as the starting point, 2) a factor of 1.5 was applied to the synthesized flux for each bolt section, and 3) peak bolt flux rather than the axial average was used as the basis for the loss of preload projection. Therefore, the staff finds RAI 1 is resolved.

The staff verified that the percentage reduction in preload assumed by the licensee matches the percentage reduction in preload that is indicated by the GEH design curve based on the predicted peak neutron fluence . The staff compared the licensee's prediction of the reduction in preload to other industry data for stress relaxation. Industry data relevant to BWRs can be found in BWRVIP-99-A, "BWR Vessel and Internals Project – Crack growth Rates in Irradiated Stainless Steels in BWR Internal Components" (Reference 9), and MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values" (Reference 10). BWRVIP-99-A provided two figures showing fraction of stress remaining for bent beams exposed at 60 and 300°C in the Chalk River Reactor, for pure nickel and Alloy X-750. BWRVIP-99-A also included the data for wedge-loaded dual cantilever beam (DCB) specimens for Type 304/316/348 that was shown in Figure 5-2 of Reference 6. This data was for higher fluence levels; the trend line extrapolated to fluence levels comparable to the core plate bolts indicates a much lower degree of relaxation (5% reduction or 95% remaining preload) than the applicant determined based on the GEH data. Even if an upper bound trend line were drawn on this figure, the reduction in preload would only be about 10% (90% preload remaining). MRP-175, Figure H-7, provides a lower bound curve for percentage of remaining stress versus displacements-per-atom (dpa) for various austenitic stainless steels and nickel-based alloys at various temperatures. It should be noted that displacements-per-atom (dpa) are a measure of irradiation damage to a material that does not exactly convert to fluence in neutrons per square centimeter (n/cm^2), but in light-water reactor neutron spectra, 1 dpa ≅ $6.7x10^{20}$ n/cm.[2] A conservative lower bound curve was used by the MRP since the intent of the curve is to screen for the potential of stress relaxation. At 0.1 displacements-per-atom (dpa), the lower bound curve is at 50% remaining stress. However, if only the data points for annealed type 304 stainless steel are considered, a more realistic lower bound is around 75% of remaining stress at 0.1 dpa. In addition, if a best estimate curve were fit to this data the remaining stress value would probably be between 85-90% which is consistent with the reduction in preload assumed in the licensee's analysis Based on the industry data, the staff finds that the licensee's estimate of remaining preload is reasonably consistent with both lower-bound and best-estimate values that would be determined from other industry data, which would range from about 75-95%.

Section 4.7.3 of NUREG-1907 (Reference 7) indicates that, as stated in Appendix B to BWRVIP-25, a 5-19% reduction in core plate hold-down bolt stress due to thermal and irradiation effects should be expected over the 40-year life of a plant. However, Appendix B to BWRVIP-25 does not provide the neutron fluence value on which the preload relaxation evaluation was based. For comparison to the predicted loss of preload (14%) used in the VYNPS analysis, in RAI 2 the staff requested the neutron fluence value on which the 5-19% loss of preload is based. In its response to RAI 2 contained in the letter dated December 9, 2011, the licensee stated that the GE evaluation of core plate relaxation determined that the BWRVIP-25 maximum reported stress relaxation value of 19% is valid to an average neutron fluence level of $8x10^{19}$ n/cm^2 or less, and that this fluence is an average fluence over the entire length of the core plate bolt, determined at the peak azimuthal flux location. The staff finds the response to RAI 2 is acceptable because it demonstrates the licensee's fluence value is bounded by the neutron fluence values analyzed in BWRVIP-25. Also, if ratio of the VYNPS peak neutron fluence to the maximum BWRVIP-25 neutron fluence is multiplied by the maximum stress relaxation from BWRVIP-25, a similar percentage of stress relaxation to that assumed by the licensee is obtained. Therefore, the staff finds the licensee's projected loss of preload as a function of neutron fluence is consistent with BWRVIP-25 and is therefore acceptable. RAI 2 is resolved.

The staff finds the licensee's evaluation of the projected loss of preload of the VYNPS core plate hold-down bolts due to irradiation-assisted stress relaxation is acceptable because 1) the licensee appropriately estimated the peak fluence for the bolts at EOL based on its EPU fluence evaluation; 2) the licensee's projection of loss of preload based on the peak bolt fluence is consistent with what would be expected based on the BWRVIP-25 generic analysis and other industry data.

However, cracking of the core plate hold-down bolts due to intergranular stress corrosion cracking (IGSCC) could also result in loss of load carrying capacity and did not appear to have been considered in the stress analysis of Reference 2. The staff requested additional information related to the possibility of cracked bolts due to IGSCC in RAI 3, discussed in detail in the next section, since this topic is related to the inspection plan for the core plate hold-down bolts.

3.2.2 Inspection Plan for Core Plate Hold-Down Bolts (EVIB)

Reference 5 indicates that the sample size of VYNPS core plate hold down bolts inspected has been changed from 50 % to 25 %. The frequency and method of the inspections will remain the same (visual VT-3 inspection from the top of the bolts every other refueling outage). This represents a deviation from the BWRVIP-25 requirements for ultrasonic inspection. This level of inspection would probably reveal if there was widespread failure of the bolts but could miss partially cracked bolts or a small number of failed bolts.

Therefore, in RAI 3, the staff requested the following information:

1. Given that VYNPS has reduced the sample size for VT-3 from that recommended by BWRVIP-25, justify that the sample size of core plate hold down bolts being inspected is adequate to ensure that there will be sufficient intact bolts to meet the load requirements of the plant-specific stress analysis.
2. Justify that performing the VT-3 inspection from above the core plate will provide a sufficient level of assurance that cracked or broken bolts will be detected, given that BWRVIP-25 recommends performing the VT-3 inspection from below the core plate.
3. Does the core plate stress analysis account for some portion of the core plate bolts being either completely or partially cracked due to intergranular stress corrosion cracking or irradiation assisted stress corrosion cracking? If so, describe how the cracking was accounted for.
4. If cracking was not accounted for in the stress analysis, provide a justification for cracking not being considered.

In its response by letter dated December 9, 2011 (Reference 8), the licensee indicated the following:

With respect to RAI 3 Item 1, VYNPS performed inspection of 50% of the core plate hold-down bolts for four successive outages with no noted degradation. The licensee cited section 3.2.2.2 of BWRVIP-25, which allows the re-inspection schedule for the core plate hold-down bolts to be adjusted based on good inspection results combined with good operating experience. Based on performance, the licensee adjusted the inspection frequency and sample size to 25% of the bolts every other outage beginning in 2007 and has performed these inspections since that time with no noted degradation. The staff notes that the inspections performed were VT-3 visual examinations performed from above the core plate rather than VT-1 visual examinations performed from below the core plate as prescribed by BWRVIP-25.

With respect to RAI 3 Item 2, VYNPS stated that it is currently industry practice only to perform VT-3 inspections from above the core plate, because performing VT-1 examination from below the core plate requires extensive disassembly and a UT technique has yet to be developed. The licensee also referenced its March 18, 2011 letter (Reference 11) documenting its deviation from the BWRVIP-25 inspection requirements. Reference 11 provides a summary of the licensee's justification for the deviation, which cites the following factors supporting the deviation:

- Low susceptibility to cracking and high flaw tolerance of the bolting,
- Postulated flaws would not grow to a size that significantly reduces the bolt preload over the life of the plant
- Redundancy of structural components that would prevent adverse displacement of the core plate even if significant cracking occurs in the bolts.

- Even if all the core plate hold-down bolts and the redundant hardware failed, preventing insertion of the control blades, the standby liquid control system could be used to bring the reactor to a safe shutdown.

In response to RAI 3 Item 4, the licensee stated that the core plate stress analysis did not account for some portion of the core plate [hold-down] bolts either completely or partially cracked due to IGSCC or irradiation assisted stress corrosion cracking (IASCC). In response to RAI 3 Item 3, the licensee provided its justification for not assuming that some portion of the core plate bolts were either completely or partially cracked due to IGSCC or IASCC. In its justification, the licensee cited Section 2.2.9 of BWRVIP-25, which notes that the core plate hold-down bolts are not sensitized, which reduces the IGSCC susceptibility, and that there have been no instances of IGSCC in the field of these bolts.

The staff agrees that the IASCC susceptibility of these bolts is low, because the peak fluence level of the bolts is below the range at which IASCC can typically begin to be a factor in BWRs ($5x10^{20}$ n/cm^2). However, although bolts are not sensitized, the staff was concerned they could potentially be cold worked which can increase the susceptibility to IGSCC.

The licensee did not account for the possibility of some cracked or broken bolts in their analysis. Since the licensee is inspecting only a sample of the bolts, and the inspection method used is visual VT-3 examination, which only allows the ends of the bolts and nuts to be examined, the staff had concerns that the current inspection plan is not capable of detecting cracked or broken bolts. Only the top end of the bolt and the nut can be viewed from above the core plate. The nut is fillet welded to the bolt to prevent loosening. To address these issues, the staff requested the following additional information:

1. Provide a justification that the VT-3 visual examinations would be effective at detecting failed core-plate hold-down bolts.

2. What percentage of core plate bolts for VYNPS must be intact to avoid exceeding the allowable stresses on the bolts as given by Table 8-1 of the analysis (Reference 6)?

3. Considering the effectiveness of the VT-3 examination at detecting cracked or broken bolts, does the percentage of the bolts being sampled support demonstration that the required number of bolts are intact, assuming no failed bolts are found in the sample? Provide a statistical argument or analysis similar to that provided in BWRVIP-25, Section 3.2.2.2.

4. If a statistical argument cannot be made, provide a more detailed basis supporting a very low probability of significant loss of load bearing capability due to IGSCC of the bolts, and/or revise the analysis to account for the possibility of some bolt failures due to SCC.

In response to the follow-up RAI 1 by letter dated February 1, 2012 (Reference 12), the licensee justified the effectiveness of the VT-3 visual examinations by citing a portion of General Electric Services Information Letter (SIL) No. 588R1. The information indicates that the core plate hold down bolts for older BWRs have low susceptibility to SCC because they were procured to a specification prohibiting cold forming operations after solution heat treatment, and have a low preload (10-15 ksi). Therefore, the SIL 588 R1 recommended inspection is to show the bolts have not loosened and rotated due to a combination of vibration and failure of the welds on the locking device, which should be obvious by visual VT-3 examination. The staff finds the licensee's response to follow-up RAI 1 acceptable because the information provided demonstrates the core plate hold-down bolts should have low IGSCC susceptibility.

In its response to follow-up RAI 1, the licensee also cited Section 3.2.5 of BWRVIP-47-A, "BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines," which states that

> "The BWRVIP has determined that removing or dismantling of internal components for the purpose of performing inspections is not warranted to assure safe operation. However, on occasion, utilities may have access to the lower plenum due to maintenance activities not part of normal refueling outage activities. In such cases, utilities will perform a visual inspection to the extent practical. Results of the inspection will be reported to the BWRVIP and will be forwarded by the BWRVIP to the NRC."

The licensee further stated that the VYNPS Reactor Vessel Internals (RVI) Program contains a provision for performing inspections when access to the lower plenum is available due to maintenance activities.

Although the specification of no cold forming and low preload for the bolts would not completely preclude IGSCC, these factors combined with operating experience for core plate bolts across the BWR fleet, which has noted no failures of these bolts, provides reasonable assurance that widespread IGSCC failure of these bolts is unlikely. Further, the staff agrees that the VT-3 examination should detect loosening of the bolts due to vibration combined with failure of the locking device welds. Finally, in accordance with BWRVIP-47-A, inspections of opportunity when access to the lower plenum is possible due to maintenance should provide additional assurance that core plate bolts are intact since it should be possible to view the threaded portion of the bolts from below the lower plenum region. Therefore, follow-up RAI 1 is resolved.

In response to follow-up RAI 2, the licensee indicated that the VYNPS core-plate stress analysis did not assume any of the bolts were initially failed or cracked, and that this is consistent with the methodology of BWRVIP-25, Appendix A. Therefore, the staff could not determine from the licensee's response if there is an acceptable number of bolts that could be failed that would not result in the allowable stresses being exceeded in one of the design-basis scenarios analyzed in the stress analysis.

In response to follow-up RAI 3, the licensee indicated that they had performed a statistical evaluation using ANSI-ASQ Standard Z1.4 Table 1. This table indicated a sample size of 13 for a nonconformance value of 1% - i.e., the finding of no failures in the sample of 13 bolts indicates that less than 1% of the bolts in the overall population of 30 bolts would be defective. Based on this statistical evaluation, the licensee determined that their previous sample size of 25% for the VT-3 examination is inadequate, and stated that they would increase the sample size to 50% or

15 bolts, beginning with RFO 31. The licensee also included this change in sample size as a commitment in Attachment 2 to the February 1, 2012 letter. The licensee stated that no response to follow-up RAI 4 is required because a statistical argument was made in response to Item 3.

The staff notes that the licensee's statistical evaluation is based on a standard used to determine the acceptance quality limit (AQL), which is defined as the quality level that is the worst tolerable process average when a continuing series of lots is submitted for acceptance sampling. This standard is typically used for quality assurance of manufactured products. The standard does not describe the statistical analysis behind the determination of the proportion of the population that is defective. Therefore, the staff performed an independent statistical evaluation of the probable number of cracked bolts in the overall population given that no cracked bolts are found in the 50% sample. The staff used a hypergeometric distribution, which can be used as the basis for a sampling scheme (a hypergeometric experiment) that samples a population for attributes without replacement and which satisfies the following conditions (Reference 13):

- The sampled population is finite;
- Once an item is selected, it cannot be selected again;
- The size of the population is known;
- The number of items with the attribute of interest is known;
- Each item in the sample is drawn at random.

The staff determined that if no cracked bolts are present in the 50% sample, the probability that the number of cracked bolts in the overall population would result in the ASME Code allowable stresses being exceeded, based on the margins given in Table 8-1 of Reference 6, is less than 5%.

The staff also notes there are several conservatisms in the VYNPS stress analysis that make it even less likely the ASME Code allowable stresses would be exceeded. First, as noted in the response to RAI 4 via letter dated January 5, 2012 (Reference 14), a conservative coefficient of friction was used in determining the reduction in the applied horizontal loading due to frictional resistance. Second, in Scenarios 1 and 3, no credit was taken for load being borne by the aligner pins.

Based on the staff's independent statistical evaluation, and considering the conservatisms in the VYNPS core plate hold-down bolt structural analysis, follow-up RAI's 2 and 3 are resolved because there is reasonable assurance that the number of bolts that could possibly be cracked, given the finding no cracked bolts in the proposed sample inspection, would not result in the allowable stresses being exceeded in the event of a design basis accident

Based on the information submitted by the licensee supporting low IGSCC susceptibility for the VYNPS core plate hold-down bolts, and the margins present in the VYNPS core plate bolt stress analysis as supported by the staff's statistical evaluation, the staff finds the licensee's proposal to visually inspect a 50% sample of the bolts every other refueling outage to be acceptable until the BWRVIP revises its guidance for core plate hold-down bolt inspection and evaluation.

3.2.3 Stress Analysis of Vermont Yankee Core Plate Hold-Down Bolts (EMCB)

The licensee performed stress calculations to demonstrate the structural adequacy of the VYNPS core plate bolts and aligner pins. The methodology and assumptions utilized are consistent with BWRVIP-25. The results of the stress evaluations for three different scenarios in accordance with BWRVIP-25 Appendix A are summarized. The three scenarios considered by VYNPS are as follows.

i) Loads on the core plate bolts taking no credit for the aligner pins. In this case, the bolts take all of the horizontal and vertical loads.
ii) Shear load on the aligner pins with no credit for horizontal restraint from bolts. In this case, the bolts take vertical loads and the aligner pins take all of the horizontal loads.
iii) Loads on the core plate bolts with no credit for aligner pins. This case also assumes the stiffener beam to rim weld cracked. In this case, the core plate bolts take all of the horizontal and vertical loads.

The staff's review of the three scenarios considered in VYNPS core plate bolts analysis indicates that the scenarios considered are acceptable because they are consistent with the the scenarios discussed in Appendix A of BWRVIP-25 topical report that was previously reviewed by the staff. These scenarios represent the most limiting conditions for the core plate bolts.

3.2.3.1 Loads

The stress evaluation of the core plate bolts included the effects of dead weight (DW), Fluid drag load due to reactor internal pressure difference (RIPD) across core plate for normal and faulted conditions, Seismic loads from operating basis earthquake and safe shutdown earthquake (OBE, and SSE), Fuel Lift load (FL), and bolt preload. DW of the core plate assembly is a vertical downward load. The seismic loads OBE & SSE are calculated based on Vermont Yankee seismic accelerations and act in both horizontal and vertical directions. The fluid drag load RIPD is an upward load on core plate bolts. The fuel lift load FL is an upward load considered for the faulted condition. Friction at the interface of core shroud ledge and core plate rim is also considered. Safety relief valve (SRV) actuation loads and torus induced loss of coolant (LOCA) accident loads are not significant for Vermont Yankee because the torus and drywell are not substantially coupled for Mark I type containment. The annulus pressurization (AP) load is not part of VYNPS design basis, and is not a significant. The acoustic load (AC) resulting from the initial transient phase from a double ended guillotine break of the recirculation suction line (RSL) is very abrupt relative to the shroud inertia and frequencies and therefore has insignificant effect on the shroud. The steady state portion of the load from RSL break affects the shroud and components external to the shroud. The core plate being inside of the shroud is essentially unaffected by the RSL break steady state load. The staff's review finds that the licensee appropriately considered the applicable loadings in the structural evaluation of the core plate bolts.

3.2.3.2 Load Combinations and Acceptance criteria

The VYNPS core plate bolt stress analysis utilized the criteria for allowables in accordance with the Updated Final Safety Analysis Report (UFSAR, appendix section C.2, Ref.15), and ASME Boiler and Pressure Vessel (ASME B&PV) Code, Section III (Ref. 16). The material properties for the core plate bolts and the aligner pins are based on type 304 austenitic stainless steel of Ref. 16. The staff notes and accepts that ASME B&PV Code is not mandatory for the design of the VYNPS reactor vessel internals due to the vintage of the plant. However, the licensee committed to meet the intent of the ASME B&PV Code as described in UFSAR (Ref. 15).

The staff's review determined that the licensee utilized for Normal & Upset, emergency, and faulted condition general membrane stress allowables of $1S_m$, $1.5S_m$, and $2S_m$ respectively, where S_m is the allowable stress intensity of the material. The licensee utilized for Normal & Upset, emergency, and faulted condition, membrane plus bending stress allowables of $1.5S_m$, $2.25S_m$, and $3S_m$ respectively The licensee utilized for Normal & Upset, emergency, and faulted condition, shear stress allowables of 0.6 S_m, $0.9S_m$, and $1.2S_m$ respectively. Based on the review of the licensee's stress evaluations for the core plate hold down bolts and aligner pin, the staff concludes that the acceptance criteria are in accordance with the ASME B&PV Code, and UFSAR commitment

3.2.3.3 Stress Evaluations

The Vermont Yankee core plate design contains 30 core plate bolts of 2 inch diameter each and four vertically oriented aligner pins of 2.625 inch diameter The finite element (FE) model used for the core plate assembly is not exactly VYNPS plant specific but is based on FE model in Appendix-A of BWRVIP-25. In response to an RAI for not having Vermont Yankee plant specific FE model, the licensee provided justification that the analysis is linear and the results are appropriately scaled to account for the plant specific items. The staff reviewed the VYNPS plant specific items provided in the licensee's response that the licensee considered for scaling the results. The stress evaluations for the VYNPS core plate bolts and aligner pins considered appropriately for scaling the BWRVIP-25 analysis results based on Vermont Yankee geometry items that include the number of bolts, size of core plate components, bolts and aligner pins, Vermont Yankee loadings, and Vermont Yankee specific bolt relaxation due to fluence and thermal effects.

In response to an RAI (Ref. 14) on the justification of friction in Vermont Yankee calculations, the licensee stated that ignoring friction is overly conservative. The staff reviewed and agrees with the licensee's justification that not considering friction at the interface of the core plate rim and shroud ledge because (i) the frictional resistance in a clamped connection of this type with a large clamping force has significant friction, and (ii) the licensee used a smaller frictional coefficient of 0.2 to be conservative compared with GEH tests (Ref. 6) that determined a frictional coefficient close to 0.5 for 304 (stainless steel (SS) sliding on 304 SS with deoxygenated water as a lubricant.

In its core plate bolt evaluations, the licensee appropriately accounted for bolt preload relaxation of 14 percent from neutron fluence due to 60 year plant life (see SE Section 3.2.1), and 6.2 percent relaxation from modulus of elasticity decrease due to temperature effect between 70° F and 550° F. The preload loss from fluence is based on conservative fluence that uses peak fluence at azimuthal location for all bolts, and the use of the highest axial fluence at the bottom of active fuel for all bolts. The preload on core plate bolts is accounted for by adding the membrane stress due to preload to the calculated membrane stress, which is consistent with BWRVIP-25 Appendix-A.

The licensee performed evaluations for core plate bolts for the Normal & Upset (DW+Normal RIPD+OBE), Emergency DW+Normal RIPD+SSE), and Faulted load combinations DW+Faulted RIPD+SSE+FL), and summarized the results for the bounding faulted load combinations for the three scenarios described above. The licensee considered the applicable loads and demonstrated that the membrane and membrane plus bending stresses in core plate bolts and the shear stresses in the aligner pins satisfy the corresponding allowable criteria in the ASME B&PV Code. The results show that the computed mean membrane stress is 12200 pounds per square inch (psi) compared to its allowable of 32000 psi, and computed mean membrane plus bending stress of 41700 psi compared to its allowable of 48000 psi for the faulted condition cases (i) when all the vertical and horizontal loads are taken by the core plate bolts with no credit for aligner pins, and (ii).when all the vertical and horizontal loads are taken by the core plate bolts with no credit for aligner pins, and the stiffener beam to rim weld cracked. The results also show that the shear stress in the aligner pin is 7700 psi compared to its allowable of 19200 psi for the faulted condition case when the aligner pins take all of the horizontal loads with no credit for horizontal restraint from bolts. The core plate stresses and aligner pin stresses are acceptable because they meet the respective allowables with some conservative assumptions regarding friction, and preload relaxation.

The staff requested the licensee to provide the cumulative usage factor (CUF) for the core plate bolts for 60 year plant life. In response to an RAI (Ref. 14), the licensee demonstrated based on a simplified analysis that the alternating stress for the core plate bolts is only 1150 psi from normal & upset loadings and is well below the endurance limit of 25000 psi. The normal and upset cycles are less than 10000 cycles and the number of cycles for endurance limit is over one million. Thus, the CUF is negligible. Based on a review of this information, the conclusion that the CUF is negligible for the core plate bolts, is acceptable to the staff.

4.0 CONCLUSIONS

With respect to the effects of neutron irradiation on the core plate bolt properties, specifically the loss of preload determined by the licensee, the staff found the licensee's evaluation to be acceptable.

With respect to the inspection plan propose by the licensee for the core plate bolts, the staff finds the inspection plan as modified by the commitment contained in Attachment 2 to the licensee's February 1, 2012 letter, to be acceptable. Specifically, the licensee committed to inspect of 50% of the VYNPS core plate hold down bolts every other refueling outage, commencing with RFO 31, using the VT-3 [visual examination] method in accordance with the VYNPS Reactor Vessel Internals Inspection Program until BWRVIP-25 is revised. The licensee further committed to implement the revised BWRVIP-25 guidance for the core plate bolts.

With respect to the stress analysis of the core plate bolt, including the preload relaxation due to thermal effects and fluence for a 60 year life, the staff finds the licensee's evaluation acceptable because the core plate bolts satisfy the ASME B&PV Code criteria for the applicable loads and load combinations The methodology and assumptions utilized in stress analysis are reasonable and consistent with BWRVIP-25, and therefore are acceptable. The NRC staff concludes that there is reasonable assurance that the VYNPS core plate bolts are structurally acceptable for 60 year plant life.

<u>References</u>

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3. Letter from Vaughn Wagoner to NRC dated July 17, 1997, Subject: License Renewal Appendix B to BWR Vessel and Internals Project BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25), EPRI Report TR-107284, December 1996

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5. Letter from Michael J. Colomb to NRC dated March 18, 2011, Subject: "Core Plate Hold Down Bolt Inspection Plan and Analysis," Vermont Yankee Nuclear Power Station Docket No. 50-271 License No. DPR-28 (BVY 11-021) (ADAMS Accession No. ML110840068)

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12. Letter from Christopher J. Wamser to NRC dated February 1, 2012, Subject: "Response to Request for Additional Information Regarding Core Plate Hold-down Bolt Inspection Plan and Analysis, Vermont Yankee Nuclear Power Station Docket No. 50-271 License No. DPR-28 (BVY 12-008)(ADAMS Accession No. ML12037A066)

13. NUREG-1475, Rev. 1,"Applying Statistics", March 2011

14. Letter from Christopher J. Wamser to NRC dated January 5, 2012, Subject: "Response to Request for Additional Information Regarding Core Plate Hold-down Bolt Inspection Plan and Analysis, Vermont Yankee Nuclear Power Station Docket No. 50-271 License No. DPR-28 (BVY 12-008)(ADAMS Accession No. ML120100126)

15. Vermont Yankee Nuclear Power Station (VYNPS) Updated Final Safety Analysis Report (UFSAR), Appendix Section C.2.5.1, Rev. 24.

16. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1965.