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2.0 REGULATORY EVALUATION

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(3)b 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[1]) 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

[1] Note, these bolts are variously referred to as the "core plate rim hold-down bolts," core plate hold-down bolts, and core plate bolts. In this SE, they will be referred to as "core plate bolts."

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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 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

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 (3.09×10^{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 7.09×10^9 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 5.2×10^{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. 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[2] (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 8×10^{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

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.

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[2] Displacements-per-atom (dpa) are a measure of irradiation damage to a material that does not exactly convert to fluence in n/cm^2, but in light-water reactor neutron spectra, 1 dpa $\cong 6.7 \times 10^{20}$ n/cm.[2]

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 reinspection 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 11provides 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, 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 of 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.

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.

<u>References</u>

1. BWR Vessel and Internals Project BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25), EPRI Report TR-107284, December 1996 (Proprietary Information. Not Publicly Available)
2. Letter from Jack Strosnider to Carl Terry dated December 19, 1999, Subject: Final Safety Evaluation of BWRVIP Vessel and Internals Project, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)," EPRI Report TR-107284, December 1996 (TAC No. M97802) (ADAMS Accession No. ML993620267) (Proprietary Information. Not Publicly Available)

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
4. Letter from Christopher Grimes to Carl Terry dated December 7, 2000, Subject: Safety Evaluation for Referencing of BWR Vessel and Internals Project, BWR Core Plate Inspection And Flaw Evaluation Guidelines (BWRVIP-25) Report for Compliance With the License Renewal Rule (10 CFR Part 54) and Appendix 0, BWR Core Plate Demonstration of Compliance with the Technical information Requirements of the License Renewal Rule (10 CFR 54.21)
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)
6. NEDC-33618P – Revision 0, Vermont Yankee Core Plate Bolt Stress Analysis, March 2011, (ADAMS Accession No. ML110840070 - Proprietary Version, ML110840069 – Non-Proprietary Version)
7. Safety Evaluation Report Related to the License Renewal of Vermont Yankee Nuclear Power Station (NUREG-1907 Vol. 2), May 2008 (ADAMS Accession No. ML081430109)
8. Letter from Christopher J. Wamser to NRC dated December 9, 2011, Subject: "Response to Request for Additional Information for Core Plate Hold Down Bolt Inspection Plan and Analysis, Vermont Yankee Nuclear Power Station Docket No. 50-271, License No. DPR-28 (BVY 11-078) (ADAMS Accession No. ML11353A407)
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10. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and threshold Values (MRP-175) 1012081, Topical Report, December 2005 – Proprietary (ADAMS Accession No. ML063470637); Non-proprietary version (ADAMS Accession No. ML061880278)
11. Letter from Michael J. Colomb to NRC dated March 18, 2011, Subject: "Deviation from BWRVIP-25 Inspection Requirements, Vermont Yankee Nuclear Power Station," Docket No. 50-271, License No. DPR-28 (BVY 11-024) (ADAMS Accession No. ML110840044)
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)