UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 23, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – SAFETY EVALUATION RE: LICENSE RENEWAL COMMITMENT NO. 23, CORE PLATE RIM HOLD-DOWN BOLTING (TAC NO. ME9698)

Dear Sir or Madam:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 28, 2012, as supplemented by letters dated April 17, September 27, and October 3, 2013, Entergy Nuclear Operations (Entergy) submitted a plant-specific analysis concerning the core plate rim hold-down bolts located at the James A. Fitzpatrick Nuclear Power Plant (JAFNPP). In Amendment 5 of the JAFNPP license renewal application, Entergy committed to either install core plate wedges or complete a plant-specific analysis to determine acceptance criteria for continued inspection of the core plate rim 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 2-years prior to the period of extended operation.

The NRC staff has found that Entergy has satisfactorily addressed License Renewal Commitment No. 23 of the license renewal safety evaluation and the regulatory requirements associated with this commitment. The staff's safety evaluation is enclosed.

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure:
Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO LICENSE RENEWAL COMMITMENT NO. 23

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

1.0 INTRODUCTION

By letter dated September 28, 2012 (Reference 1), as supplemented by letters dated April 17, 2013 (Reference 5), September 27, 2013 (Reference 3), and October 3, 2013 (Reference 2), Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a plant-specific analysis concerning the core plate hold-down bolts for the James A. Fitzpatrick Nuclear Power Plant (JAFNPP). This report was submitted in accordance with License Renewal Commitment No. 23, as documented in the "Safety Evaluation Report Related to the License Renewal of JAFNPP" dated February 2008 (Reference 6), whereby the licensee committed to either (1) Install core plate wedges prior to the period of extended operation (PEO), or (2) Complete a plant-specific analysis to determine acceptance criteria for continued inspection of core plate rim hold-down bolting in accordance with BWR [boiling-water reactor] Vessel and Internals Project BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) (Reference 4) and submit the inspection plan, along with acceptance criteria and justification for the inspection plan, to the Nuclear Regulatory Commission (NRC) 2 years prior to the PEO for NRC review and approval. Entergy selected Option 2.

Reference 1 states that a plant-specific analysis was performed to evaluate the susceptibility of the JAFNPP core plate bolts to known degradation mechanisms, calculate the relaxation of bolt preload over 60 years of operation, evaluate the flaw tolerance of the bolts, and calculate the minimum number of bolts required to prevent horizontal displacement. Conservative methods were used for each evaluation, and these conservatisms are compounding. This plant-specific analysis was conducted by Structural Integrity Associates, and concluded that:

- JAFNPP core plate bolts have a low susceptibility to inter-granular stress corrosion cracking (IGSCC) and that crack initiation would be unlikely based on plant properties,
- JAFNPP core plate bolts would not be susceptible to irradiation assisted stress corrosion cracking based on material type and water chemistry, and

- JAFNPP core plate bolts would not be susceptible to degradation by thermal fatigue or flow induced vibration.

Having concluded this, the licensee further stated that:

- If single or multiple IGSCC flaws initiated in the JAFNPP core plate bolting subsequent to 1995, then the residual life of the core plate bolting is on the order of 40-50 years,
- The minimum number of core plate bolts required would be no more than 56 under conservatively posed analyses assumptions, and
- Accounting for additional supporting structural components, a minimum number of core plate bolts would be 48 bolts, and potentially 16 bolts, if no gap was assumed for the same service level and same number of aligner pins in contact.

The licensee provided confidence in these results by asserting that no core plate bolt failures had occurred, nor any signs of degradation have been observed during the three previous visual inspections of the JAFNPP core plate bolts, including a 100-percent baseline inspection of all 72 bolts.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(a)(3) requires that for each component identified in 10 CFR 54.21(a)(1), the licensee 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 PEO, as defined in 10 CFR 54.4, "Scope." Section 54.21(a)(1) of 10 CFR also 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) Considers 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.

Section 54.21(c)(1) of 10 CFR requires the licensee to demonstrate that for each TLAA:

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

3.0 TECHNICAL EVALUATION

3.1 Licensee's Basis for Acceptance of Core Plate Hold-down Bolt Stress Analysis

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 a 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. 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. The 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 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. The critical number of intact hold-down bolts required to prevent lateral displacement during a seismic event is unique to each plant, and can be determined from a plant-specific analysis. Even if hold-down bolt failures result in significant core plate movement preventing the insertion of control rods, the plant can still be brought to a safe shutdown condition using the standby liquid control system. Core plates experience tensile stresses and have stress concentrations in the threaded regions. General Electric-Hitachi has also determined that core plate bolt stress relaxation occurs due to thermal and irradiation effects.

Section 2.4.2 of Reference 1 states that the gap between the aligner pin and the core support ring results in an impact load between the aligner pin and the core support ring when the reactor internals are subjected to horizontal loading resulting from a seismic event. However, neither the summary of this evaluation in the aforementioned attachment nor the calculation that considers the aligner pins contribution to resisting the horizontal loading (JAF-CALC-12-00017, "Minimum Required Number of Core Plate Bolts – Consideration of Aligner Brackets," Reference 4, Enclosure 1) discusses the core plate bolt hole gaps. In this context, the NRC staff, in Request for Additional Information (RAI) 1, by letter dated September 27, 2013 (Reference 3), requested that the licensee provide information regarding the nominal design values for the gaps between the core plate bolts and core plate bolt holes. The staff also requested confirmation that the gap between the aligner pin and the core support ring is smaller than any potential gap between the core plate bolts and the core plate bolt holes on the core plate. Further, the staff requested the licensee to determine whether the core plate bolt hole gap could potentially be closed prior to the aligner pin gap such that core plate bolt bending action and frictional resistance at the core plate rim would resist the applied horizontal loads instead of the scenario postulated in JAF-CALC-12-00017. If the core plate bolt hole gap is smaller, such that the core plate bolt and core plate impact occurs prior to the impact between the aligner pin and core support ring, the staff requested the licensee to determine what effect this has on the minimum number of bolts required to withstand the applied horizontal loads.

In Reference 2, the licensee states:

> ... the core plate bolt holes have a nominal diameter of 1.375 inches with a fractional tolerance of 1/16 or 0.0625 inches and a center line (X_{CS} and Y_{CS}) tolerance of 0.060 inches on diameter. Since the tolerance of 0.060 inches is the diameter of the tolerance zone about the true position given by the Basic Dimension, the as-built centerline of the hole can be anywhere in a cylinder of 0.060 inches diameter about the true position, which results in a maximum of 0.030 inches offset from true position. Therefore, the minimum diameter of the bolt holes is:
>
> $$1.375 - 0.0625 = 1.283 \text{ inches}$$
>
> ... the nominal diameter of the bolts is 1.125 inches with a tolerance of ± 1/32 or 0.03125 inches on diameter; therefore, the maximum diameter of the bolt is:
>
> $$1.125 + 0.03125 = 1.156 \text{ inches}$$
>
> The average minimum radial clearance of the bolts within the bolt holes is:
>
> $$\frac{(1.288-1.156)}{2} = 0.0635 \text{ inches}$$
>
> ... the aligner pin assembly has a typical gap of 0.010 ± 0.005 inches, so the maximum typical gap is 0.015 inches. The average minimum radial clearance between the through hole in the core plate and the core plate bolt is approximately 4 times larger than the maximum typical gap of 0.015 inches expected between the aligner pin and bracket assembly; therefore, it is reasonable to expect that the aligner pin gap will close prior to the bolt hole gap.
>
> It should also be noted that the results of JAF-CALC-12-00017 were not intended to justify taking credit for a reduced number of bolts, but rather to show that redundant load carrying capacity is present when considering the contribution of the aligner pins. The inspection protocol developed in JAF-CALC-12-00018 was developed without taking credit for the aligner pin and bracket assemblies.

In Reference 5, Enclosure 1, the sixth assumption provided on page 7 of calculation JAF-CALC-12-00016, "Minimum Required Number of Core Plate Bolts," notes that the remaining numbers of bolts calculated are assumed to be evenly distributed around the core plate. The basis for this assumption is that no region of the core plate is more susceptible to degradation than others. As described in Section 3.3.4.1 of the JAFNPP Final Safety Analysis Report, the JAFNPP core plate consists of the typical circular stainless steel plate stiffened with a rim and beam structure. The basic construction of the core plate ensures that the loading will not be evenly distributed due to the core plate stiffener beams that tend to place higher loads on some core plate bolts than others. This results in an uneven load distribution on the core plate bolts. In this context, the staff, in RAI 2, by letter dated September 27, 2013, requested that the licensee describe what impact this has on the analysis results for both calculations (JAF-CALC-12-00017 and JAF-CALC-12-00016).

In Reference 2, the licensee states:

> It is recognized that an uneven load distribution exists on the core plate bolts for some loading scenarios, such as those encountered during transient loading events. Appendix A of BWRVIP-25 provides an illustration of the uneven vertical and horizontal load distributions on the core plate bolts for an example analysis considering Level A/B and Level C/D loadings.
>
> The degradation referred to in JAF-CALC-12-00016 is the failure of bolts due to intergranular stress corrosion cracking (IGSCC), and the assumption states that the locations of the remaining intact bolts are assumed to be evenly distributed. IGSCC is a time dependent degradation mechanism and occurs primarily under sustained loads during normal operating conditions. Short duration transient loads do not act over a long enough time period to result in significant IGSCC growth.
>
> The normal operating loads on the core plate are deadweight (compressive force on the core plate rim reacted by the core plate support ring) and the reactor internal pressure difference (RIPD) across the core plate. The RIPD load is expected to be fairly evenly distributed since it acts over the entire core plate surface and is not an inertial load which will by definition accumulate where there is more mass and stiffness in the structure (at the stiffener beams). The dominant sustained load acting on the core plate bolts is due to bolt preload which is not affected by the basic construction of the core plate. Further, proper specification of bolt preload is such that the preload exceeds the operating loads; thereby preventing separation of the joint and consequent vibration or movement. Under the applied sustained loads there is no reason to expect that any region of the core plate is more susceptible than others to IGSCC degradation at the bolt locations.
>
> With respect to the evaluations for the minimum required number of remaining bolts in JAF-CALC-12-00016 and JAF-CALC-12-00017, the vertical loads are used to develop a resultant normal force that contributes to the friction force between the core plate rim and the core plate support ring. The friction force is compared to the applied horizontal forces to determine the amount of margin. The total resultant normal force is not affected by an uneven load distribution because it is the sum of the individual normal forces contributed by the individual intact bolts. Consequently, the friction force available to resist the applied seismic load is not affected by a potential distribution of vertical loading as observed in the sample BWRVIP-25 analysis in which a configuration with fewer bolts and low preload is evaluated. Further, the JAF [JAFNPP] core plate configuration has 72 bolts compared to the 32 used in the example analysis in BWRVIP-25, which would have resulted in a more even load distribution than that shown in the sample BWRVIP-25 analysis due to closer proximity of the bolts.
>
> Accordingly, the analysis results for JAF-CALC-12-00016 and JAF-CALC-12-00017 remain valid.

Reference 4, Appendix A ("Example Core Plate Bolt Analysis") to Boiling Water Reactor Vessel and Internals Project (BWRVIP)-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," considers three different loading scenarios for the core plate bolts under design basis loads: (1) loads on the core plate bolts with no credit for aligner pins; (2) shear-only load on the aligner pins with no credit for horizontal bolt restraint; and (3) load on the core plate bolts with no credit for aligner pin and also with the stiffener beam-to-rim weld cracked. Scenario 1 is the only scenario that appears to be explicitly considered in the present evaluation of the JAFNPP core plate bolts. In this context, the NRC staff, in RAI 3, by letter dated September 27, 2013 (Reference 3), requested that the licensee discuss how the other scenarios used in BWRVIP-25 are addressed by the evaluations performed for the JAFNPP core plate bolts.

In Reference 2, the licensee states:

> The example analysis presented in Appendix A of BWRVIP-25 is a sample analysis that investigates structural characteristics and considers three postulated scenarios as follows:
>
> 1. Determine load on core plate bolts with no credit for aligner pins.
>
> 2. Determine shear load on aligner pins with no credit for core plate bolts.
>
> 3. Determine load on core plate bolts with no credit for aligner pins or rim weld.
>
> As indicated in RAI 3, Scenario 1 is considered in JAF-CALC-12-00016 by evaluating the friction force needed to resist the horizontal forces due to Level A/B and Level C/D events. The second and third scenarios used in BWRVIP-25 are specifically not addressed because they do not apply when determining an inspection protocol for the core plate bolts.
>
> Scenario 2 does not apply because not all bolts are assumed to be failed in the JAF analysis. The impact of the aligner pins is considered in JAF-CALC-12-00017, but the objective of the evaluation is to determine the additional margin provided by the aligner pins, not to assume that the bolts have all failed and do not contribute structurally.
>
> Scenario 3 does not apply because whether the rim weld is intact or not is irrelevant as long as sufficient preload force in the bolts exists to keep the joint together and provide enough friction force to prevent movement of the core plate. Further, calculation JAF-CALC-12-00016 does not take any credit for the aligner pin assembly. Table 3-2 of BWRVIP-25 specifically indicates that inspections of the rim weld are not required because failure of this weld has no adverse safety consequences.
>
> Therefore, Scenario 1 is the only relevant scenario for determining an inspection protocol for the JAF core plate bolts. The evaluations in JAF-CALC-12-00016 and JAF-CALC-12-00017 determine the minimum number of bolts required to ensure sufficient friction force exists to prevent movement of the core plate. Further, the inspection protocol developed in JAF-CALC-12-00018 was

developed without taking credit for the aligner pin and bracket assemblies, which is conservative.

In Reference 5, the licensee states:

> By taking structural credit for the aligner pin and bracket assembly, the minimum number of core plate bolts required to ensure negligible relative horizontal displacement of the core plate is between 7 and 39 for Service Level A/B and between 10 and 48 for Service Level C/D. These are comparable to the 45 bolts for Service Level A/B and 56 bolts for Service Level C/D calculated in JAF-CALC-12-00016 as shown:

Results Summary

Service Level	No. of Bolts	Friction Force (lbf)	Horizontal Force (lbf)
A/B	72	135759	65250
	45	66402	
C/D	72	129311	87000
	56	88210	

Note: Friction force values are shown as absolute values.

The present calculation gives a lower number of required core plate bolts by taking structural credit for the core plate aligner pin bracket assembly.

Loss of Preload of Core Plate Bolts Due to Fluence

The core plate bolt preload relaxation evaluation and relevant assumptions are contained in SI-Calculation No. 1101291.302, Revision 0, "Core Plate Bolt Preload Relaxation." The licensee performed a literature review regarding the relevant mechanisms of preload reduction and the associated analysis methods. The following mechanisms were identified by the licensee as relevant to the JAFNPP core plate bolts:

- Thermal Relaxation
- Stress Relaxation
- Radiation Relaxation

In its evaluation, the licensee used the term "thermal relaxation" to describe the loss of preload associated with thermal effects on temperature dependent material properties. These effects will contribute to the reduction in preload at operating temperatures. Thermal relaxation occurs due to thermal effects on temperature dependent material properties and can result in both a temporary (i.e., recoverable) reduction in preload due to a change in the modulus of elasticity and a permanent loss of preload due to a change in the yield strength and consequent yielding of the material at an elevated temperature. For this evaluation, the licensee took the elevated temperature as the operating temperature. Thermal relaxation was evaluated by the licensee using representative stress-strain curves for Type 304 SS [stainless steel] by identifying the

strain due to the preload stress at room temperature and determining the equivalent stress for constant strain on an elevated temperature curve representing the operating temperature.

From Figure 5-1 of SI Calculation No. 1101291.302, Revision 0, the licensee determined that the approximate preload stress, after thermal relaxation, was 17,500 pounds per square inch (psi). The licensee noted that this value corresponds to a 23.4-percent reduction in preload due to thermal relaxation, and accounts for both the reduction in modulus and the effect of yielding.

The licensee noted that stress relaxation occurs due to a creep mechanism in the material. Stress relaxation was evaluated by the licensee based on the temperature, stress, and time of operation. The potential relaxation effects of both primary and secondary creep were assessed for Type 304 SS, through evaluation of available information. It was noted by the licensee that creep deformation of metals occurs in three stages: primary creep, secondary creep and tertiary creep. For the core plate bolts, the licensee determined that primary creep is most relevant, and further stages of creep (i.e., secondary and tertiary) are considered negligible. The licensee noted that secondary (steady state) creep is typically considered a high temperature phenomenon, and the temperatures in a Report No. 1101291.401.R0 2-6 for operating BWRs are generally regarded to be outside of the secondary creep regime. Since secondary creep is negligible, tertiary creep is also negligible. Stress relaxation does occur at lower temperatures and results in primary creep. The licensee noted that after thermal relaxation, the remaining preload stress is 17,500 psi. The licensee took 17,500 psi as the initial preload that will be affected by primary creep and considered the average curve in Figure 5-2 of Calculation No. 1101291.302, Revision 0, which results in a relaxation stress of approximately 1,200 psi. This value corresponds to a 6.8 percent additional reduction in preload due to stress relaxation. For the licensee's evaluation, the thermal relaxation occurs over a short time scale (first heat-up); therefore, the primary creep affects the preload after thermal relaxation has occurred. Further, the licensee noted that since this evaluation considers primary creep for all core plate bolts, it is more appropriate to evaluate the average primary creep relaxation and apply this to all core plate bolts than to assume the maximum creep relaxation occurs for all core plate bolts.

Radiation relaxation or irradiation creep is a fluence (time) dependent deformation process that affects stainless steels in the light-water reactor (LWR) environment. In their evaluation of radiation relaxation, the licensee used a fluence value of energy greater than 0.1 MeV [mega-electron volt]. The licensee noted that this value was consistent with the approach in General Electric Document No. NEDE 13334, Class II, "A Study of Stress Relaxation in AISI 304 Stainless Steel," April 1973, SI File No. EPRI [Electric Power Research Institute]-179-202. The licensee also stated that this value is conservative compared to the use of fluence values for energy greater than 1.0 MeV. The maximum value of average fluence, along the loaded length of the core plate bolts, for all bolts around the core plate, was used to calculate the relaxation along each bolt and was bounding for all other azimuthal locations. Radiation relaxation was evaluated by the licensee utilizing data from three different sources as discussed in SI Calculation No. 1101291.302, Revision 0, "Core Plate Bolt Preload Relaxation," and by comparing the results to develop a reasonable and bounding loss of preload.

The licensee found that the results of the three separate resources used to evaluate radiation relaxation in the 304 SS core plate bolts agreed well with each other. The range of average preload stress reduction due to radiation relaxation is approximately 3 to 8 percent. To be conservative for this evaluation the licensee selected 8 percent.

Radiation relaxation occurs over an extended period of time. Therefore, the equivalent total percent reduction in preload was calculated by the licensee as follows:

Total % Reduction = 1 – (1 – 0.234) * (1 – 0.068) * (1 – 0.08) = 34.3%

Based on an initial preload of 19,556 lbf, the remaining preload in each bolt at 54 effective full power years (EFPY) is 12,844 lbf.

3.2 NRC Staff Evaluation

The NRC staff reviewed the information from the licensee for completion of License Renewal Commitment No. 23, whereby the licensee provided an analysis of the JAFNPP core plate hold-down bolts that demonstrates their adequacy considering potential loss of preload through the PEO, and found it to be acceptable based on the following with the exception that no further inspections are required:

1. No obvious signs of degradation have been observed during the three previous visual inspections of the JAFNPP core plate bolts,
2. The JAFNPP core plate bolts have a low susceptibility to IGSCC, and initiation is unlikely based on the material and manufacturing specification requirements, and
3. The JAFNPP core plate design provides at least 22-percent excess number of bolts (56/72 bolts), even when considering the relaxation of bolt preload over the 60-year plant life; thus, ensuring the ability to insert the control rod drives and safely shut down the plant in a design basis seismic event. This excess capacity doesn't credit the aligner pins, which would further increase margin.

The NRC staff used BWRVIP-25 as guidance for the review of the licensee's evaluation of stress relaxation of the core plate 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 applicable for most BWRs. However, because the staff has not previously approved a calculated or estimated plant-specific value for the core plate bolt neutron fluence, the staff requested that the licensee provide the details of the flux evaluation that was used to determine projected total fast neutron fluence for the core plate bolts. The licensee provided the fluence evaluation conducted specifically for the core plate bolts as part of the RAI response (Reference 5). The licensee indicated that the flux evaluation was based on a best-estimate flux evaluation performed using the NRC approved Radiation Analysis Modeling Application (RAMA) fluence methodology (Reference 7).

The NRC staff notes that the RAMA neutron fluence methodology has only been generically approved by the NRC for determining projected neutron fluence for the reactor pressure vessel. As stated in the safety evaluation accompanying BWRVIP-145NP-A, "it should be noted that the RAMA fluence methodology is based on the guidance of Regulatory Guide 1.190 [Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence], which was developed specifically for reactor pressure vessels in order to satisfy General Design Criteria (GDC) 14, 30, and 31 pertaining to reactor coolant pressure boundaries. Since reactor vessel internals are not reactor coolant pressure boundary components, GDC 14, 30 and 31 do not apply to the reactor vessel internals." The staff does note that RAMA's application to the core shroud and top guide components has received limited approval by the staff for those cases where the licensee has demonstrated on a plant-specific basis that the calculational

results are conservative for the intended application, as discussed in the staff's safety evaluation for BWRVIP-145NP-A.

The RAMA methodology has not been approved by the NRC staff for neutron fluence projections for the core plate hold-down bolts because it has not been benchmarked based on analysis of irradiated core plate hold-down bolt material specimens. However, the staff finds that there are substantial margins available for 54 EFPY core plate bolt stress relaxation values based on the stress analysis results provided in in Reference 1. Therefore, given the conservatisms assumed in the licensee's fluence calculation in concert with the staff's findings regarding the consistency of the licensee's calculated core plate bolt preload reduction compared to the results provided in Appendix B of BWRVIP-25 (discussed below), the staff finds the licensee's application of the RAMA methodology to be adequate for the core plate bolt stress relaxation analysis.

The purpose of the fluence evaluation was to identify the peak bounding fluence profile for the core plate bolts. The fluence was estimated implicitly from values for adjoining components through which the bolts passed. The power history at Fitzpatrick was modeled from operations data and forecasted future operations resembling recent cycles. An uncertainty analysis was then performed to conservatively estimate the bolt fluences. The NRC staff examined the report in detail and concluded that the licensee's estimate of bolt fluence was credible.

The NRC staff finds the licensee's RAI response acceptable because it provides an adequate description of how the core plate hold-down bolt flux was calculated, and includes appropriate conservatisms to ensure the flux used to project the loss of preload is bounding. Specifically, the lack of bias correction provides a significant source of conservatism. Therefore, the staff finds this RAI concern resolved.

The NRC staff verified the percentage reduction in preload by confirming the preload analysis provided in the RAI response. The staff compared the licensee's prediction of the reduction in preload to other industry data relevant to stress relaxation in BWRs, such as that contained in BWRVIP-99-A, "BWR Vessel and Internals Project – Crack growth Rates in Irradiated Stainless Steels in BWR Internal Components" (Reference 8). The licensee calculated thermal and creep based relaxation according to well recognized principles and published material behavior data. The staff reviewed the calculations cited in the licensee's analysis and confirmed their accuracy. The licensee concluded that due to all three relaxation mechanisms, the bolt preload would be reduced by 34.3 percent by 54 EFPY.

More significantly, the licensee conducted a comparison by using an average fluence versus using a discrete numeric technique to calculate the overall relaxation of a bolt. The analysis results indicated that the two approaches produce virtually identical results. The NRC staff reviewed the analysis and concluded that it credibly demonstrated the value and accuracy of using average fluence.

The NRC staff finds the licensee's evaluation of the projected loss of preload of the Fitzpatrick core plate hold-down bolts due to irradiation-assisted stress relaxation was appropriately determined because a) the licensee appropriately estimated the peak fluence for the bolts at end of life based on its extended power uprate fluence evaluation; and b) 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.

Inspection Plan for Core Plate Hold-Down Bolts

The licensee's proposed inspection plan was to conduct no further inspections relying on the results of previous inspections and analyses included in the application. The licensee's inspection plan was found by the NRC staff to be unacceptable because it was a deviation from the BWRVIP-25 recommendations. BWRVIP-25 recommends inspection of 50 percent of the bolts and if cracking is detected in any of these first 50 percent, the remaining 50 percent should be inspected. Subsequently, by letter dated September 27, 2013 (Reference 3), the licensee proposed a revised inspection plan that will inspect 50 percent of the bolts every other outage by VT-3 from the top starting with R022 in October, 2016, and that this plan will remain in place until the BWRVIP-25 guidance on inspecting core plate bolting is revised and approved by the staff.

Based on the NRC staff's review, the staff finds that the licensee's inspection plan provided in its letter dated September 27, 2013 (Reference 3), acceptable and meets License Renewal Commitment No. 23 of the license renewal safety evaluation and the regulatory requirements associated with this inspection commitment.

4.0 CONCLUSIONS

The NRC staff has completed its assessment of the information presented by the licensee to License Renewal Commitment No. 23, as stated in the license renewal safety evaluation. The licensee performed a plant-specific analysis of the core plate bolts in accordance with BWRVIP-25 to show that the bolts will maintain adequate preload such that they will be able to perform their required safety function during the PEO. Based on the information presented by the licensee, the staff finds the licensee's assessment of the stresses in the core plate bolts acceptable for the PEO, considering the effects of stress relaxation and subsequent loss of preload, such that these bolts would be expected to maintain adequate structural integrity to withstand applicable design basis loads throughout the 60-year period of licensed operation of JAFNPP.

In addition, the NRC staff concludes that the licensee's evaluation of the projected loss of preload of the JAFNPP core plate hold-down bolts due to irradiation-assisted stress relaxation is acceptable because a) the licensee appropriately estimated the peak fluence for the bolts at end of life based on its extended power uprate fluence evaluation; and b) 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. Furthermore, the staff concludes that the licensee's proposed inspection plan provided in its letter dated September 27, 2013 (Reference 3), to inspect 50 percent of the bolts every other outage by VT-3 from above the core plate starting with R022 in October 2016, which will remain in place until BWRVIP-25 guidance on inspecting core plate bolting is revised, is acceptable. Therefore, the staff concludes that the licensee has satisfactorily addressed License Renewal Commitment No. 23 of license renewal safety evaluation and the regulatory requirements associated with this inspection commitment.

5.0 REFERENCES

1. Letter from Chris Adner, Entergy, to NRC Document Control Desk, "Core Plate Rim Hold-down Bolting, Plant Specific Analysis and Inspection Plan, License Renewal Commitment # 23, James A. FitzPatrick Nuclear Power Plant" dated September 28, 2012 (ADAMS Accession No. ML12275A140).

2. Letter from Chris M. Adner, Entergy, "Response to Request for Additional Information Core for Plate Rim Hold-down Bolting, License Renewal Commitment #23, (TAC No. ME9698), dated October 3, 2013 (ADAMS Accession No. ML13276A482).

3. Letter from Chris M. Adner, Entergy, "Response to Request for Additional Information Core for Plate Rim Hold-down Bolting, Plant Specific Analysis and Inspection Plan, License Renewal Commitment #23), James A. Fitzpatrick Nuclear Power Plant Docket No. 50-333 License No. DPR-59," dated September 27, 2013 (ADAMS Accession No. ML13273A358).

4. 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).

5. Letter from Chris M. Adner, Entergy, "Response to Request for Additional Information Core for Plate Rim Hold-down Bolting, License Renewal Commitment #23, JAFNPPP-13-0035, dated April 17, 2013 (ADAMS Accession No. ML14063A653).

6. Safety Evaluation Report Related to the License Renewal of James A. Fitzpatrick Nuclear Power Plant Docket No. 50-333," dated February 2008 (ADAMS Accession No. ML080250372).

7. Letter from William H. Bateman, U.S. NRC, to Bill Eaton, BWRVIP, "Safety Evaluation of Proprietary EPRI Reports BWRVIP-114, -115, -117, and -121 and TWE-PSE-001-R-R001," dated May 13, 2005 (Proprietary Information. Not Publicly Available).

8. Letter from Christopher Grimes, NRC, to Carl Terry, BWRVIP Chairman, dated December 7, 2000, "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)," dated December 7, 2000 (ADAMS Accession No. ML003775989).

Principal Contributors: Thomas McLellan
Dan Hoang

Date: July 23, 2014

July 23, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – SAFETY EVALUATION RE: LICENSE RENEWAL COMMITMENT NO. 23, CORE PLATE RIM HOLD-DOWN BOLTING (TAC NO. ME9698)

Dear Sir or Madam:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 28, 2012, as supplemented by letters dated April 17, September 27, and October 3, 2013, Entergy Nuclear Operations (Entergy) submitted a plant-specific analysis concerning the core plate rim hold-down bolts located at the James A. Fitzpatrick Nuclear Power Plant (JAFNPP). In Amendment 5 of the JAFNPP license renewal application, Entergy committed to either install core plate wedges or complete a plant-specific analysis to determine acceptance criteria for continued inspection of the core plate rim 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 2-years prior to the period of extended operation.

The NRC staff has found that Entergy has satisfactorily addressed License Renewal Commitment No. 23 of the license renewal safety evaluation and the regulatory requirements associated with this commitment. The staff's safety evaluation is enclosed.

Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333
Enclosure:
Safety Evaluation
cc w/encl: Distribution via Listserv

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