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

November 23, 2015

Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT RE: REQUEST FOR APPROVAL OF 10 CFR PART 50 APPENDIX G EQUIVALENT MARGINS ANALYSIS (CAC NO. MF5163)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 258 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant. The amendment approves changes to the final safety analysis report (FSAR) in response to your application dated November 12, 2014, as supplemented by letter dated January 28, 2015.

The amendment approves the licensee's proposed revisions to information in the FSAR regarding the reactor pressure vessel (RPV) Charpy upper-shelf energy (USE) requirements of Title 10 of the *Code of Federal Regulations* Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements," IV.A.1. The change updates the analysis for satisfying the RPV Charpy USE requirements through the end of the renewed operating license.

A copy of our related safety evaluation is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Jennivine K. Rankin, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 258 to DPR-20
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
License No. DPR-20

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee), dated November 12, 2014, as supplemented January 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, by Amendment No. 258, the licensee is authorized to revise the final safety analysis report (FSAR), as set forth in the application dated November 12, 2014, as supplemented by letter dated January 28, 2015. The revised FSAR incorporates the equivalent margins analysis for satisfying the reactor pressure vessel Charpy upper-shelf energy requirements. The analysis is described in the licensee's application dated November 12, 2014, as supplemented by letter dated January 28, 2015, and the NRC staff's related safety evaluation for this amendment.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: November 23, 2015

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 258 FOR

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

ENTERGY NUCLEAR OPERATIONS, INC.

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

Enclosure 2

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1.0 INTRODUCTION

By letter dated October 21, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13295A448; part of ADAMS Package Accession No. ML13295A446), Entergy Nuclear Operations, Inc. (ENO, the licensee) stated that, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, it transmits for U.S. Nuclear Regulatory Commission (NRC) review and approval an equivalent margins analysis (EMA) documented in WCAP-17651-NP, Revision 0, "Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis" (ADAMS Accession No. ML13295A451) for two traditional beltline and one extended beltline reactor vessel materials. The licensee supplemented its request by letters dated June 12, 2014, and June 26, 2014 (ADAMS Accession Nos. ML14163A662 and ML14177A707, respectively). During the NRC staff review, the NRC determined that a license amendment request (LAR) under 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site permit," was required. This is documented in the licensee's letter dated December 18, 2014, in which the licensee withdrew the submittal (ADAMS Accession No. ML14352A044). By letter dated December 29, 2014 (ADAMS Accession No. ML14356A044), the NRC acknowledged the request to withdraw the application.

Subsequently, by letter dated November 12, 2014 (ADAMS Accession No. ML14316A190; part of ADAMS Package Accession No. ML14316A370), the licensee submitted a LAR for approval of the EMA for the Palisades Nuclear Plant (PNP) reactor pressure vessel (RPV). The licensee seeks to demonstrate that lower values of Charpy upper-shelf energy (USE) (less than 50 ft-lb) will provide margins of safety against fracture, equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Code, as discussed in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Section IV.A.1. As part of the LAR, the licensee attached the following:

- Description and assessment of the requested changes (ADAMS Accession No. ML14316A193).
- Responses to request for additional information questions (ADAMS Accession No. ML14316A198).
- Westinghouse WCAP-17403-NP, Revision 1, "Palisades Nuclear Power Plant Extended Beltline Reactor Vessel Integrity Evaluation," dated January 2013 (ADAMS Accession No. ML14316A199).
- Westinghouse WCAP-15353, Supplement 2-NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation," dated July 2011 (ADAMS Accession No. ML14316A207).
- Westinghouse WCAP-17651-NP, Revision 0, "Palisades Nuclear Power Plant Reactor Vessel Equivalent Margins Analysis," dated February 2013 (ADAMS Accession No. ML14316A208).

The licensee supplemented its request by letter dated January 28, 2015 (ADAMS Accession No. ML15028A525; part of ADAMS Package Accession No. ML15029A061). The response to a request for additional information (RAI) was attached to the letter (ADAMS Accession No. ML15028A527).

The supplement, dated January 28, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 6, 2015 (80 FR 523).

2.0 REGULATORY EVALUATION

The NRC staff determined that the following regulations and guidance aree applicable to the proposed amendment:

10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," states that whenever a holder of an operating license desires to amend the license, the holder must file a request with the Commission, fully describing the changes desires, and following as far as applicable the form prescribed for original applications.

10 CFR 50.92(a), states that in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. The considerations for issuance of initial operating licenses are given in 10 CFR 50.57, "Issuance of operating license."

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operations," states that unless the Commission has granted exemptions pursuant to 10 CFR 50.12, "Specific exemptions," reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in 10 CFR 50, Appendix G and Appendix H.

10 CFR 50, Appendix G, "Fracture toughness requirements," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operations, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Code forms the basis for the requirements of Appendix G.

As stated in 10 CFR 50 Appendix G, Section IV.A.1.a:

> Reactor vessel beltline materials must have Charpy upper-shelf energy[1] in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This [equivalent margins] analysis must use the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2) at the time the analysis is submitted.

[1] Defined in ASTM E185-79 and -82 which are incorporated by reference in Appendix H to part 50.

Additionally, 10 CFR 50 Appendix G, Section IV.A.1.c states, in part, the following:

> The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in §50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of Section IV.A.1....

Appendix K, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels," to the ASME Code, Section XI (Appendix K) specifies the methodology for demonstrating safety margins equivalent to those in Appendix G of Section XI of the ASME Code. For the PNP reactor vessel EMA, the applicable Code is the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI. The Appendix K methodology is based on the principles of elastic-plastic fracture mechanics (EPFM) and postulates flaws in the RPV at locations of predicted low USE. Appendix K states that the applied J-integral for these flaws should be calculated and compared with the J-integral fracture resistance of the material to determine acceptability.

Regulatory Guide (RG) 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb," dated June 1995 (ADAMS Accession No. ML003740038), describes guidance acceptable to the NRC staff for evaluating RPVs when the Charpy USE falls below the 50 ft-lb limit of 10 CFR 50, Appendix G. The document contains guidance on selecting transients for consideration and selecting appropriate material properties to be used in the EMA analyses. Therefore, if a licensee chooses to follow Appendix K, the EMA should be based on ASME Code, Section XI, Appendix K, as supplemented by RG 1.161, using the acceptance criteria, analysis methods, material properties, and selection of transients as described in RG 1.161. RG 1.161 provides methods based on generic J-integral fracture resistance (J-R) data for modeling the J-R curve for the following classes of materials: welds manufactured with Linde 80 flux, generic welds, and high toughness plate (defined as having a sulfur content less than 0.018 percent). RG 1.161 also allows (not requires) use of J-R curves for plant-specific materials, if justified on a case-by-case basis.

RG 1.161 provides limited guidance for the evaluation of plates that are low toughness due to high sulfur content because J-integral fracture toughness data for these materials are limited. RG 1.161 allows high toughness plate models to be used for base materials with sulfur contents 0.018 percent and higher, as long as sufficient justification is provided to show that the material can be considered high toughness. Alternately, RG 1.161 allows high sulfur plates to be evaluated based on compact tension specimen data available in NUREG/CR-5265, "Size Effects on J-R Curves for A-302B Plate," dated January 1989 (ADAMS Accession No. ML15201A766), with adjustments for specimen temperature and Charpy V-Notch (CVN) value. The NRC staff recently performed a periodic review of RG 1.161 which is documented in a memo dated March 13, 2014 (ADAMS Accession No. ML14070A206; part of ADAMS Package Accession No. ML14070A200). The NRC staff concluded during this periodic review that RG 1.161 is acceptable for continued use and that there are no safety concerns or a need to revise the RG at this time.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988 (ADAMS Accession No. ML003740284), describes guidance acceptable to the NRC staff for predicting decrease in USE as a function of neutron fluence and material copper content.

RG 1.99 Regulatory Position 1.2, which is based on surveillance data collected from the fleet of nuclear power reactors operating in the United States, may be used when surveillance data from the RPV in question are not available. RG 1.99 Regulatory Position 2.2, based on the measured decrease in USE from the plant-specific capsule surveillance material, may be used when two or more surveillance data sets are available.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (ADAMS Accession No. ML010890301) describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Branch Technical Position (BTP) 5-3, Revision 2, "Fracture Toughness Requirements," dated March 2007 (ADAMS Accession No. ML070850035), summarizes NRC fracture toughness requirements and provides guidance for older plants that may have been designed and built before these requirements went into force. Paragraph 1.2 of BTP 5-3 provides a method to estimate USE in the transverse direction if transverse Charpy data are not available.

3.0 TECHNICAL EVALUATION

As part of its technical evaluation, the NRC staff performed the following reviews:

- The NRC staff compared the licensee's neutron fluence methodology with the guidance provided in RG 1.190 (Section 3.3.1 of SE).
- The NRC independently verified the licensee's projected EOLE USE by applying the guidance in BTP 5-3 and RG 1.99 (Section 3.3.3 of SE).
- The NRC staff reviewed the licensee's margin of safety against ductile fracture by comparing the licensee's proposal against the regulatory criteria of 10 CFR 50 Appendix G and the guidance of RG 1.161. (Section 3.3.4 of the SE).

3.1 Background

The PNP RPV is constructed of alloy steel plates joined by axial and circumferential welds. As is common for all ferritic materials, the PNP RPV materials can be characterized by Charpy curves that have an upper shelf of high toughness at high temperatures, a lower shelf of lower toughness at low temperatures, and a transitional region between the two shelves. Ferritic materials that are exposed to high neutron fluence in the RPV beltline region will show a reduction in the USE over time. In order to provide a margin of safety against fracture, the NRC has established USE limits in 10 CFR 50 Appendix G, Section IV.A.1.a. The licensee determined that three RPV materials at PNP are projected to drop below the 50 ft-lb USE limit of 10 CFR 50 Appendix G at the end-of-life extension (EOLE, corresponding to 42.1 effective full power years). These three materials are: (1) the lower shell plate D-3804-1; (2) the intermediate-to-lower shell circumferential weld 9-112 in the beltline region (defined as directly surrounding the effective height of the active core); and (3) the upper shell plate D-3802-3 in the extended beltline region (defined as predicted to receive fluence greater than 1.0×10^{17} n/cm^2). In accordance with the requirements of 10 CFR 50 Appendix G, Section IV.A.1, an analysis, also known as an EMA, must be performed to show that the lower values of USE will provide margins of safety against fracture equivalent to those required by 10 CFR 50 Appendix G.

In the beltline region, lower shell plate D-3804-1 is projected to reach the threshold of 50 ft-lb in December 2016 and drop to 48.2 ft-lb by the EOLE. The intermediate-to-lower shell circumferential weld 9-112 is projected to reach the threshold of 50 ft-lb in November 2027 and drop to 49.6 ft-lb by the EOLE.

In the extended beltline region, upper shell plate D-3802-3 is projected to fall below 50 ft-lb at EOLE when calculated from test data points conservatively averaged from the 95 percent shear values (the maximum shear obtained during testing of this plate). The licensee projects an EOLE USE of 47.5 ft-lb for plate D-3802-3 when performing the calculations in this manner. In Westinghouse Electric Company LLC's topical report WCAP-17403-NP, Revision 1, which was identified as Attachment 3 in the licensee's November 12, 2014, EMA submittal, the licensee notes that if upper shell plate D-3802-3 is evaluated by refitting a hyperbolic tangent curve to the Charpy data and projecting the upper shelf at 100 percent shear, then an EOLE USE of 50.1 ft-lb can be obtained. Nevertheless, the licensee elected not to rely on the alternatively projected EOLE USE of 50.1 ft-lb, but to perform an EMA on plate D-3802-3.

3.2 Detailed Methodology

3.2.1 Determination of EOLE USE

EOLE USE values must be calculated for beltline materials to determine whether 10 CFR 50 Appendix G, Section IV.A.1.a criteria have been met. EOLE USE is calculated from initial USE (obtained from the Charpy test data at the beginning of service before irradiation) minus a reduction in USE calculated as a function of neutron fluence and copper content. If the calculated EOLE USE is below 50 ft-lb (in the transverse direction for base material and along the weld for weld material), an EMA must be performed as discussed in Section 2.0 of this SE. RG 1.99 provides two methods that can be used to determine the reduction in USE. Regulatory Position 1.2 of RG 1.99 may be used when surveillance data from the RPV in question is not available. RG 1.99 Regulatory Position 2.2 may be used when two or more surveillance data sets are available. Both regulatory positions require an analysis using Figure 2 of RG 1.99, which plots the predicted decrease in USE as a function of copper content and neutron fluence. The Regulatory Position 1.2 analysis uses the existing plots in Figure 2 of RG 1.99, while the Regulatory Position 2.2 analysis requires the creation of new plots derived from surveillance data and that are plotted parallel to the existing plots. In general, every plant has unirradiated Charpy data for all RPV materials obtained during the fabrication of the RPV, but has irradiated Charpy data only for the RPV materials required by the applicable edition of the American Society of Testing and Materials (ASTM) Standard Practice E185 to be part of the surveillance program. For PNP, the licensee used RG 1.99 Regulatory Position 1.2 to calculate EOLE USE for the materials projected to drop below 50 ft-lb.

As stated above, initial USE is a factor in determining EOLE USE which must be determined in the transverse direction for base material and along the weld for weld material. When transverse Charpy data are missing or insufficient to determine the initial USE value, the licensee must use a method acceptable to NRC staff to estimate transverse Charpy values based on longitudinal Charpy data. BTP 5-3 provides methods for predicting the nil-ductility transition reference temperature (RT_{NDT}) and USE for materials that do not have complete test data to determine these material properties in accordance with the ASME Code. According to BTP 5-3 Position 1.2, if Charpy tests were only made on longitudinal specimens, the values of USE should be reduced by 35 percent to estimate the transverse properties. For PNP, pre-irradiation testing of the two subject plates was performed for specimens in the longitudinal direction. Therefore, the licensee used BTP 5-3 Position 1.2 to estimate the initial USE values for these two plates in the transverse direction by reducing the longitudinal values by 35 percent. The NRC staff's evaluation of the licensee's EOLE USE calculations is in Section 3.3.3 of this SE.

3.2.2 Determination of Margins of Safety

Section IV.A.1.a of 10 CFR Part 50, Appendix G requires that, for materials which do not meet the USE criteria as discussed above, the licensee must demonstrate that "lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code." The regulation does not specify how the demonstration must be performed, other than that it must be "demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate." In 1982, the NRC staff evaluated an early version of the EMA in resolution of the unresolved safety issue A-11, "Reactor Vessel Materials Toughness," regarding RPV materials having USE below 50 ft-lb. The resolution as documented in NUREG-0744, Volume 1, Revision 1, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," dated October 1982 (ADAMS Accession No. ML15230A136) states, "When RPV steel exhibits a C_v USE level of less than 50 ft-lb...a safety analysis must be performed to ensure continued safe operation of the reactor...TAP [Task Action Plan] A-11 was designed to provide an acceptable elastic-plastic engineering method [as described in Appendix B of this NUREG]." This NUREG goes on to define safety margin as follows:

> Safety margins can be determined by comparing the loads (RPV pressure) for a condition of interest to the calculated failure load where both have been derived from elastic-plastic fracture mechanics concepts. To ensure an adequate margin of safety, the operating (or transient) pressure must remain well below the calculated failure pressure.

Subsequently, the NRC requested Section XI of the ASME Boiler and Pressure Vessel Code Committee to develop appropriate EMA criteria for demonstrating margins of safety equivalent to those in Appendix G of the ASME Code. This effort resulted in issuance of both ASME Code, Section XI, Appendix K and RG 1.161, which jointly provide an acceptable method to demonstrate that for an RPV material with a USE less than 50 ft-lb, the margins of safety against ductile fracture of an RPV with postulated flaws are equivalent to those in ASME Code, Section XI, Appendix G. RG 1.161 provides acceptance criteria based on EPFM that must be satisfied for four defined transients: Level A (normal), Level B (upset), Level C (emergency), and Level D (faulted). The equations and criteria are not the same for each level, but in each case, margin of safety is achieved by meeting the following criteria:

- For crack extension, the crack driving force must be less than the material toughness or mathematically:
    - $J_{applied} < J_{0.1}$
- For crack stability, the postulated flaw must be stable under ductile crack growth or

mathematically:

- $\partial J_{applied}/\partial a < \partial J_{material}/\partial a$ at $J_{applied} = J_{material}$

where $J_{applied}$ is the J-integral from the applied loads, $J_{0.1}$ is the material's J-integral fracture resistance at a ductile flaw growth of 0.10 in., and $J_{material}$ is the material's J-integral fracture resistance, or the J-R curve.

These criteria must be met for Levels A, B, C, and D loading as defined below.

Level A and Level B Loading

For Level A and B Conditions, the RG 1.161 method postulates a semi-elliptical surface flaw at a depth of one-quarter of the wall thickness (1/4T) of the RPV and an aspect ratio of 6 to 1 (surface length to flaw depth). For base metal evaluations, both axial and circumferential flaws should be postulated. For weld metal evaluations, the postulated flaw should be oriented along the weld of concern with the flaw plane oriented in the radial direction. For crack extension, $J_{applied}$ is calculated assuming 1.15 times maximum accumulation pressure and 1 times thermal loading using plant-specific heatup and cooldown characteristics. During evaluation, the actual safety factor can be calculated by dividing $J_{0.1}$ by $J_{applied}$. The calculated actual safety factor must exceed 1.15 for Level A and B loading. For crack stability, $J_{applied}$ is calculated assuming 1.25 times maximum accumulation pressure, with thermal loading.

Level C Loading

For Level C Conditions, the RG 1.161 method postulates a semi-elliptical surface flaw with an aspect ratio of 6 to 1 and at a depth of one-tenth of the wall thickness of the RPV plus the cladding thickness. For crack extension and crack stability, $J_{applied}$ is calculated assuming 1 times maximum accumulation pressure and 1 times thermal loading using plant-specific heatup and cooldown characteristics. During evaluation, the actual safety factor can be calculated by dividing $J_{0.1}$ by $J_{applied}$. The calculated actual safety factor must exceed 1 for Level C loading.

Level D Loading

For Level D Conditions, only the crack stability criterion is required, and the loading is the same as the Level C criteria. In addition the flaw depth, including stable tearing, should not be greater than 75 percent of the RPV thickness, and the remaining ligament should be safe from tensile instability.

3.3 NRC Technical Evaluation

3.3.1 Comparison of Neutron Fluence Methodology in WCAP-15353, Supplement 2-NP, Revision 0 with RG 1.190

The NRC staff compared the methods used to calculate neutron fluence in WCAP-15353, Supplement 2-NP, Revision 0 with the guidance provided in RG 1.190. The guidance provided in RG 1.190 indicates that the following attributes compose an acceptable fluence calculation:

- a fluence calculation performed using an acceptable methodology
- analytic uncertainty analysis identifying possible sources of uncertainty
- benchmark comparison to approved results of a test facility
- plant-specific qualification by comparison to measured fluence values

Neutron fluence is a direct input into the EOLE USE calculations used to determine if an EMA is required. Neutron fluence is also an indirect input into the EMA calculations. An evaluation of methodology applicable to the PNP RPV beltline was performed in WCAP-15353, Revision 0, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation" (ADAMS Accession No ML003686582) and WCAP-15353, Supplement 1-NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation" (ADAMS Accession No. ML110060695). The NRC staff approved the fluence methodology applicable to the RPV beltline by letters dated November 14, 2000, and December 7, 2011 (ADAMS Accession Nos. ML003768802 and ML112870050, respectively). An evaluation of methodology applicable to the RPV extended beltline was performed in WCAP-15353, Supplement 2-NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation," and submitted with the license amendment dated November 12, 2014, as Attachment 4. The NRC staff evaluation below documents the staff's review of WCAP-15353, Supplement 2-NP, Revision 0.

The licensee stated that the neutron calculations provided in WCAP-15353, Supplement 2—NP, Revision 0 are performed in a manner consistent with the guidance set forth in RG 1.190. A solution to the Boltzmann transport equation is approximated using the two-dimensional discrete ordinates transport code. The licensee uses a cross-section library based on the ENDF/B-VI nuclear data. Numeric approximations include a P_5 Legrende expansion for anisotropic scattering, and the modeling uses S_{16} order of angular quadrature. These cross-section data and approximations are in accordance with the modeling guidance contained in RG 1.190. Since the licensee used NRC-approved RG 1.190 adherent methods to determine the RPV neutron fluence, the NRC staff determined that the fluence calculations are acceptable.

Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel-cycle-specific basis. Three-dimensional flux solutions are constructed using a synthesis of azimuthal, axial, and radial flux. Source distributions include cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions, which are used to develop spatial and energy dependent core source distributions that are averaged over each fuel cycle. This method accounts for source energy spectral effects by using an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history for each fuel assembly. The neutron source and transport calculations, as described above, are performed in accordance with the guidance set forth in RG 1.190. Based on the consistency with the guidance contained in RG 1.190, the NRC staff determined that the source and transport calculations are acceptable.

The fluence methods are supported by an analytic uncertainty analysis and the estimated uncertainty is less than 20 percent, which is in accordance with RG 1.190 and hence acceptable. Details of the analytic uncertainty analysis are provided in WCAP-15353, Supplement 2—NP, Revision 0.

WCAP-15353, Revision 0, describes the methods qualification using the standard benchmark problems found in RG 1.190. WCAP-15353, Revision 0 compared the calculations with the benchmark measurements from the Poolside Critical Assembly simulator at the Oak Ridge National Laboratory and the surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment. The comparison of PNP demonstrated that the plant-specific measurement data are less than 20 percent different from the analytical prediction. The NRC staff determined that these constitute acceptable test facilities, as they are specifically referenced in RG 1.190 and are within the acceptable analytical prediction.

WCAP-15353, Supplement 2—NP, Revision 0, contains acceptable plant-specific benchmarking for PNP as it contains a database of Pressurized Water Reactor (PWR) dosimetry benchmarking. The PNP unit-specific geometry, a PWR RPV, is well-represented within the database. PWR-specific benchmarking documented in WCAP-15353, Supplement 2—NP, Revision 0, indicates that surveillance capsule fluence can be calculated within 20 percent of measured values, which is in accordance with RG 1.190. The NRC staff has determined, therefore, that these uncertainties are acceptable.

To summarize, the NRC staff has reviewed the neutron fluence methodology as documented in WCAP-15353, Supplement 2—NP, Revision 0 and finds that it follows RG 1.190 as described above. As documented, WCAP-15353, Revision 0, and WCAP-15353, Supplement 1—NP, Revision 0, have been previously reviewed and approved by the NRC staff. Therefore, the NRC staff concludes the neutron fluence is acceptable for use in the EMA analysis.

3.3.2 Reactor Vessel Material Metallurgy

<u>PNP Shell Plates D-3802-3 and D-3804-1</u>

The licensee stated in the EMA submittal that the RPV at PNP is constructed using SA-302B modified alloy steel plates and that this material has a minimum nickel content of 0.4 percent. The licensee has also stated that shell plates D-3802-3 and D-3804-1 have a high sulfur content (0.029 percent and 0.024 percent respectively).

The NRC staff notes that increasing sulfur content is detrimental to the toughness of alloy steel. This sulfur content is an important factor in analyzing the equivalent margins because RG 1.161 provides a model for high toughness RPV alloy steels, and sets 0.018 percent sulfur content as a line of demarcation between high toughness RPV alloy steels and low toughness RPV alloy steels. The RG 1.161 model for RPV base material was derived from data from SA-302B shell plates with low nickel and low sulfur. In comparison to the data which form the basis for the RG 1.161 model, the high sulfur in shell plates D-3802-3 and D-3804-1 acts to reduce toughness, but the high nickel counteracts to increase toughness. As discussed in Section 2.0 of this SE, RG 1.161 allows the high toughness model to be used for materials with sulfur contents higher than 0.018 percent, provided justification is given. Alternately, RG 1.161 allows high sulfur plates to be evaluated based on compact tension specimen data available in NUREG/CR-5265 with adjustments for specimen temperature and CVN value. The NRC staff's evaluation of PNP SA-302B modified alloy steel plates (D-3802-3 and D-3804-1) is contained in Section 3.3.4 of this SE.

<u>PNP Weld 9-112</u>

The licensee's description of weld 9-112 is found in WCAP-17651-NP, Revision 0, which was submitted as Attachment 5 in the November 12, 2014, license amendment request, as supplemented by information from WCAP-17341-NP, "Palisades Nuclear Power Plant Heatup and Cooldown Limit Curves for Normal Operation and Upper-Shelf Energy Evaluation," (ADAMS Accession No. ML110730083). Weld 9-112 was fabricated by the automatic submerged arc process using alloy steel weld wire, heat number 27204, with Linde 124 flux. The weld metal specimens in the surveillance capsules were fabricated from the same heat of weld wire but with a different flux (Linde 1092 flux), and therefore, do not represent weld 9-112. Therefore, the licensee used RG 1.99 Regulatory Position 1.2 (Charpy USE Surveillance Data Not Available) to calculate EOLE USE. The NRC staff's evaluation of the licensee's calculation of EOLE USE is discussed in Section 3.3.3 of this SE. The NRC staff's evaluation of Weld 9-112 is contained in Section 3.3.4 of this SE.

3.3.3 Staff Independent Verification of the Licensee's Determination of EOLE USE

As stated in Section 3.1, the licensee projected that plates D-3804-1 and D-3802-3, and weld 9-112 would fall below the EOLE USE threshold of 50 ft-lb. EOLE USEs for these three materials were not based on surveillance data, because none of these materials were included in the RPV surveillance capsules which were assembled to the requirements of ASTM E185-66 as documented in Section 4.5 of the PNP FSAR, Revision 31 (ADAMS Accession No. ML15226A016). At the time the PNP surveillance capsules were assembled, the E185-66 requirement was that a base material sample be selected based on the highest ductile-to-brittle transition temperature, not on the EOLE USE. Thus, intermediate shell plate D-3803-1, which had the highest initial ductile-to-brittle transition temperature, was selected to represent the reactor vessel base material in the PNP surveillance program.

Section III.B.1 of 10 CFR Part 50 Appendix H states, "The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased." Since early editions of ASTM E185 did not require that base material samples be selected based on the EOLE USE, some plants do not have the limiting USE materials in their surveillance capsules; this situation is not uncommon in the operating RPV fleet. For PNP, future capsule withdrawal and testing of the surveillance specimens will give plant-specific embrittlement information for only the RPV materials represented by the surveillance materials, but not for these three subject materials which do not have their specimens in the surveillance capsules.

<u>PNP Shell Plates D-3802-3 and D-3804-1</u>

The licensee stated in the EMA submittal that the initial USE for shell plates D-3804-1 (110 ft-lb) and D-3802-3 (91 ft-lb) was measured in the longitudinal direction. To estimate the initial USE for these shell plates in the transverse direction, the licensee reduced the longitudinal values by 35 percent in accordance with BTP 5-3. The EOLE USE for the shell plates was then calculated by applying a reduction of 33 percent to the shell plate D-3804-1 initial USE value and 19.5 percent to the shell plate D-3802-3 initial USE value. These reduction factors are based on Regulatory Position 1.2 of RG 1.99 and were based on the 1/4T fluence and copper values of these materials using Figure 2 of RG 1.99. The licensee's calculated projected EOLE USE for shell plate D-3804-1 is 48.2 ft-lb and for shell plate D-3802-3 is 47.5 ft-lb.

The NRC staff independently verified the licensee's projected EOLE USE by performing the calculations associated with applying the reduction factors described above, and confirmed the licensee's calculated projected EOLE USE for shell plate D-3804-1 and shell plate D-3802-3 is accurate. Therefore, the NRC staff concludes the projected EOLE USE for shell plates D-3804-1 and shell plate D-3802-3 is below 50 ft-lb. In addition, the license's use of Regulatory Position 1.2 of RG 1.99 was appropriate because there were no surveillance data for these plates. The NRC staff's evaluation of the EMA is in Section 3.3.4 of this SE.

PNP Weld 9-112

The initial USE for weld 9-112 was estimated by the licensee by using a generic initial value of 84 ft-lb. As documented in WCAP-17341-NP, Revision 0, the generic initial value of 84 ft-lb corresponds to the mean-minus-two-sigma value for Linde 124 flux welds. Starting with this generic initial value, the licensee then calculated the EOLE USE for weld 9-112 by applying a reduction of 41 percent based on RG 1.99 Regulatory Position 1.2. The licensee calculated projected EOLE USE for weld 9-112 is 49.6 ft-lb.

The NRC staff independently verified the licensee's projected EOLE USE by performing the calculations associated with applying the reduction factor described above, and confirmed the licensee's calculated projected EOLE USE for weld 9-112 is accurate. In addition, the license's use of Regulatory Position 1.2 of RG 1.99 was appropriate because there were no surveillance data for this weld. Therefore, the NRC staff concludes the projected EOLE USE for weld 9-112 is below 50 ft-lb. The NRC staff's evaluation of the EMA is in Section 3.3.4 of this SE.

Conclusion

The NRC staff finds that the licensee's use of Regulatory Position 1.2 of RG 1.99 was appropriate because there was no surveillance data for these materials. Additionally, the NRC staff independently verified the licensee's projected EOLE USE calculations, and concludes that for shell plate D-3804-1 and D-3802-3, and weld 9-112, the projected EOLE USE are less than 50 ft-lb and pursuant to Appendix G of 10 CFR 50, the licensee is required to submit an analysis to demonstrate that these lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G or Section XI of the ASME Code.

3.3.4 Staff Evaluation of the Margin of Safety Against Ductile Fracture

To provide reasonable assurance that the margin of safety between the reactor material toughness and the applied driving force to fracture produced by both normal operations including design basis transients (i.e., RG 1.161 Levels A and B) and emergency and faulted conditions (i.e., RG 1.161 Levels C and D) is sufficient to preclude ductile fracture, the NRC evaluated the safety margins provided by the licensee, as required by 10 CFR 50 Appendix G. As stated in Section 3.2.2 of this SE, RG 1.161 was developed to provide acceptance criteria for this type of evaluation.

The licensee has opted to use RG 1.161 to determine the equivalent margins of safety against ductile fracture for the three RPV materials (plates D-3802-3 and D-3804-1, and weld 9-112), which are projected to drop below 50 ft-lb by the EOLE. RG 1.161 provides separate materials property evaluation models for plates (see RG 1.161 Section 3.3) and for generic welds (see RG 1.161 Section 3.2).

Evaluation of Safety Margins of Plates D-3802-3 and D-3804-1

For the evaluation of plates D-3802-3 and D-3804-1, the licensee used the RG 1.161 base material model and the most limiting inputs from plates D-3802-3 and D-3804-1. This provides a conservative evaluation that bounds both plates. The staff evaluation of the licensee analysis of plates D-3802-3 and D-3804-1 is documented below.

High Toughness and High Sulfur Models of Base Metals

In RG 1.161, the NRC provides a fracture toughness model for high toughness plate materials intended for materials with less than 0.018 percent sulfur, but allows the high toughness model to be used for materials with higher sulfur content, provided justification is given. Since shell plates D-3802-3 and D-3804-1 have a high sulfur content (0.029 percent and 0.024 percent, respectively), the licensee must justify the use of the RG 1.161 high toughness model for base materials. In the submittal dated November 12, 2014, the licensee noted the scarcity of data for base materials with high sulfur content. The licensee discussed the V-50 plate that formed part of the data set for NUREG/CR-5265, which is referred to in RG 1.161. Plate V-50 had reported sulfur content in excess of 0.018 percent, very low toughness which is not representative of SA-302B modified material, and unusual size effects in J-R test specimens which are attributed to high inclusion content and a banded microstructure. The licensee concluded that V-50 plate is not representative of the PNP shell plates since the PNP shell plates have higher nickel content and higher toughness than the V-50 plate. The licensee also concluded that the RG 1.161 high toughness plate model can be justified for the analysis of shell plates D-3802-3 and D-3804-1, and that "the J-R curve test data from the V-50 plate data can be conservatively viewed as the worst possible case and can be compared to the J-applied values from this evaluation." The licensee analysis of base metal circumferential flaws using the high toughness model showed a safety factor of 2.87 for the Level A and B transients, and a safety factor of 5.8 for the Level C and D transients.

The NRC staff considered the justification given by the licensee in its October 21, 2013, submittal for applying the high toughness model of RG 1.161 to be insufficient to support the EMA because the licensee did not demonstrate that beneficial contribution of the nickel content would outweigh the detrimental contribution of the sulfur content and justify the use of the RG 1.161 high toughness model. Thus, the NRC staff issued a RAI dated May 13, 2014 (ADAMS Accession No. ML14133A684), requesting the licensee to demonstrate that when the high sulfur model of NUREG/CR-5265 is used, (1) the updated safety factors after adjusting for temperature will still be greater than the minimum required safety factor of 1.15 for crack extension, per RG 1.161, and (2) the crack stability criterion of RG 1.161 is also satisfied ($\partial J_{applied}/\partial a < \partial J_{material}/\partial a$ at $J_{applied} = J_{material}$).

The licensee responded in a letter dated June 12, 2014 (ADAMS Accession No. ML14163A662) which is included in Attachment 2 of the November 12, 2014, submittal, providing updated calculations using the J-R curves for base material circumferential flaws considering the V-50 plate data. The licensee's updated calculations using the high sulfur model of NUREG/CR-5265 showed that when considering the V-50 plate data, the margins on base material circumferential flaws are 1.6 for Levels A and B, 1.9 for Level C, and 1.5 for Level D, and, therefore, still meet the crack extension criterion of RG 1.161. The licensee also demonstrated for the transients evaluated that the postulated flaw is stable under ductile crack growth in accordance with the stability criterion of RG 1.161.

The NRC staff reviewed the licensee's RAI responses and concludes the licensee appropriately applied the RG 1.161 models for base materials (plates D-3802-3 and D-3804-1). In addition, the staff concludes that for base material circumferential flaws, the licensee's calculations exceed the RG 1.161 minimum safety margin of 1.15 for both the high sulfur and high toughness models.

Flaw Orientation of Base Metals

As stated in RG 1.161, both axial and circumferential flaws should be postulated for base materials. Thus, axial and circumferential flaw analysis should be performed for shell plates D-3802-3 and D-3804-1.

The licensee included in Attachment 2 of the submittal dated November 12, 2014, the RAI responses from the NRC review of the October 21, 2013, submittal. The submittal dated October 21, 2013, stated "[o]nly circumferential base metal flaws are considered in this analysis, because only the 'weak' orientation USE is projected to drop below 50 ft-lbs..." Therefore, in RAI-2 dated May 13, 2014, the NRC staff requested the licensee to demonstrate that assuming a circumferential flaw in the base material in the "weak" (or T-L) orientation is more limiting than assuming an axial flaw in the base material in the "strong" (or L-T) orientation.

By letter dated June 26, 2014 (ADAMS Accession No. ML14177A707), and as attached to the November 12, 2014, submittal in Attachment 2, the licensee responded with an analysis of base material axial flaws. The licensee noted that, for its analysis considering V-50 plate data, the lack of data in the "strong" orientation required a conversion between available "weak" orientation data and unavailable "strong" orientation data. For its analysis considering V-50 plate data, the licensee used a ratio of 65 percent between "weak" properties and "strong" properties to convert the $J_{material}$ values from the "weak" orientation used in circumferential flaw analysis to the "strong" orientation used in axial flaw analysis.

The licensee's response to RAI-2 included an analysis of the V-50 J-R curve data in NUREG/CR-5265. The licensee's analysis of base material axial flaws using the high toughness model showed a safety factor of 1.7 for the Level A and B transients, a safety factor of 3.4 for the Level C transient and a safety factor of 2.9 for the Level D transient. For the licensee's analysis of base material axial flaws considering V-50 plate data (high sulfur model), the safety factor was 1.4 for Levels A and B, 2.8 for Level C, and 2.8 for Level D. The licensee also demonstrated for the transients evaluated that the postulated flaw is stable under ductile crack growth in accordance with the stability criterion of RG 1.161.

The NRC staff reviewed the licensee's RAI responses and concludes the licensee appropriately applied the RG 1.161 models for base materials (plates D-3802-3 and D-3804-1). In addition, the staff concludes that for base material axial flaws, the licensee's calculations exceed the RG 1.161 minimum safety margin of 1.15 for both the high sulfur and high toughness models.

Evaluation of Safety Margins of Weld 9-112

For the evaluation of weld 9-112, the licensee used the generic weld model in RG 1.161. As stated in RG 1.161, the postulated flaw for the weld material should be oriented with its major axis along the axis of the weld. Since weld 9-112 is a circumferential weld, the NRC staff determined that the licensee's approach of considering only circumferential flaws in weld 9-112 in the EMA analysis is appropriate. The licensee evaluated the postulated weld flaws using the generic RPV weld model of RG 1.161. The licensee calculations showed that the safety factors on weld circumferential flaws are 2.49 for Levels A and B and 5.2 for Levels C and D. The licensee also demonstrated for the transients evaluated that the postulated flaw is stable under ductile crack growth in accordance with the stability criterion of RG 1.161.

The NRC staff reviewed the licensee's evaluation of weld 9-112 and concludes that the licensee appropriately applied the generic welds model in RG 1.161 and that the safety factors exceed the RG 1.161 minimum safety margin of 1.15.

Conclusion

With the licensee's analysis updated to consider base material axial flaws and the high sulfur model of NUREG/CR-5265, the base material axial flaw becomes limiting for the Level A and B transients with a safety factor of 1.4, and the base material circumferential flaw is limiting for the C and D transients with safety factors of 1.9 and 1.5, respectively. As described above, for all postulated flaws and transient levels evaluated in this EMA, the safety factor exceeded 1.15 for all three RPV materials projected to drop below 50 ft-lb by the EOLE, which is the minimum margin in accordance with the methodology described in RG 1.161 to demonstrate margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. For the base material postulated flaws, the licensee calculated safety margins using two different models which bound the high sulfur high nickel material used at PNP. The high toughness model of RG 1.161 provides the upper toughness bound for the base material, and the most conservative (worst) J-R data associated with the high sulfur model of NUREG/CR-5265 provide the lower toughness bound for the base material. For the weld metal postulated flaws, the licensee appropriately used the generic RPV welds model of RG 1.161. Therefore, the NRC staff concludes the licensee's EMA analysis is acceptable.

3.4 Evaluation of Potential Non-Conservative Positions in BTP 5-3

As stated in Section 3.3.3 of this SE, the staff has independently verified the licensee's projected EOLE USE using the guidance in BTP 5-3 and RG 1.99. However, a letter from AREVA Inc. dated January 30, 2014 (ADAMS Accession No. ML14038A265), to the NRC has called into question the conservative nature of a position in BTP 5-3 unrelated to USE. In response to this information, the NRC staff performed evaluations of all positions in BTP 5-3 and presented assessments of this information during public meetings held on June 4, 2014, and February 19, 2015[2].

The NRC staff is continuing its technical evaluation that was presented in February 19, 2015, as part of its process to determine if BTP 5-3 needs to be revised. The NRC staff notes that the potential non-conservatism in BTP 5-3 could impact the circumferential flaw analysis of the plates D-3802-3 and D-3804-1, which uses estimated transverse plate data. Given the potential for a non-conservative estimate of USE, the NRC performed an independent analysis using a reduction factor based on test data; this reduction factor is more conservative than the 65 percent reduction factor used in BTP 5-3 and appropriately bounds the experimental data. The resulting EMA analysis using this further reduced USE determined that the calculated safety

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[2] The June 4, 2014, and February 19, 2015 presentations of the NRC staff assessment of BTP 5-3 protocols can be found at ADAMS Accession Nos. ML14163A524 and ML15061A065, respectively.

factors for the high toughness and high sulfur models of circumferential flaws continue to meet the RG 1.161 minimum safety margin of 1.15 for all transient levels. Therefore, the NRC staff concluded that the licensee's calculated margins in the EMA are sufficient to cover any potential non-conservatism in BTP 5-3. As a result, the NRC staff determined there was no additional information required from the licensee.

3.5 Conclusion

Pursuant to 10 CFR 50 Appendix G, Section IV.A.1.a:

> Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This [equivalent margins] analysis must use the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2) at the time the analysis is submitted.

The NRC staff concludes that the licensee has demonstrated that the three materials: lower shell plate D-3804-1, intermediate-to-lower shell circumferential weld 9-112, and upper shell plate D-3802-3, for which the Charpy EOLE USE are projected to drop below 50 ft-lb have sufficient margins of safety against ductile fracture through the end of the unit's current operating license. This conclusion is based on the NRC's determination that the EMA was performed in accordance with ASME Code, Section XI, Appendix K and the guidance of RG 1.161. Further, the margins of safety exceeded the minimum margins specified in RG 1.161. Therefore, the NRC approves the licensee's EMA and authorizes the licensee to revise the FSAR, as described in the application dated November 12, 2014, as supplemented by letter dated January 28, 2015, to incorporate the EMA for satisfying the RPV Charpy USE requirements.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified on August 26, 2015, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, or any effluents that may be released offsite, and that there is no significant increase in individual, or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding

(80 FR 523, January 6, 2015). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Jenkins
M. Hardgrove

Date: November 23, 2015

November 23, 2015

Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT RE: REQUEST FOR APPROVAL OF 10 CFR PART 50 APPENDIX G EQUIVALENT MARGINS ANALYSIS (CAC NO. MF5163)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 258 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant. The amendment approves changes to the final safety analysis report (FSAR) in response to your application dated November 12, 2014, as supplemented by letter dated January 28, 2015.

The amendment approves the licensee's proposed revisions to information in the FSAR regarding the reactor pressure vessel (RPV) Charpy upper-shelf energy (USE) requirements of Title 10 of the *Code of Federal Regulations* Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements," IV.A.1. The change updates the analysis for satisfying the RPV Charpy USE requirements through the end of the renewed operating license.

A copy of our related safety evaluation is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Jennivine K. Rankin, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 258 to DPR-20
2. Safety Evaluation

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ADAMS Accession Nos. Amendment ML15106A682 **via email**

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