Attachment 1

Description and Assessment of Requested Change

1.0 SUMMARY DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) approval of a proposed license amendment for Palisades Nuclear Plant (PNP) for an equivalent margins analysis (EMA) completed in accordance with 10 CFR 50 Appendix G, Section IV, “Fracture Toughness Requirements.”

2.0 DETAILED DESCRIPTION

In the PNP license renewal application (Reference 1), Nuclear Management Company (NMC), the former license holder for PNP, committed to submit an EMA for NRC approval at least three years before any reactor vessel beltline material Charpy upper-shelf energy (USE) decreases to less than 50 ft-lb, in accordance with 10 CFR 50 Appendix G, Section IV.

ENO submitted the required EMA in Reference 2 under 10 CFR 50.4, “Written communications,” as required by 10 CFR 50 Appendix G. In Reference 3, ENO received a request for additional information (RAI) concerning the EMA submittal. The ENO response to RAI questions 1, 3, 4, 5, and 6 was provided in in Reference 4. The ENO response to RAI question 2 was provided in Reference 5. The ENO responses to the six RAI questions are repeated in Attachment 2.

During a conference call with ENO on October 21, 2014, the NRC requested that the EMA be submitted under 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” rather than under 10 CFR 50.4.

Pursuant to 10 CFR 50.90, ENO hereby submits an amendment application for the PNP operating license. The proposed amendment requests approval of an EMA completed in accordance with 10 CFR 50 Appendix G, Section IV.

3.0 TECHNICAL EVALUATION

In the PNP license renewal application (Reference 1), NMC, the former license holder for PNP, committed to submit an EMA for NRC approval at least three years before any reactor vessel beltline material Charpy upper-shelf energy (USE) decreases to less than 50 ft-lb, in accordance with 10 CFR 50 Appendix G, Section IV.

The EMA is to demonstrate that material predicted to possess Charpy USE values 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) Boiler and Pressure Vessel Code.

As documented in the PNP license amendment request for primary coolant system pressure-temperature limits (Reference 6), a plate material and a weld material in the PNP reactor vessel traditional beltline region are predicted to drop below the Appendix G 50 ft-lb screening criterion prior to the PNP end-of-license-extension (EOLE). The lower shell (LS) plate material, D-3804-1, is predicted to drop below the screening criterion in December 2016 and the intermediate shell (IS) to LS circumferential weld material, 9-112 (heat no. 27204), is predicted to drop below the criterion in November 2027.

As documented in WCAP-17403-NP (Attachment 3), the Charpy USE of an upper shell (US) plate material, D-3802-3, in the reactor vessel extended beltline region is predicted to remain above the 50 ft-lb Appendix G screening criterion at EOLE when considering an initial Charpy USE value based on a curve-fit of the available Charpy V-Notch data. However, this material is predicted to drop below the Appendix G screening criterion, to 47.5 ft-lb at EOLE, when considering an initial Charpy USE value based only on the available 95% shear Charpy V-Notch data for this material.

The remaining beltline and extended beltline materials in the reactor vessel are projected to maintain above the Charpy USE screening criterion of 50 ft-lb at EOLE.

The results of the fluence calculations for the extended beltline region materials of the PNP reactor vessel are provided in Table E-2 of WCAP-15353 – Supplement 2 – NP (Attachment 4). Supplement 2 was generated to address the neutron fluence experienced by materials located in the extended beltline regions above and below the reactor core that were not included in either Revision 0 of WCAP-15353 or in Supplement 1 of that report.

In accordance with 10 CFR 50 Appendix G, this letter transmits for NRC review and approval the EMA report WCAP-17651-NP (Attachment 5) for the two traditional beltline and one extended beltline reactor vessel materials discussed above. Extended beltline US plate material D-3802-3 was analyzed due to the possibility that it may fall below the 50 ft-lb limit if future operation includes higher flux levels, longer operating cycles, or changes to the reactor internals. The analysis of the three materials used the equivalent margins methodology specified in ASME Code Section XI, Division 1, Appendix K, “Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels,” and concluded that all three of the reactor vessel materials are acceptable.

The work described herein was performed in accordance with industry and NRC accepted practices.

The extended beltline regions of the reactor pressure vessel with EOLE neutron fluence (E > 1.0 MeV) greater than 1.0 E+17 n/cm^2 have been included in the extended beltline evaluation in WCAP-17403-NP (Attachment 3). Figure 1-2 of the evaluation illustrates the boundary of the extended beltline region with neutron fluence in excess of 1.0 E+17 n/cm^2. It is noted that the neutron fluence for the inlet and outlet nozzles remain below 1.0 E+17 n/cm^2 at EOLE. The evaluation of these regions concluded that the materials are predicted to remain below the pressurized thermal shock screening criteria and the traditional beltline materials remain limiting. Also, all adjusted reference temperature values are predicted to remain below those contained in the analysis of record, so the pressure-temperature limit curves and low temperature overpressure protection (LTOP) setpoint limit curve continue to be governed by the traditional beltline materials.

The fluence evaluation in WCAP-15353 – Supplement 2 – NP (Attachment 4) used to assess the material properties is compliant with Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” Previous PNP reactor pressure vessel neutron fluence evaluation submittals have been reviewed and approved by the NRC as being consistent with the requirements of Regulatory Guide 1.190. The methodology used for the WCAP-15353 fluence evaluation is detailed in

- WCAP-14040-NP-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” and

- WCAP-16083-NP-A, Revision 0, “Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,”

which have been previously reviewed and approved by the NRC.

The EMA in WCAP-17651-NP is based upon ASME Code Section XI, Division 1, Appendix K, but has been supplemented with additional criteria specified in Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb,” for material fracture toughness, transients, and fracture toughness resistance (J-R) model restrictions on sulfur content.

The USE values in the EMA have been matched to the proper orientation of the plate material. For axial flaws, the USE value for the lateral transverse “strong” orientation in the vessel wall has been used. Similarly, for circumferential flaws, the USE value for the transverse-lateral “weak” orientation has been used. In addition, the initial longitudinal USE values have been reduced to 65 percent per NUREG-0800 Branch Technical Position 5-3, “Fracture Toughness Requirements,” to approximate the transverse “weak” direction.

Per Regulatory Guide 1.161, three additional cooldown transients at 100 °F/hr, 400 °F/hr, and 600 °F/hr have been considered in the EMA.

As discussed in the EMA, extended beltline plate material D-3802-3 and traditional beltline plate material D-3804-1 have sulfur content in excess of the J-R model 0.018 weight percent limitation. Additional analysis has been conducted using available information for the V-50 plate in NUREG/CR-5265, “Size Effects on J-R Curves for A 302-B Plate,” to demonstrate that the PNP reactor vessel high sulfur plate materials remains below the measured very conservative lower bound V-50 A-302 B plate J-R data.

Additional technical evaluation information is provided in Attachment 2.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

An assessment of the proposed changes concluded that there are no exceptions to any of the following regulations. Therefore, ENO would remain in compliance with the following regulations and guidance:

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality Standards and Records," requires the structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

GDC 31, "Fracture prevention of the reactor coolant pressure boundary," requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

GDC 32, "Inspection of the reactor coolant pressure boundary," requires components that are part of the reactor coolant pressure boundary be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," requires that all lightwater reactors 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 H, "Reactor Vessel Material Surveillance Program Requirements," ensures that changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment are monitored. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

10 CFR 50, Appendix G, Section IV.A.1, “Reactor Vessel Charpy Upper-Shelf Energy Requirements,” requires submission of an EMA for approval at least three years before any reactor vessel beltline material Charpy USE decreases to less than 50 ft-lb.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, describes methods for determining reactor pressure vessel fluence.

Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 Ft-Lb,” June 1995, describes methods for demonstrating that the margins of safety against ductile fracture are equivalent to those in Appendix G of the ASME Code.

4.2 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   This amendment request is for approval of an equivalent margins analysis (EMA) in accordance in accordance with 10 CFR 50 Appendix G, Section IV, “Fracture Toughness Requirements.” The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy Upper Shelf Energy (USE) values 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) Boiler and Pressure Vessel Code.

The EMA does not involve a significant increase in the probability or consequences of an accident, and does not result in physical alteration of a plant structure, system or component (SSC) or installation of new or different types of equipment. The EMA does not affect plant operation or any design function. The EMA verifies the capability of a SCC to perform a design function. Further, the EMA does not significantly affect the probability of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR), or cause a change to any of the dose analyses associated with the UFSAR accidents because accident mitigation functions would remain unchanged.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The amendment request is for approval of an EMA in accordance in accordance with 10 CFR 50 Appendix G, Section IV. The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy USE values less than 50 ft-lb will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code. The EMA does not change the design function, operation, or integrity of the reactor vessel, and does not challenge the performance or integrity of any safety-related systems. No physical plant alterations are made as a result of the proposed change. The EMA will not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The amendment request is for approval of an EMA in accordance in accordance with 10 CFR 50 Appendix G, Section IV. The EMA is to demonstrate that reactor vessel beltline material predicted to possess Charpy USE values less than 50 ft-lb will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code. As such, there is no significant reduction in the margin of safety as a result of the

EMA. No design bases or safety limits are exceeded or altered due to the EMA. The margin of safety associated with the acceptance criteria of accidents previously evaluated in the UFSAR is unchanged. The proposed change has no effect on the availability, operability, or performance of the safety-related systems and components.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Palisades Nuclear Plant, *Application for Renewed Operating License*, dated March 22, 2005 (ADAMS Accession No. ML050940446).
2. Entergy Nuclear Operations, Inc. letter PNP 2013-028, *Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margins Analysis,* dated October 21, 2013 (ADAMS Accession No. ML13295A448).
3. NRC email to Entergy Nuclear Operations, Inc., *Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis - MF 2962,* dated May 13, 2014 (ADAMS Accession No. ML14133A684).
4. Entergy Nuclear Operations, Inc. letter PNP 2014-054, *Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50*

Appendix G Equivalent Margin Analysis – MF 2962, dated June 12, 2014 (ADAMS Accession No. ML14163A662).

5. Entergy Nuclear Operations, Inc. letter PNP 2014-066, *Supplemental Response to NRC Request for Additional Information - Palisades Nuclear Plant 10 CFR 50 Appendix G Equivalent Margin Analysis – MF 2962*, dated June 26, 2014 (ADAMS Accession No. ML14177A707).
6. Palisades Nuclear Plant, *License Amendment Request for Primary Coolant System Pressure-Temperature Limits*, dated March 7, 2011 (ADAMS Accession No. ML110730082)