Entergy

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PNP 2014-049

July 29, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: License Amendment Request to Implement 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

REFERENCES:

1. Palisades Nuclear Plant, "Application for Renewed Operating License," dated March 22, 2005 (ADAMS Accession Number ML050940429).

2. Consumers Energy letter to the NRC, "Palisades Reactor Vessel Neutron Fluence Reevaluation," dated February 21, 2000 (ADAMS Accession Number ML003686516).

3. Entergy Nuclear Operations, Inc. letter, *Updated Palisades Reactor Vessel Pressurized Thermal Shock Evaluation*, dated December 20, 2010 (ADAMS Accession No. ML110060692).

4. NRC letter, *Updated Reactor Pressure Vessel Pressurized Thermal Shock Evaluation for Palisades Nuclear Plant (TAC No. ME5263)*, dated December 7, 2011 (ADAMS Accession No. ML112870050).

5. Entergy Nuclear Operations, Inc. letter PNP 2013-046, *Updated Palisades Nuclear Plant Reactor Vessel Fluence Evaluation*, dated June 25, 2013 (ADAMS Accession No. ML13176A412).

6. NRC letter, *Palisades Nuclear Plant – Updated Reactor Vessel Fluence Evaluation Supporting a Revised Pressurized Thermal Shock Screening Criteria Limit (TAC No. MF2326)*, dated December 18, 2013 (ADAMS Accession No. ML13346A136).

A001
NRR

Dear Sir or Madam:

The Palisades Nuclear Plant (PNP) license renewal application (Reference 1) stated that the PNP reactor vessel (RV) was projected to reach the 10 CFR 50.61 pressurized thermal shock (PTS) screening criterion limit prior to the end of the license renewal period. The projection was based on a previous analysis (Reference 2) that determined the RT_{PTS} values for the PNP vessel beltline materials using RV fluence and materials information available at that time. The limiting RV welds, which are the beltline axial welds fabricated with weld wire heat no. W5214, were projected to reach the 10 CFR 50.61 PTS screening criterion limit in 2014.

In Reference 3, Entergy Nuclear Operations, Inc. (ENO) submitted an updated RV 10 CFR 50.61 PTS evaluation for PNP. The submittal included an updated RV fluence evaluation, WCAP-15353 - Supplement 1 - NP, Revision 0, "Palisades Reactor Pressure Vessel Fluence Evaluation," that calculated fluence based on actual plant operation through Cycle 20 and the finalized design for Cycle 21, with future fluence based on planned future operations. Using data from the fluence calculation, the PTS evaluation concluded that the PNP RV would not reach the 10 CFR 50.61 PTS screening criteria limit until April 2017.

In Reference 4, the Nuclear Regulatory Commission (NRC) concluded that the updated PNP PTS evaluation was in accordance with 10 CFR 50.61 and that the PTS screening criteria limit would not be reached until April 2017.

In Reference 5, ENO submitted an updated fluence calculation that reflected actual plant operation through Cycle 22, and future fluence based on planned operations through Cycle 26. The updated fluence calculation would extend the date that PNP RV is projected to reach the 10 CFR 50.61 PTS screening criteria to August 2017.

In Reference 6, the NRC reviewed the updated fluence calculation and determined that the revision to August 2017 for PNP to reach the PTS screening criteria was acceptable. Regulation 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," provides an alternative for PTS management. It requires determining projected reference temperature (RT_{MAX-X}) values for each RV beltline material, performing an examination and assessment of flaws in the RV beltline, and comparing projected RT_{MAX-X} values for RV beltline materials to the PTS screening criteria in Table 1 of 10 CFR 50.61a.

Regulation 10 CFR 50.61a(c) requires that an application for implementation of 10 CFR 50.61a be submitted under 10 CFR 50.90 at least three years before the RV is projected to exceed the PTS screening criteria under 10 CFR 50.61.

Pursuant to 10 CFR 50.61a(c) and 10 CFR 50.90, ENO hereby submits an amendment application for the PNP operating license. The proposed amendment would authorize the implementation of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," in lieu of 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

The enclosure contains the PNP 10 CFR 50.61a PTS evaluation, “Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades,” Revision 1. In the evaluation, RT_{MAX-X} values were generated for the beltline and extended beltline region materials of the PNP RV for fluence values at the end-of-license extension (EOLE) (i.e., 42.1 effective full power years). The RV extended beltline for PNP is defined as the region of materials that meet or exceed a neutron fluence exposure of 1.0×10^{17} n/cm^2 (E > 1.0 MeV). The RT_{MAX-X} values were calculated using RV beltline and extended beltline material copper, nickel, phosphorus, and manganese content, reference temperature for an unirradiated reactor vessel material (RT_{NDT}), projected EOLE neutron fluence values, and time-weighted averaged reactor cold-leg temperature.

The evaluation concludes that the PNP RV meets the alternate PTS rule acceptance criteria. All of the beltline and extended beltline region materials in the RV have end-of-license extension RT_{MAX-X} values below the screening criteria values. After conducting surveillance data statistical tests, it was determined that the surveillance data satisfied the alternate PTS rule requirements. Lastly, a review of the latest RV inservice inspection report for the PNP RV showed that the flaw density and size distribution is acceptable per the alternate PTS rule requirements.

To allow for normal NRC processing, ENO requests approval of the proposed license amendment by July 29, 2015. Also, an implementation period of 120 days following the effective date of the amendment is requested.

This submittal contains no proprietary information.

This submittal makes no new commitments or revisions to previous commitments.

In accordance with 10 CFR 50.91(b), a copy of this application, with attachments, is being provided to the designated State of Michigan official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 29, 2014.

Sincerely,

ajv/jse

Attachments:
1. Description and Assessment of Requested Change
2. Mark-up of Operating License Page
3. Renewed Operating License Page Change Instructions and Revised Operating License Page

Enclosure: Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

Attachment 1

Description and Assessment of Requested Change

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, “Amendment of License or Construction Permit at Request of Holder,” Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) approval of a proposed license amendment for Palisades Nuclear Plant (PNP) for implementation of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events."

PNP currently complies with 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," which establishes screening criteria below which the potential for a reactor vessel to fail due to a pressurized thermal shock (PTS) event is deemed to be acceptably low. The 10 CFR 50.61 screening criteria define a limiting level of embrittlement beyond which plant operation cannot continue without further evaluation. As described in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (Reference 1), the screening criteria in the PTS rule is overly conservative and the risk of through-wall cracking due to a PTS event is much lower than previously estimated. As such, the specified screening limits and associated compensatory actions may impose an unnecessary burden on licensees whose pressurized water reactor (PWR) vessel is projected to exceed the PTS rule screening criteria.

The alternate PTS rule, which was included in the Federal Register with an effective date of February 3, 2010, provides fracture toughness requirements for protection against PTS events for PWR pressure vessels that are less burdensome than the requirements of the PTS rule.

The PNP RV is projected to exceed the screening criteria of the PTS rule in August 2017. Compliance with the alternate PTS rule would maintain adequate protection to public health and safety.

2.0 DETAILED DESCRIPTION

ENO requests NRC approval of a proposed license amendment for PNP for implementation of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," in lieu of the requirements of 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

The amendment would replace the existing license condition in PNP Renewed Facility Operating License section 2.C (8). The existing license condition states the following:

> "Upon implementation of Amendment 237, within one year of completing each of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Category B-A and B-D reactor vessel weld inspections, submit information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) to the NRC."

This existing license condition was added to the PNP Renewed Facility Operating License under Amendment Number 237, which supported a proposed change to the PNP in-service inspection (ISI) program (Reference 2). The change to the ISI program required that the license condition be added to the PNP operating license since 10 CFR 50.61a had not been implemented at PNP at that time.

This license condition would be replaced with a statement that the amendment authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61 (see Attachments 2 and 3).

3.0 BACKGROUND

During plant operation, the walls of reactor vessels (RVs) are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the core area. If an embrittled RV had a flaw of critical size and certain severe system transients were to occur, the flaw could propagate through the vessel, resulting in a through-wall crack. The severe transients of concern are known as pressurized thermal shock events. PTS events in PWRs are caused by severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

As summarized in NUREG-1806, in the early 1980s, the nuclear industry and the NRC staff performed a number of investigations to assess the risk of vessel failure posed by PTS and to establish the operational limits needed to ensure that the likelihood of RV failures caused by PTS transients is maintained at an acceptably low level. These efforts led to the development of the PTS rule. The nil ductility (fracture toughness) transition reference temperature (RT_{NDT}) of the reactor vessel material increases as a result of irradiation throughout the operational life of the vessel. The PTS rule establishes screening criteria (or maximum values of RT_{NDT} permitted during the operating life of the plant) of 270 degrees Fahrenheit (°F) for axial welds, plates, and forgings, and 300 °F for circumferential welds. The reference temperature value RT_{NDT} evaluated for the end-of-life (EOL) fluence for each of the vessel beltline materials, using the procedures in paragraph (c) of the PTS rule, is referred to as RT_{PTS}.

The PNP license renewal application (Reference 3) stated that the RT_{PTS} value for the PNP limiting RV welds was projected to reach the 10 CFR 50.61 PTS screening criterion limit prior to the end of the license renewal period. The projection was based on a previous analysis (Reference 4) that determined the RT_{PTS} values for the PNP vessel beltline materials using RV fluence and materials information available at that time. The limiting welds, which are the beltline axial welds fabricated with weld wire heat no. W5214, were projected to reach the PTS screening criterion limit in 2014.

Subsequently, new information became available that affected the date when the PTS screening criterion limit would be reached, and the PTS screening date was re-evaluated. The re-evaluation, which was documented in Structural Integrity Associates, Inc. Report No. 1000915.401, "Revised Pressurized Thermal Shock Evaluation for the Palisades Reactor Pressure Vessel," concluded that the PTS screening criterion would be reached in April 2017 (Reference 5). NRC review concluded that the updated PNP PTS evaluation was in accordance with 10 CFR 50.61 and that the PTS screening criteria limit would not be reached until April 2017 (Reference 5).

In 2013, ENO re-calculated PNP reactor vessel fluence, based on actual reactor operation through fuel Cycle 22, and expected fluence based on projected operations through Cycle 26. This evaluation was documented in "Palisades Reactor Pressure Vessel Fluence Evaluation," WCAP-15353 – Supplement 3 – NP, Revision 0, which was submitted in Reference 7. The evaluation concluded that the PNP RV limiting welds would not reach the 10 CFR 50.61 PTS screening criteria limit until August 2017. The NRC reviewed the calculation and determined that the revised date upon which the PTS screening criteria would be reached was acceptable (Reference 8).

In 1999, the NRC undertook a project to develop a technical basis to support a risk-informed alternative to the existing PTS rule. Realistic input values and models and an explicit treatment of uncertainties were used to develop the alternate PTS rule, which was approved by the NRC and included in the Federal Register with an effective date of February 3, 2010. In order to implement the alternate PTS rule, a licensee must submit a request for approval in the form of an application for a license amendment request in accordance with 10 CFR 50.90 and include documentation required by alternate PTS rule paragraphs (c)(1), (2), and (3).

4.0 TECHNICAL EVALUATION

The PNP alternate PTS rule evaluation is documented in the enclosed Westinghouse report WCAP-17628-NP, Revision 1, "Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades," dated June 2014.

Section 1 of the alternate PTS rule evaluation report is introductory. Section 2 discusses the alternate PTS rule and its requirements. Section 3 provides the methodology for calculating RT_{MAX-X} and performing the examination and flaw assessment required per the alternate PTS rule. Sections 4 through 7 provide inputs necessary to conduct the alternate PTS rule evaluations described in Section 3. Specifically, these sections provide the material properties, neutron fluence values, surveillance capsule analysis results, and inservice inspection data of the RV beltline and extended beltline materials. The results of the RT_{MAX-X} calculations and flaw assessment are presented in Section 8. The conclusion and references for the PTS evaluation follow in Sections 9 and 10, respectively.

Paragraph (a) of the alternate PTS rule defines the reference temperature RT_{MAX-X}. The reference temperature RT_{MAX-X} means any or all of the reactor vessel material properties that characterize the resistance to fracture initiation from flaws found along axial weld fusion lines (RT_{MAX-AW}), in plates (in regions not associated with welds) (RT_{MAX-PL}), in forgings (in regions not associated with welds) (RT_{MAX-FO}), along circumferential weld fusion lines (RT_{MAX-CW}), or the sum of RT_{MAX-AW} and RT_{MAX-PL}.

The alternate PTS rule primary requirements consist of the following, which are addressed in the enclosed evaluation report as applicable:

- Each licensee shall have projected values of RT_{MAX-X} for each reactor vessel beltline material for the end-of-license (EOL) fluence of the material. The assessment of RT_{MAX-X} values must use the calculation procedures described in Section 3.1 of the enclosed evaluation report. The assessment must specify the bases for the projected value of RT_{MAX-X} for each reactor vessel beltline material, including the assumptions regarding future plant operation (e.g., core loading patterns, projected capacity factors); the copper, phosphorus, manganese, and nickel contents; the reactor cold leg temperature; and the neutron flux and fluence values used in the calculation for each beltline material.

- Each licensee shall evaluate the results from a plant-specific or integrated surveillance program if the surveillance data satisfy the criteria described in paragraphs (f)(6)(i)(A) and (f)(6)(i)(B) of 10 CFR 50.61a.

- Each licensee shall perform an examination and an assessment of flaws in the reactor vessel beltline as described in Section 3.3 of the enclosed report. The licensee shall verify that the requirements described in Section 3.3 have been met.

- Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria in Table 3-2 of this report, for the purpose of evaluating a reactor vessel's susceptibility to fracture due to a PTS event.

- If any of the projected RT_{MAX-X} values are greater than the PTS screening criteria in Table 3-2, then the licensee may propose the compensatory actions or plant-specific analyses as required in paragraphs (d)(3) through (d)(7) of

10 CFR 50.61a, as applicable, to justify operation beyond the PTS screening criteria in Table 3-2. The licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criteria. If this analysis indicates that no reasonably practicable flux reduction program will prevent the RT_{MAX-X} value for one or more of the reactor vessel beltline materials from exceeding the PTS screening criteria, then the licensee shall perform a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent the potential for an unacceptably high probability of failure of the reactor vessel as a result of postulated PTS events. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results and plant surveillance data, and may use probabilistic fracture mechanics techniques.

Two alternate PTS rule subsequent requirements consist of the following:

- Whenever there is a significant change in projected values of RT_{MAX-X}, so that the previous value, the current value, or both values, exceed the screening criteria before the expiration of the plant operating license; or upon the licensee's request for a change in the expiration date for operation of the facility; a re-assessment of RT_{MAX-X} values must be conducted. If the surveillance data used to perform the re-assessment of RT_{MAX-X} values meet the requirements discussed in alternate PTS rule paragraphs (f)(6)(v) or (f)(6)(vi), the data must be analyzed in accordance with the alternate PTS rule and the RT_{MAX-X} values must be recalculated and resubmitted for approval.

- The licensee shall verify that the requirements of alternate PTS rule paragraphs (e), (e)(1), (e)(2), and (e)(3) have been met. The licensee must submit, within 120 days after completing a volumetric examination of reactor vessel beltline materials as required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, the adjustments made to the volumetric test data to account for NDE-related uncertainties as described in paragraph (e)(1) and all information required by paragraph (e)(1)(iii) for review and approval. If a licensee is required to implement paragraphs (e)(4), (e)(5), and (e)(6) of the alternate PTS rule, the information required in these paragraphs must be submitted within one year after completing a volumetric examination of reactor vessel materials as required by ASME Code, Section XI.

In the enclosed evaluation report, RT_{MAX-X} values were generated for the beltline and extended beltline region materials of the PNP RV for fluence values at the end-of-license extension (EOLE) (i.e., 42.1 effective full power years). These values were calculated using RV beltline and extended beltline material copper, nickel, phosphorus, and manganese content, unirradiated RT_{NDT}, projected EOLE neutron fluence values, and time-weighted averaged reactor cold-leg temperature.

The evaluation report concludes the following:

1. The RV beltline and extended beltline materials have EOLE, 42.1 effective full-power years, RT_{MAX-X} values below the alternate PTS Rule screening criteria;

2. The surveillance data for the vessel passed all of the surveillance data statistical tests for each material; and

3. The RV beltline and extended beltline weld flaw density and size distribution are acceptable based on the latest Palisades vessel inservice inspection results from an ASME Section XI, Appendix VIII qualified examination.

5.0 REGULATORY EVALUATION

5.1 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.

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

5.2 <u>No Significant Hazards Consideration</u>

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 would allow implementation of the 10 CFR 50.61a alternate pressurized thermal shock (PTS) rule in lieu of the 10 CFR 50.61 PTS rule, and would not involve a significant increase in the probability or consequences of an accident. Application of 10 CFR 50.61a in lieu of 10 CFR 50.61 would not result in physical alteration of a plant structure, system or component, or installation of new or different types of equipment. Further, application of 10 CFR 50.61a would 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. Use of 10 CFR 50.61a would change how fracture toughness of the reactor vessel is assessed and does not affect reactor vessel neutron radiation fluence. As such, implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61 would not increase the likelihood of a malfunction.

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 would allow implementation of the 10 CFR 50.61a alternate PTS rule in lieu of 10 CFR 50.61. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. No physical plant alterations are made as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system.

   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 would authorize implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61. Regulation 10 CFR 50.61a would maintain the same functional requirements for the facility as 10 CFR 50.61. It establishes screening criteria that limit levels of embrittlement beyond which operation cannot continue without further plant-specific evaluation or modifications. Sufficient safety margins are maintained to ensure that any potential increases in core damage frequency and large early release frequency resulting from implementation of 10 CFR 50.61a are negligible. As such, there would be no significant reduction in the margin of safety as a result of use of the alternate PTS rule. The margin of safety associated with the acceptance criteria of accidents previously evaluated in the UFSAR is unchanged. The proposed change would have 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.

5.3 <u>Conclusions</u>

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.

6.0 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.

7.0 REFERENCES

1. NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," August 2007 (ADAMS Accession Numbers ML072830076 and ML07282069).

2. NRC letter, "Palisades Plant – Issuance of Amendment RE: Change to In-Service Inspection Interval (TAC No. MD9266)," February 11, 2009 (ADAMS Accession Number ML090220442)

3. Palisades Nuclear Plant, "Application for Renewed Operating License," dated March 22, 2005 (ADAMS Accession Number ML050940429).

4. Consumers Energy letter to the NRC, "Palisades Reactor Vessel Neutron Fluence Reevaluation," dated February 21, 2000, (ADAMS Accession Number ML003686516).

5. Entergy Nuclear Operations, Inc. letter, "Updated Palisades Reactor Vessel Pressurized Thermal Shock Evaluation," dated December 20, 2010 (ADAMS Accession No. ML110060692).

6. NRC letter, "Updated Reactor Pressure Vessel Pressurized Thermal Shock Evaluation for Palisades Nuclear Plant (TAC No. ME5263)," dated December 7, 2011 (ADAMS Accession No. ML112870050).

8. ENO letter, PNP 2013-046, "Palisades Reactor Pressure Vessel Fluence Evaluation," dated June 25, 2013 (ADAMS Accession Number ML13176A412).

9. NRC letter, "Palisades Nuclear Plant – Updated Reactor Vessel Fluence Evaluation Supporting a Revised Pressurized Thermal Shock Screening Criteria Limit (TAC Number MF2326)," dated December 18, 2013 (ADAMS Accession Number ML 13346A136).

Attachment 2

Mark-up of Operating License Page

(additions are highlighted and deletions are strikethrough)

One Page Follows

(8) ~~Upon implementation of Amendment 237, within one year of completing each of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Category B-A and B-D reactor vessel weld inspections, submit information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275 prior to issuance of the final 10 CFR 50.61a) to the NRC.~~

Amendment xxx authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.

Attachment 3

Renewed Operating License Page Change Instructions and Revised Operating License Page

Two Pages Follow

Page Change Instructions

ATTACHMENT TO LICENSE AMENDMENT NO. xxx

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following page of Renewed Facility Operating License, and replace with the attached revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

REMOVE	INSERT
Page 5b	Page 5b

(8) Amendment xxx authorizes the implementation of 10 CFR 50.61a in lieu of 10 CFR 50.61.