Entergy

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PNP 2017-068

December 1, 2017

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

SUBJECT: Proposed Alternative - Relief Request Number RR 5-6, Alternative to the Reexamination Frequency for a Relevant Condition - Foreign Material in Reactor Vessel

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

Dear Sir or Madam:

Pursuant to Title 10 of the Code of Federal Regulations (CFR) 50.55a(z)(2), *Hardship without a compensating increase in quality and safety*, Entergy Nuclear Operations, Inc. (ENO) hereby requests Nuclear Regulatory Commission (NRC) authorization for the Palisades Nuclear Plant (PNP) of proposed alternative, relief request number RR 5-6, *Alternative to the Reexamination Frequency for Relevant Condition – Foreign Material in Reactor Vessel.*

Relief request RR 5-6 proposes an alternative to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, Subarticle IWB-2420, *Successive Inspections, subparagraph (b)*, requirements. The information provided in the attached request demonstrates that this code requirement results in hardship without a compensating increase in quality and safety.

The fifth ten-year inservice inspection (ISI) interval complies with the ASME BPV Code, Section XI, Division 1, 2007 Edition through the 2008 Addenda, and is the ASME Code of record applicable to the proposed alternative relief request RR 5-6.

The duration of the proposed alternative is for the remainder of the current fifth ten-year ISI interval.

ENO requests NRC approval by May 31, 2018 to support preparations for a planned fall 2018 refueling outage.

<u>Summary of Commitments</u>

This letter contains no new commitments and no revised commitments.

Sincerely,

JAH/jpm

Attachment: 10 CFR 50.55a, Relief Request Number RR 5-6, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2), Alternative to the Reexamination Frequency for a Relevant Condition – Foreign Material in Reactor Vessel

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

PNP 2017-068

Attachment

10 CFR 50.55a

Relief Request Number RR 5-6

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(2)

Alternative to the Reexamination Frequency for a Relevant Condition – Foreign Material in Reactor Vessel

7 pages follow

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

Code Class:	ASME Code Class 1
Component Numbers:	N-50, Reactor Vessel
Code References:	ASME Boiler and Pressure Vessel (BPV) Code, Section XI, Division 1, 2007 Edition through the 2008 Addenda, Paragraph IWB-2420, *Successive Inspections*, subparagraph (b) ASME Section XI, Table IWB-2500-1, *Examination Categories*
Examination Category:	B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, B-N-3, Removable Core Support Structures
Item Number(s):	B13.10, Reactor Vessel, Vessel Interior (B-N-1)
Unit/Inspection Interval:	Palisades Nuclear Plant (PNP) / Fifth Ten-Year Inservice inspection (ISI) Interval December 13, 2015 – December 12, 2025

2. Applicable Code Edition and Addenda

ASME BPV Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Division 1, 2007 Edition through the 2008 Addenda.

3. Applicable ASME Code Requirements

IWB-2420, *Successive Inspections*, subparagraph (b) requires that for components accepted by analytical evaluation in accordance with subparagraph IWB-3142.4, *Acceptance by Analytical Evaluation,* the areas containing relevant conditions be reexamined during the next three inspection periods.

Entergy Nuclear Operations, Inc. (ENO) identified in accordance with IWB-3520.2, *Visual Examination*, *VT-3*, subparagraph (c), a PNP primary coolant pump impeller piece (foreign material) stuck in the reactor vessel during the third period of the fourth inservice inspection (ISI) interval and accepted this condition for continued service by analytical evaluation in accordance with IWB-3142.4. Since ENO identified the relevant indication in the third period of the fourth ISI interval, the IWB-2420(b) successive examinations are required to be performed in the first period of the fifth ISI interval (ending December 12, 2018), the second period of the fifth ISI interval (ending December 12, 2022), and the third period of the fifth ISI interval (ending December 12, 2025).

4. Reason for Request

Pursuant to 10 CFR 50.55a, *Codes and standards*, paragraph (z), *Alternatives to codes and standards requirements, (2), Hardship without a compensating increase in quality and safety*, relief is requested from the reexamination requirements of the ASME BPV Code Section XI. The basis for relief is that the ASME BPV Code Section XI requirements present an undue hardship without a compensating increase in the level of quality and safety.

During PNP 2014 refueling outage, 1R23, a piece of the primary coolant pump P-50C impeller was discovered wedged between the inside wall of the reactor vessel and the bottom of the flow skirt in the annulus section of the reactor vessel between the vessel wall and the core support barrel (see Figures 1 and 2). Attempts were made to remove the piece without success.

The location of the wedged impeller piece inhibits examination of the impeller piece, using currently available inspection techniques, without fully off-loading the core of fuel and removing the core support barrel. Attempting to perform the examination with the core support barrel in place, using a yet to be developed alternate inspection technique, would increase the risk of introducing additional unrecoverable (without removing the core support barrel) foreign material into the reactor vessel. Performing a full core off-load and removing the core support barrel for the sole purpose of conducting the IWB-2420(b) examinations would result in increased plant risk and a significant amount of additional radiation dose to station personnel, estimated to be approximately 12.7 Rem.

5. Proposed Alternative and Basis for Use

<u>Proposed Alternative</u>

Pursuant to 10 CFR 50.55a(z)(2), ENO requests authorization to use an alternative to the requirement for three successive reexaminations of the wedged impeller piece: one during the first period, one during the second period, and one during the third period of the fifth ten-year ISI interval, as currently required by IWB-2420(b).

As an alternative to the reexamination requirements of IWB-2420(b), ENO proposes to perform one reexamination of the wedged impeller piece during either the second or the third period of the current fifth ten-year ISI interval.

<u>Basis for Use</u>

ENO has accepted this relevant condition by analysis in accordance with ASME Code requirements, supported by an operability evaluation described in an NRC integrated inspection report (Reference 1, pages 25-29). The evaluation determined that the impeller piece will not dislodge over the remainder of the plant life, and will not fragment due to low hydraulic loads acting on the piece. Therefore, leaving the impeller piece in place will have no adverse effect on any structure, system, or component (SSC) associated with the reactor vessel or the primary coolant system. The evaluation considered the wedged impeller piece location, the likelihood that it could become dislodged, the likelihood that it could fracture into smaller pieces, and its impacts on the adjacent reactor vessel and flow skirt.

Reactor coolant flow and control rod motion are not affected due to the location of the wedged impeller piece. The impeller piece is located in the bottom of the reactor vessel below the core support cylinder and is not covering any of the flow holes in the flow skirt. Also, the PNP control rods are inserted through the top of the reactor head and therefore control rod motion is not impacted (see Figure 3).

The evaluation documented that the piece is unlikely to become dislodged because it is tapered in thickness from 3/16 inches to roughly one inch thick, and the gap between the vessel wall and flow skirt where the piece is wedged has only a maximum gap of 1/2 inch wide. A fluid dynamics analysis was performed, in support of the operability evaluation, to determine the forces that act on the piece during plant operation. The analysis concluded that the maximum lift force acting on the piece is 350 pounds, which is significantly less than the 3000 pounds of force applied to the wedged impeller piece in attempts to hydraulically dislodge and remove it. Finally, heatup and cooldown effects on gap size were considered and it was determined that the flow skirt and vessel would move together such that the gap size would remain constant.

The evaluation documented that the piece is unlikely to break up into smaller pieces during plant operation. This was based on a fracture analysis, which concluded that for all assumed initiating crack sizes in the piece that the crack growth rate would reduce and essentially stop once the crack depth approached 75 percent of the thickness of the piece. Plant history has shown that prior broken impeller pieces small enough to pass through the gap have been found at the bottom of the vessel. Based on this operating experience and the size of the wedged impeller piece, in the unlikely event that the piece was to become dislodged or it were to break up into smaller pieces, the pieces would settle to the bottom of the vessel, and remain in the bottom of the vessel and not affect the safe operation of the plant.

Additionally, the evaluation determined that the cladding under the wedged impeller piece is not removed nor is it deflected in a way that would impact its ability to keep PCS fluid in the vessel and no damage to the cladding has occurred or is expected to occur during plant operations. Therefore the wedged impeller piece does not impact the pressure retaining function of the reactor vessel.

ENO has concluded, based on the above evaluation results, that the impeller piece will not move, will not break up, will not impede primary coolant system flow, and will not affect the pressure-retaining capability of the reactor vessel.

While it is highly unlikely that the condition of the wedged impeller piece will change, there are other inspections that occur each refueling outage that provide an opportunity to identify a change in the wedged impeller piece's condition. For example, ENO conducts a visual inspection of the top of the reactor core as part of planned refueling activities for the purpose of identifying foreign material. Likewise, ENO performs visual inspections of select fuel bundles inside the core and of select discharged fuel assemblies for foreign material, which is another opportunity to identify foreign material that may be indicative of a change in condition of the wedged impeller piece.

6. Duration of Proposed Alternative

The proposed duration of this alternative is for the remainder of the fifth ten-year ISI interval which commenced on December 13, 2015 and ends on December 12, 2025.

7. Precedent

None

8. References

1. NRC report, *Palisades Nuclear Plant Integrated Inspection Report 05000255/2014002*, dated May 7, 2014 (ADAMS Accession Number ML14127A543)

Figure 1
View of the wedged impeller piece looking up from below the flow skirt

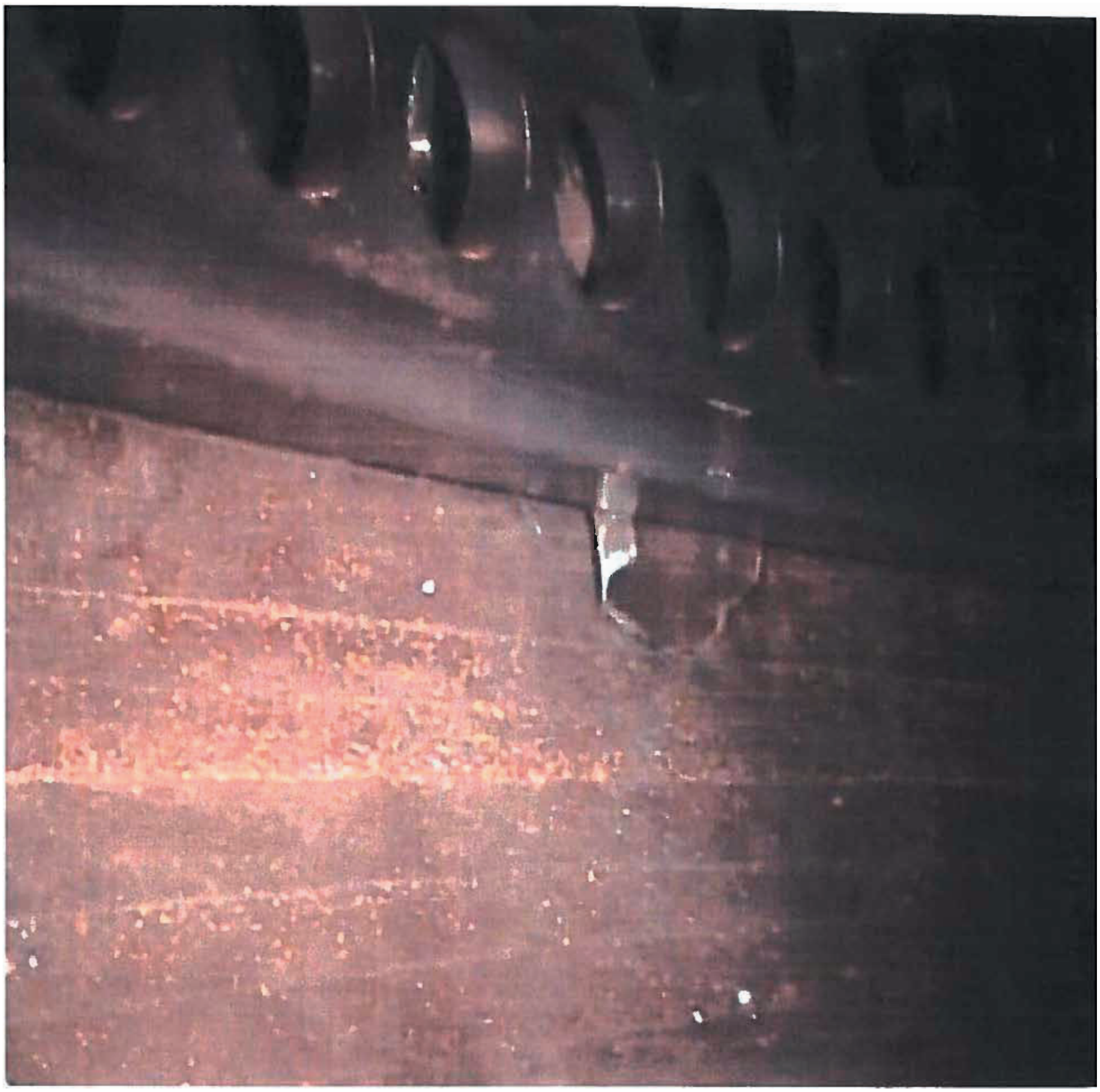

Figure 2
View of the wedged impeller piece looking down from above the flow skirt

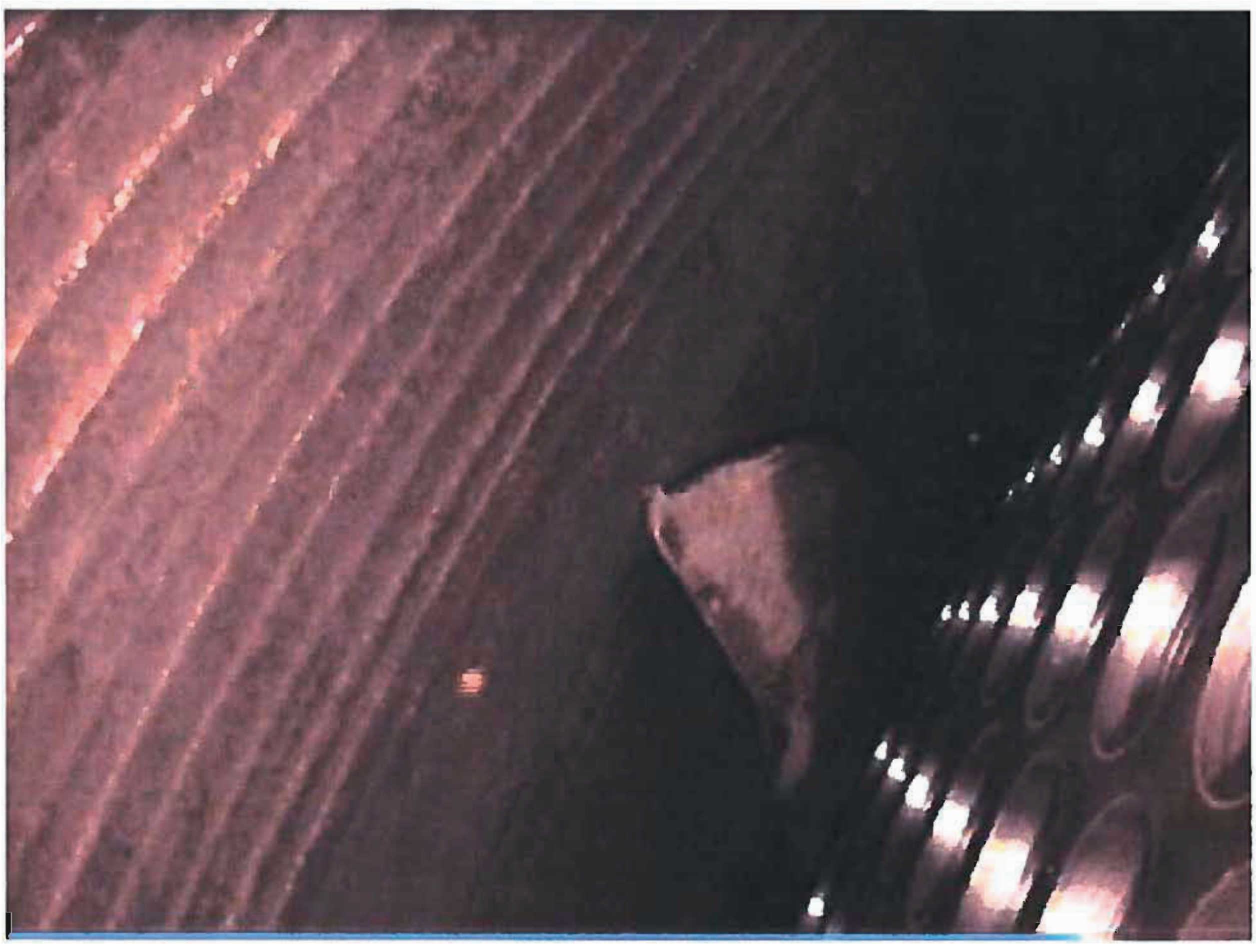

Figure 3
Typical Combustion Engineering Pressure Vessel and Internals Showing Location of the Wedged Impeller Piece