



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

October 13, 2026

Site Vice President
Palisades Energy, LLC
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT – ISSUANCE OF AMENDMENT NO. 283
REGARDING MODIFICATION OF THE LICENSING BASIS TO ALLOW
ALTERNATIVE FUEL HANDLING METHOD FOR THE RECOVERY OF FUEL
ASSEMBLY DD41 (EPID L-2026-LLA-0201) (**EXIGENT CIRCUMSTANCES**)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 283 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant. The amendment consists of changes to the license basis in response to your application dated September 30, 2026, as supplemented by letters dated October 2, 2026, October 8, 2026, and October 12, 2026.

The amendment revises the Updated Final Safety Analysis Report to allow for the removal of the DD41 fuel assembly from the reactor cavity and move the assembly to the spent fuel pool.

This amendment is issued under exigent circumstances as provided in the provisions of paragraph 50.91(a)(6) of Title 10 of the *Code of Federal Regulations* due to the time-critical nature of the amendment. In this instance, an exigent situation exists in that the amendment is needed to allow movement of the fuel assembly out of the reactor cavity in order to resume loading fuel and move toward reactor startup.

A copy of our related safety evaluation is also enclosed. The safety evaluation describes the exigent circumstances under which the amendment was issued and the NRC staff's final no significant hazards determination.

The Notice of Issuance and opportunity for a hearing associated with the exigent circumstances will be included in a future biweekly *Federal Register* notice.

Sincerely,

/RA/

Justin C. Poole, Senior Project Manager
Operating Reactor Licensing Branch 3
Division of Licensing Projects 1
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:

1. Amendment No. 283 to DPR-20
2. Safety Evaluation

cc: Listserv



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

HOLTEC PALISADES, LLC

PALISADES ENERGY, LLC

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 283
Renewed License No. DPR-20

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by Palisades Energy, LLC, dated September 30, 2026, as supplemented by letters dated October 2, 2026, October 8, 2026, and October 12, 2026, 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.

2. Accordingly, by Amendment No. 283, Renewed Facility Operating License No. DPR-20 is hereby amended to authorize revision to the Palisades Nuclear Plant, Updated Final Safety Analysis Report (UFSAR), as set forth in the licensee's application dated September 30, 2026, as supplemented by letters dated October 2, 2026, October 8, 2026, and October 12, 2026, and evaluated in the NRC staff's evaluation enclosed with this amendment.
3. This license amendment is effective as of the date of issuance and shall be implemented within 10 days of the date of issuance. The UFSAR changes shall be included in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

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BERRIOS
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Ilka Berrios, Chief
Operating Reactor Licensing Branch 3
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-20

Date of Issuance: October 13, 2026



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 283 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

HOLTEC PALISADES, LLC.

PALISADES ENERGY, LLC

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated September 30, 2026 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML26273A469), as supplemented by letters dated October 2, 2026, October 8, 2026, and October 12, 2026 (ML26275A214, ML26281A333, and ML26285A034, respectively), to the U.S. Nuclear Regulatory Commission (NRC or the Commission), Palisades Energy, LLC (the licensee), submitted a license amendment request (LAR) to make certain changes to the Renewed Facility Operating License DPR-20 for Palisades Nuclear Plant (Palisades, PNP).

The proposed amendment would modify the PNP Updated Final Safety Analysis Report (UFSAR) to allow for the removal of the DD41 fuel assembly from the reactor cavity and move the assembly to the spent fuel pool using an alternative process. Specifically, the proposed amendment would modify UFSAR sections 9.11 and 14.19, on a one-time basis, to describe an alternative fuel-handling process. While the licensee requested approval of the proposed change as an emergency license amendment in accordance with the regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.91(a)(5), the NRC processed this amendment request as an exigent amendment in accordance with 10 CFR 50.91(a)(6).

2.0 REGULATORY EVALUATION

2.1 System Description

Section 9.11 of the PNP UFSAR describes the operation and requirements of the fuel handling and storage system for new and spent fuel. This section of the UFSAR contains descriptions of the structural analysis of the Spent Fuel Pool (SFP) and criticality prevention analysis, as well as descriptions of the new fuel storage areas. Section 14.19 of the PNP UFSAR describes the possibility of an accident with significant consequences during refueling. For the purpose of

defining the upper limit of the radiological consequences of a fuel handling accident, it is assumed that the fuel bundle is dropped during handling.

2.2 Licensee's Proposed Changes

PNP has proposed revisions to UFSAR sections 9.11 and 14.19 to describe an alternative fuel-handling process. The process would be a one-time operation limited to recovery of fuel assembly DD41. Once recovery of fuel assembly DD41 is completed, PNP plans to transition the changes to historical UFSAR content. As stated in Section 2.3 of the LAR, below is a description of the proposed revisions:

The alternative fuel-handling process would lift DD41 using the Containment Polar Crane (L-1) and purpose-designed fuel movement device (FMD) supported by slings and chain-falls to remove the assembly from the reactor. Once removed from the reactor, DD41 would be loaded underwater into the FTC [Fuel Transfer Canister] staged on the reactor cavity floor. The FTC containing DD41 would be lifted from the water, drained, and then transported through containment using the crane and existing rail system along a lift path that exits through the containment equipment hatch into the SFP area of the auxiliary building. Once in the SFP area, the FTC would be lifted using the Fuel Pool Building Crane (L-3) and placed vertically in the Region 2 cask laydown area of the SFP.

All existing radiation protection analyses, heavy-load lift analyses, and procedures have been reviewed and revised, as necessary, to incorporate requirements associated with this lift. These changes would be incorporated into the applicable procedures before any fuel-handling activity using the alternative process.

2.3 Regulatory Requirements and Guidance

The regulatory requirements and guidance on which the NRC based its acceptance and evaluation of this LAR are listed below.

Regulatory Requirements

PNP was designed prior to the publishing of the Atomic Energy Commission (AEC) "General Design Criteria (GDC) for Nuclear Power Plant Construction Permits," draft for comment, 32 FR 10214 (July 1967). Therefore, PNP was not licensed to the current 10 CFR Part 50, Appendix A, GDC. The PNP principal design criteria for the design, construction, and operation of the plant are listed in PNP UFSAR, section 1.4, "Principal Design Criteria." An evaluation comparing the PNP design basis to the AEC GDCs as they appeared in 10 CFR Part 50, Appendix A on July 7, 1971, is presented in PNP UFSAR, section 5.1, "General Design Criteria." The portions of the UFSAR applicable to this amendment include GDC 61, "Fuel storage and handling and radioactivity control," and PNP UFSAR sections 1.4.8 "RADIOACTIVE WASTES AND RADIATION PROTECTION" and 1.4.9 "FUEL HANDLING AND STORAGE". UFSAR section 1.4.8 states:

The radioactive waste treatment system was designed so that discharge of radioactivity to the environment is in accordance with the requirements of 10 CFR, Part 20, and Appendix I to 10 CFR 50. The Facility was provided with a centralized control room having adequate shielding to permit occupancy during

all credible accident situations. The radiation shielding in the Facility, in combination with Facility radiation control procedures, ensures that personnel do not receive radiation exposures in excess of the applicable limits of 10 CFR, Part 20, during normal evolutions and maintenance.

UFSAR section 1.4.9 states: "Fuel handling and storage facilities were provided for the safe handling, storage and shipment of fuel and will preclude accidental criticality."

Section 50.67, "Accident source term," of 10 CFR states that any licensee that was initially authorized to operate prior to January 10, 1997, and who seeks to revise the current accident source term in its design basis radiological consequence analyses, must apply for a license amendment under 10 CFR 50.90, "Application for amendment or license, construction permit, or early site permit." Paragraph (b)(2) of 10 CFR 50.67 provides that the NRC may issue the amendment if the application's analysis demonstrates with reasonable assurances the following:

- (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).
- (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), and
- (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Additionally, under Section 50.92(a) of 10 CFR, in determining whether an amendment to a license will be issued, the NRC staff is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be "reasonable assurance" that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission's regulations.

Guidance Documents

- NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980.
- NUREG-0800, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," March 2007.
- NUREG-0800, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," July 1981.
- NUREG-1565, "Dry Oxidation and Fracture of LWR Spent Fuels," November 1996.
- Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
- Interim Staff Guidance (ISG)-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3.
- NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material."

- NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities."

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the LAR to determine if the proposed revisions to the UFSAR meet the requirements in section 2.3 above.

3.1 Change to UFSAR Section 9.11, Fuel Handling and Storage Systems

The licensee has proposed a one-time change to UFSAR section 9.11.4 to define an alternate fuel handling method using a FTC and a Fuel Handling Device.

UFSAR section 9.11.4 defines the normal handling of fuel assemblies during fuel loading. Normally the refueling machine removes a spent fuel assembly from the core, transports it to the tilt machine, and deposits it in the transfer carriage within the tilt machine. The carriage is then rotated from a vertical position to a horizontal position and moved through the transfer tube to the spent fuel storage area. The carriage is then rotated to a vertical position, the spent fuel removed and placed in a storage rack by the service platform. Due to the conditions described in the LAR, fuel assembly DD41 cannot be recovered using the refueling machine or transfer tube as the method described in UFSAR, and the licensee has proposed an alternative method.

The alternative transfer process described in the LAR consists of using the Containment Polar Crane (L-1) to transfer DD41 into the laydown area within the reactor cavity. To minimize risk while moving fuel within the reactor cavity, the licensee developed a specially designed fuel movement device (FMD) to move fuel assembly to the FTC. DD41 is currently oriented at an approximate 30-degree angle with the lower tie plate in core location H-1 and the upper tie plate leaning against the core shroud on the opposite side of the core. Fuel assembly DD41 will be maintained at a minimum of 10 feet underwater while being lifted and transferred to the FTC within reactor cavity. This FTC encloses the fuel assembly to enable lifting out of the reactor cavity. The fuel will be laid down into the FTC horizontally and an FTC cover installed. During the operation, the radiological consequences of fuel handling accident (FHA) with assembly DD41 within the reactor cavity under the assumed 10 feet of water or when placed in the spent fuel pool, remain bounded by the existing FHA described in UFSAR section 14.19.

Once the fuel assembly and lid are secured on the FTC, the canister is lifted out of water using the Containment Polar Crane (L-1), drained, and moved to the equipment hatch location where it is placed on an equipment hatch trolley for transfer to the SFP within auxiliary building. As indicated by licensee, the FTC is handled along established safe load path using developed procedures. To minimize consequence of accident, the licensee has performed a load drop analysis. The review of the FTC Drop Analysis is covered below in section 3.4 Recovery Equipment – Drop Analysis.

Within the auxiliary building, a single failure proof fuel handling crane is used to lift the FTC and rotate into vertical configuration to lower into the cask pit located within the spent fuel pool. The licensee has taken safety precautions, including use of single failure proof crane and administrative controls, to minimize risk while rotating and lowering the FTC into the SFP. The licensee has further described the loaded FTC movements within the fuel building following a procedurally established safe load path. In letter dated October 8, 2026, the licensee's response to the Request for Additional Information (RAI) verified that the L-3 fuel building crane is

currently configured as single failure proof from previously supporting the last dry fuel loading campaign and will retain the single failure proof configuration while handling DD41 and FTC.

The assembled and loaded FTC has an approximate lift weight of 49,000 pounds. Based on weight, the loads moved during the alternative method are considered heavy loads. NUREG-0612 provides criteria for control of heavy loads. If a heavy load were to drop, they could impact on stored spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. General criteria for programmatic guidelines for design, operation, testing, maintenance, and inspection are specified in NUREG-0612, including safe load paths and procedures which have been developed. As described, the licensee has evaluated handling fuel assembly from reactor cavity to spent fuel pool and developed corresponding procedures for safe handling utilizing the proposed alternative transfer process. The staff has accessed and considered the licensee-provided fuel handling procedure (FHSO-19A) on their electronic portal to gain an understanding of the process.

Based on the alternative recovery process described in the LAR for this one-time handling of fuel assembly relocation, the staff finds adequate description is provided in updated UFSAR 9.11.4 for safe handling of fuel and minimizing risk of a dropped load.

3.2 Fuel Handling Accident

3.2.1 Change to Fuel Handling Accident

A fuel handling accident is considered a design bases accident at Palisades that requires a radiological consequences analysis. As documented in PNP UFSAR section 14.11, "Postulated Cask Drop Accidents," and section 14.19, "Fuel Handling Accident," this analysis was previously performed to comply with Regulatory Guide (RG) 1.183, Revision 0 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." RG 1.183 outlines various assumptions and methodologies the staff find acceptable for performing the radiological consequence analysis. Appendix B of RG 1.183 details assumptions for evaluating the radiological consequences of an FHA. HI-2261348, Revision 1 "Radiological Consequences of DD41 FHA at Palisades" – which documents the analysis performed in alignment with RG 1.183 – was included as an attachment to the LAR.

Based on previous radiological consequence analyses, an FHA was assumed to occur using equipment that was dedicated to fuel handling (such as the fuel handling manipulator crane). Additionally, all fuel movement during refueling operations was assumed to occur under water with a minimum depth of 23 ft of water on top of the assembly in either the containment refueling cavity or spent fuel pool. The water in the cavity and the pool during fuel movement supports the radiological consequence analyses because it provides shielding to workers, and during a hypothetical FHA the water retains some of the fission products that are released.

The staff evaluated the proposed movement of fuel assembly DD41 since the licensee proposed a new proposed pathway out of containment to the spent fuel pool. The proposed use of the polar crane (as opposed to the fuel manipulator crane) enables PNP to move fuel directly out of the refueling cavity (and water). The licensee's plan is to place DD41 in an engineered Fuel Transfer Canister (FTC) and then take the FTC, with DD41 within it, out of the water. The planned removal pathway takes the FTC out of containment through an equipment hatch, so any mitigation of an FHA accident dose consequence analysis from water in the refueling cavity or SFP cannot be credited.

3.2.2 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged because of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident.

The source term for the Palisades DD41 FHA was developed in accordance with NRC Regulatory Guide 1.183, Appendix B, which prescribes conservative assumptions for accident radiological consequence analyses. The inventory of radionuclides considered for release was derived from the calculated isotopic data for fuel assembly DD41. Only those nuclides residing in the fuel rod gap and plenum (specifically noble gases, halogens, and alkali metals) were included in the FHA analysis, consistent with regulatory position 1.2 of RG 1.183, Appendix B. For an FHA in containment, no source term was considered for fuel assemblies other than DD41.

A key conservative assumption in the source term modeling is the extent of fuel damage. The analysis assumes that all fuel rods in DD41 are damaged and their entire gap inventory is available for release. This aligns with Regulatory Guide 1.183, Appendix B regulatory position 1.1. The total inventory is then modified in the RADTRAD code by applying radionuclide-specific release fractions and timings provided in RG 1.183. For FHA, all gap activity is assumed to be released instantaneously further ensuring conservatism in the dose calculations.

Consistent with regulatory position 1.3 of RG 1.183, Appendix B, the analysis assumes that iodine is released as 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. These fractions are applied to the gap inventory of iodine isotopes, reflecting the expected speciation under accident conditions. The release fractions for other radionuclide groups are also taken directly from Table 3 of RG 1.183. The source term is thus constructed to represent a worst-case, yet realistic, scenario modeling the FHA.

The resulting source term is processed through the RADTRAD model, which simulates transport, and mitigation of radioactive material through plant compartments and release pathways. The model accounts for the effects of filtration (e.g., HEPA and charcoal filters in the fuel handling building (FHB) stack exhaust), atmospheric dispersion, and receptor characteristics (breathing rates, occupancy factors). The NRC staff finds that the source term developed for PNP DD41 FHA was generated in accordance with RG 1.183, Revision 0.

3.2.3 Transport

The transport pathways evaluated in this FHA differ substantially from those in a standard fuel-handling accident because the proposed recovery method removes the irradiated fuel assembly from its normal underwater shielding environment and carries it above water within an engineered FTC. In a typical FHA, radionuclides released from damaged rods are assumed to be instantaneously mixed within the surrounding water of the refueling cavity or spent fuel pool, where significant mitigation occurs through water holdup and large decontamination factors are credited under RG 1.183. By contrast, the Palisades DD41 recovery introduces periods during which the assembly is transported dry, first when the FTC is lifted from the reactor cavity, then as it travels across containment, through the open containment equipment hatch, and into the

FHB. Because these steps occur outside a water environment, none of the standard FHA water-scrubbing or iodine retention mechanisms can be credited. As a result, the bounding radiological transport scenario assumes direct atmospheric mixing within containment, unmitigated flow through the open equipment hatch into the FHB, and subsequent exhaust through the FHB ventilation and filtration system to the plant stack, representing a more conservative, and fundamentally distinct transport configuration than the conventional underwater FHA pathway.

A critical component of the transport evaluation is the configuration of the containment equipment hatch. The applicant assumes the hatch is open at the time of the postulated drop—a configuration explicitly chosen because it maximizes potential release to adjacent plant areas and results in higher offsite and control room dose consequences. With the hatch open, radionuclides in the containment atmosphere are transported directly into the FHB with no credit for holdup or deposition in containment during the transfer. These assumptions bound accident transport modeling during the fuel-handling in containment.

Once radionuclides enter the FHB, transport is governed by the building's engineered ventilation system, which exhausts through a high-capacity filtration train to the plant stack. The applicant modeled the FHB exhaust at 75,000 cubic feet per minute (cfm) and credited filtration efficiencies of 99 percent for aerosols and 94 percent for elemental and organic iodine. The transport model does not take credit for deposition on surfaces, leakage into adjacent plant spaces, or additional building holdup, ensuring that the filtered stack exhaust pathway is the sole mechanism by which radionuclides leave the building.

Transport to the control room was modeled using both unfiltered and filtered intake streams, reflecting transitions between normal, and emergency ventilation modes. Prior to control room isolation, the model applies an unfiltered intake of 660 cfm, allowing a portion of airborne material to enter the control room without removal. At 20 minutes, consistent with the applicant's credited operator action, the control room shifts to emergency mode, eliminating unfiltered intake, and initiating filtered make-up and recirculation flows providing 99 percent removal efficiency for aerosols, elemental iodine, and organic iodine. The control room model also incorporates unfiltered in-leakage of 100 cfm.

Initially, it was not clear to the staff if the movement of DD41 in the FHB was considered by the licensee or bounded by their radiological consequence analysis in HI-2261348, Revision 1 (ML26273A469). As stated in the supplement to the LAR dated October 2, 2026 – the planned travel path of DD41 (in the FTC) would be “over the North Tilt Pit where spent fuel is currently stored.” HI-2261348, Rev. 1 only considers a source term from DD41 and does not consider damage to other fuel assemblies. The staff asked an RAI to determine if an existing analysis considers a cask drop over the spent fuel, and if that analysis bounds a potential drop of DD41 in the North Tilt Pit. The PNP UFSAR (section 14.11.3.1.1 “Radiological Consequences of a Cask Drop in the Spent Fuel Pool”) describes what bounding analysis has been previously performed for a cask drop in the spent fuel pool. The description in the UFSAR describes a cask drop onto 90-day decayed fuel. In their RAI response dated October 8, 2026 (ML26281A333) the licensee stated that calculation NAI-1149-026, Revision 0 (ML062830469) bounds a potential drop of DD41 onto spent fuel assemblies. NAI-1149-026 postulates a cask drop which impacts 73 fuel assemblies. The licensee stated that the FTC being used to move DD41 is not physically large enough to impact 73 fuel assemblies. Since the analysis in NAI-1149-026 considers fuel that has a decay time significantly less than the fuel in the FTC and spent fuel pool – and the FTC can only physically impact a number of fuel assemblies less than 73 – NAI-

1149-026 can be considered a bounding analysis for moving the FTC over fuel in the spent fuel pool.

The staff has accessed and considered the licensee-provided fuel handling procedure (FHSO-19A) on their electronic portal to gain an understanding of both the site's procedure and how it aligns with the FHA analysis. The NRC staff finds that the modeling of radionuclide transport during the analysis of an FHA is both conservative and bounding in accordance with RG 1.183, Revision 0, and therefore acceptable.

3.2.3.1 Fuel Handling Accidents Atmospheric Dispersion values (χ/Q)

The χ/Q values used in this analysis are part of the current licensing basis and the licensee did not request to change this aspect of its licensing basis. These values are included in the September 25, 2006 (ML062830385), LAR *Palisades - Alternative Source Term- Proposed License Amendment*.

3.2.4 Control Room (CR) Habitability for Fuel Handling Accidents

The NRC staff evaluated the applicant's assessment of Control Room (CR) habitability following a postulated DD41 FHA to ensure compliance with the dose limits of 10 CFR 50.67. Because the proposed DD41 recovery process includes periods in which the fuel assembly is transported above water and radionuclides may be released directly to containment and subsequently to the FHB, the potential exists for atmospheric transport of radioactive material without the hold up and retention of the reactor cavity or spent fuel pool water volume, to the control room via normal and emergency intake pathways and including unfiltered in-leakage. The applicant's analysis appropriately models both the early unfiltered intake phase and the mitigated emergency ventilation phase, ensuring that the bounding dose to the CR operators is captured for the duration of the event.

During the initial 20-minute period following the postulated accident, the CR is assumed to operate in normal ventilation mode, drawing unfiltered intake air at a rate of 660 cfm. This assumption is conservative because it allows airborne radionuclides present in the environment or entrained in the FHB exhaust plume to reach the CR without mitigation. The 20-minute delay period matches the delay period which has been previously used for this analysis as documented in Table 14.19-1 "Fuel Handling Accident (FHA) Radiological Analysis – Inputs and Assumptions" of the PNP UFSAR. By assuming continuous unfiltered intake until operator action, the analysis fully accounts for the most limiting period of potential CR exposure.

At 20 minutes, consistent with credited operator action and the design of the Control Room Emergency HVAC, the control room ventilation system aligns to Emergency Mode. In this configuration, unfiltered intake flow is terminated and replaced with filtered makeup and filtered recirculation flows totaling approximately 1,413 cfm. A 99-percent removal efficiency is applied to aerosols, elemental iodine, and organic iodine species, consistent with the performance capabilities of the emergency ventilation charcoal and HEPA filtration trains. The model also includes 100 cfm of unfiltered in-leakage for the duration of the accident, which conservatively represents realistic infiltration pathways and prevents over-crediting of CR isolation. This treatment is consistent with NRC staff expectations for control room habitability evaluations, particularly in analyses performed under the Alternative Source Term (AST) framework. The applicant incorporated appropriate receptor characteristics, including a breathing rate of 3.5×10^{-4} m³/s and time-weighted occupancy factors of 1.0, 0.6, and 0.4 for the 0–24 hour,

1–4 day, and 4-30 day periods, respectively. These values are directly aligned with RG 1.183 and NUREG-0800 guidance for CR dose assessments.

The NRC staff audited information provided in the portal regarding the most recent Control Room Envelope Integrated Unfiltered In-Leakage Test values. The NRC staff confirmed that the test values did not exceed the assumed unfiltered inleakage used in the LAR. The use of higher assumed unfiltered inleakage value provided in the LAR would result in higher calculated dose to control room operators that is consistent with CLB, and is therefore acceptable.

Based on review of the inputs, assumptions, and the applicant's RADTRAD modeling, the NRC staff finds that the CR habitability assessment reasonably characterizes the range of conditions expected following the postulated DD41 FHA. The analysis demonstrates that the total TEDE to the control room operator remains within the 5-rem limit of 10 CFR 50.67, even under the bounding assumptions of early unfiltered intake, continuous infiltration, and conservative radionuclide transport conditions. Therefore, the staff concludes that the proposed one-time DD41 recovery activity maintains adequate control room habitability and satisfies the guidance of RG 1.183, Revision 0 for accident dose analyses.

3.2.5 Conclusion

Holtec evaluated the radiological consequences resulting from the postulated new variation of an FHA and concluded that the radiological consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and CR are within the radiological dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in Standard Review Plan section 15.0.1. This evaluation is contained in the attachment for the LAR in calculation PNP 2026-052 "Radiological Consequences of DD41 FHA at Palisades."

The NRC staff's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in the relevant sections of this safety evaluation. The NRC staff performed an independent review of all inputs, assumptions, and initial conditions in the licensee dose assessment files. The NRC performed confirmatory calculations using RADTRAD version 5.2.1. The NRC staff finds that the EAB, LPZ, and CR doses for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.3 Movement of Fuel Assembly DD41 Outside of the Reactor Cavity or Spent Fuel Pool

3.3.1 Radiological Source Term – 4- Year Decay

The radiological profile of DD41 after 4 years of decay and ingrowth is dominated by long-lived fission products and transuranics. The NRC staff performed independent decay calculations, also considering ingrowth of radionuclides due to decay to model the inventory of radionuclides in the assembly, and compared that with the licensee modeled inventory in the assembly DD41. The values the NRC derived using the code SCALE, were consistent with those that the licensee calculated. The modeled assembly inventory forms the basis for understanding the expected radiological conditions when moving the assembly in the FTC.

3.3.2 Dry Movement–Gamma and Neutron Shine in Shielded Transfer Cask

The primary radiological hazard during dry movement is external exposure ("shine") due to gamma and neutron decay from the unshielded assembly. Water shielding is not present, and only cask engineering controls remain to attenuate these dose fields. The transfer cask must

provide sufficient thickness and density to attenuate gamma and neutron emissions, maintain shielding integrity, and prevent streaming paths.

The NRC staff performed independent confirmatory gamma-shine and neutron-shine calculations using the SCALE code to evaluate the licensee's assessment of external exposure during movement of the assembly in the shielded fuel transfer canister after dewatering. The staff's results were consistent with those provided by the licensee and were at radiation levels similar to those expected for comparable outage work activities. Further, the results supported the conclusion that the transfer cask shielding is sufficient to attenuate the expected gamma and neutron radiation fields. Based on this confirmatory analysis, the NRC staff finds that the licensee's evaluation of gamma-shine and neutron-shine hazards during dry movement is acceptable.

3.3.3 Streaming and Geometry Effects

The licensee evaluated the potential for gamma streaming that could arise if a drop or other accident involving the transfer cask were to occur as documented in section 3.2.1 "Dose Comparisons" of the LAR. Such an event could alter shielding geometry creating localized areas of elevated dose rate. The licensee performed analyses to assess these potential streaming conditions. These streaming effects could be present also due to misalignment of the cask, or any other manufacturing deviations from the drawing plans. Such an event could alter shielding geometry. In section 3.2.1 of the LAR the licensee stated:

The two conditions that are evaluated are normal conditions as designed, as well as conditions from an accidental drop of the canister that assumes a complete loss of neutron and gamma shielding.

This statement appeared to be contradictory to the information provided in calculation HI-2261348, *Radiological Consequences of DD41 FHA at Palisades* and the document HI-2261303 *Dose Evaluation for the Fuel Transfer Canister* executive summary, which state that for a drop accident no damage to the canister is expected that would have significant impact on dose rates and that dose rates around the canister are expected to be essentially the same as for normal conditions. The NRC staff questioned the licensee on this item. The licensee responded on the docket in the *Response to Request for Additional Information Regarding Emergency License Amendment Request: One-Time-Use Amendment to Allow Alternative Fuel Handling Method for the Recovery of Fuel Assembly DD41* (ML26275A214), saying:

A draft analysis was performed assuming total loss of neutron and gamma shielding. However, following the positive results from the FTC Drop Analysis contained in HI-2261346, Fuel Transfer Canister Drop Analysis [], a complete loss of shielding was deemed not credible and subsequently removed from the radiological consequences analyses prior to review and approval.

The review of the FTC Drop Analysis, *HI-2261346*, is covered below in section 3.4 Recovery Equipment – Drop Analysis. The NRC staff reviewed the expected dose rates during both normal and accident (cask drop) scenarios. The NRC staff reviewed the expected values with the assembly in the shielded fuel transfer canister after removal from the reactor cavity as provided in section 2.3.1 Summary of UFSAR Revised Text and 3.2.1 Dose Comparison.

The NRC staff performed independent confirmatory streaming dose-rate calculations using the SCALE code to assess the licensee's analysis. The staff's results showed consistency with the

licensee's calculated dose rates, and the staff did not identify information that would contradict the licensee's conclusions with respect to streaming effects related to the FTC. The NRC staff finds that the licensee's evaluation provides an adequate analytical basis for characterizing streaming-related dose rates during dry transfer.

3.3.4 Airborne Radiological Hazard

The licensee addressed the potential airborne conditions in section 3.2.3 "Airborne Radioactivity and Contamination Controls." Because DD41 appears to remain intact and reactor cavity coolant sampling directly adjacent to the assembly showed no abnormal activity from fission products, emergency ventilation actuation is not expected. In addition, the DD41 assembly's known structural damage and proposed handling outside the normal underwater fuel handling configuration are addressed through the licensee's radiation protection programs in evaluating the potential for airborne radioactivity and contamination.

While submerged under a minimum of 20 feet of water, airborne radiological consequences remain bounded by current licensing basis fuel handling accident radiological consequences. Examination of the integrity of the fuel rods is vital to ensure the potential effects of airborne contamination are minimized when the cask is removed from the reactor cavity. The cask is not sealed, and airborne contamination is a potential risk if there is any fuel breach. To address this issue and in response to the staff's question, the licensee supplemented its evaluation to include collection of a chemistry sample after the fuel assembly was placed on the FTC base and the top of the FTC was positioned on the base. This post-movement sample will provide additional evidence to confirm that no fuel rod damage occurred during the transition from the assembly's current position into the cask and thereby limit potential airborne radionuclide concerns during the movement of the assembly and cask. In response to an RAI, the licensee provided a calculation of the potential radiological conditions related to airborne contamination if the fuel assembly is damaged while placing and securing the top of the FTC onto the base holding the assembly. Based on the calculation provided, the NRC staff has reasonable assurance that regulatory limits would not be exceeded due to causing fuel failure in assembly DD41 while securing the top of the FTC to the base. Based on the information available, the NRC staff finds that incorporating the procedural step of examining the fuel assembly via chemistry sample adjacent to the FTC will provide an adequate basis for verifying the assembly's integrity prior to movement.

3.3.5 Conclusion

Based on the information provided in the LAR and the NRC staff's independent confirmatory calculations, the NRC staff finds that there is reasonable assurance that the radiological conditions during normal and accident conditions as a result of the proposed DD41 recovery activities will not endanger the health and safety of the public and will meet the provisions of 10 CFR 20.1201, "Occupational Dose Limits for Adults," and 10 CFR 50.67, "Accident Source Term."

3.4 Recovery Equipment

FTC Lifting and Handling

In the application, the licensee stated that the design and Level A service limits of ASME Section III, Subsection NF are adopted as the basis for evaluating the FTC under normal lifting and handling conditions. The licensee also stated that the calculated stresses include the

specified dynamic load factor and are compared with the allowable stresses applicable to the material, temperature, component, and stress category and that the ultimate-strength design factor of 10 specified in NUREG-0612, Section 5.1.6(3)(b), for interfacing lift points (i.e. FTC trunnions) was adopted for the combined static and dynamic lifting load. The FTC structural evaluation details under normal lifting and handling conditions are contained in HOLTEC Report HI-2261319, Revision 2, "Structural Evaluation of Fuel Transfer Canister for Palisades Nuclear Plant," which the licensee provided as part of the audit process. The analysis used dimensions and weights identified on the FTC design drawings and employed static equilibrium and strength-of-materials principles to evaluate potential failure modes associated with normal lifting and handling operations.

As part of the audit, the NRC staff reviewed the design input within the calculation package and confirmed the accuracy of the configuration, loading, load path through structural elements, material properties and boundary conditions. Furthermore, the NRC staff reviewed the licensee's methodology, assumptions, and results documented in the HOLTEC report. Based on its review, the staff finds that the licensee evaluated the FTC using the applicable requirements of ASME Section III, Subsection NF, including the design requirements and Level A Service Limits, and evaluated interfacing lift points using the ultimate-strength design factor specified in NUREG-0612, Section 5.1.6(3)(b). The staff further finds that the calculated stresses remain within the applicable allowable limits and that the lifting interfaces satisfy the specified lifting-device acceptance criteria. Accordingly, the staff concludes that the FTC is structurally adequate for normal lifting and handling operations and will perform its intended function under the evaluated loading conditions.

FTC Seismic Stability and Floor Loading Analysis

In the application, the licensee stated that the FTC will be supported on a vertical support stand within the cask loading area of the SFP. The licensee also stated that this configuration has been evaluated for seismic stability under the safe shutdown earthquake (SSE) loading condition and that the evaluation addressed the potential for overturning and sliding, and the structural adequacy of the SFP floor slab. The details of the seismic stability analysis of the FTC and floor loading evaluation are contained in HOLTEC Report HI-2261282, Revision 2, "Seismic Stability Analysis and Floor Loading Assessment of the Fuel Transfer Canister in the Spent Fuel Pool at Palisades," which the licensee provided as part of the audit process.

As part of the audit, the NRC staff reviewed the design input within the calculation package and confirmed the accuracy of the configuration, loading and seismic design parameters, section properties and boundary conditions applied. Furthermore, the staff verified that the analytical models and assumptions were appropriate for the stability check of the stowed FTC configuration and the floor assessment at the specified location in the building. Engineering drawings were reviewed to ensure that the geometric dimensions correspond with those used in the analysis models. The material properties detailed in the report were also reviewed to confirm they were properly referenced and applied.

The NRC staff notes that the licensee determined the dynamic characteristics of the combined FTC, vertical support stand, and supporting SFP slab to establish the applicable seismic accelerations for the stability evaluation. The natural frequencies were calculated using established engineering methods and incorporated the weights of the fuel assembly, FTC, and support stand, together with existing plant floor-loading information. The licensee evaluated overturning and sliding stability using force and moment equilibrium analyses. Based on its review of the methodology and results documented in the HOLTEC report, the NRC staff finds

that the licensee appropriately evaluated overturning stability, sliding resistance, and SFP floor slab loading for the safe shutdown earthquake (SSE) and adequately considered the dynamic characteristics of the FTC support configuration in the stability assessment.

Based on its review of the HOLTEC report and associated referenced documents, the NRC staff finds that the licensee demonstrated that the FTC and vertical support stand remain stable against overturning during the SSE loading condition. The NRC staff also finds that the calculated sliding displacement remains less than the available clearance within the cask loading area and that the calculated floor slab demands remain below the applicable structural capacities used in the evaluation. Accordingly, the staff concludes that the FTC, vertical support stand, and supporting SFP floor slab are structurally adequate for the evaluated SSE loading condition and provide reasonable assurance that the FTC will perform its intended function under the analyzed configuration.

Thermal Evaluation

The licensee developed a three-dimensional (3-D) Computational Fluid Dynamics (CFD) model to perform the thermal evaluation during transfer operations details of which are contained in Holtec Report HI-2261312, Revision 0, "Thermal Evaluation of Fuel Transfer Operations using Fuel Transfer Canister at Palisades NPP," which the licensee provided as part of the audit process. The NRC staff conducted a comprehensive review of the applicant's thermal model utilized in the analysis. The staff examined the code input within the calculation package and confirmed the accuracy of the material properties and boundary conditions applied. Furthermore, the staff verified that the selected code models and assumptions were appropriate for the flow and heat transfer characteristics present in the fuel assembly-loaded FTC at Palisades Nuclear Plant. Engineering drawings were evaluated to ensure that the geometric dimensions correspond with those used in the analysis models. The material properties detailed in the report were also reviewed to confirm they were properly referenced and applied.

The staff conducted audit reviews of bounding configurations to ensure that essential thermal design parameters were accurately identified and properly integrated into the thermal analysis. Energy and mass balance checks were performed, confirming adequate convergence of the applicant's thermal analyses. Additionally, several sensitivity analyses were carried out by modifying key assumptions to verify the accurate characterization of heat transfer and flow type in the analysis. The staff concluded that the licensee's calculated maximum temperatures are within allowable limits, per ISG 11, Revision 3.

The licensee stated that the allowable fuel exposure duration in air was determined using the cumulative oxidation usage methodology described in the HI-STAR 80 SAR (Package ML18262A101), and that the methodology is based on the temperature-dependent oxidation kinetics of UO₂ documented in NUREG-1565 "Dry Oxidation and Fracture of LWR Spent Fuels," and limits the cumulative oxidation usage to $U \leq 1.0$. The licensee also stated that the bounding fuel temperature calculated by the computational fluid dynamics analysis was used to determine the permissible duration of fuel exposure to air. As part of the audit process, the NRC staff reviewed an evaluation of potential fuel pellet oxidation of a fuel assembly that is transferred through the air from the reactor core to the SFP. The staff reviewed the licensee's evaluation of potential fuel pellet oxidation and calculation of permissible time limit for fuel exposure to air and determined that it is consistent with the guidance in NUREG-1565, "Dry Oxidation and Fracture of LWR Spent Fuels," Section 7.4.14.2 of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," and Appendix 8C of NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities." After

reviewing the information in the audit, the staff finds the licensee's evaluation of fuel pellet oxidation in the LAR to be acceptable.

Drop Analysis

In the application, the licensee stated that the structural response of the FTC was evaluated for three bounding horizontal drop conditions using LSDYNA analyses that included the following: FTC pure horizontal impact, FTC horizontal drop with the top end impacting first, and FTC horizontal drop with the bottom end impacting first. The licensee also stated that, based on the calculated response, the FTC maintains its overall structural configuration throughout the evaluated impact events, all bolts have bending and shear safety factors greater than unity, and all calculated combined-loading interaction ratios are below unity. The structural evaluation details of the FTC drop analysis are contained in HI-2261346 Revision 0, "Fuel Transfer Canister Drop Analysis," which the licensee provided as part of the audit process.

The NRC staff reviewed the licensee's methodology for evaluating the structural response of the Fuel Transfer Canister (FTC) under postulated drop conditions. The licensee developed a detailed finite element model of the FTC and performed nonlinear dynamic analyses using LSDYNA to evaluate three bounding drop orientations. The analyses incorporated the FTC geometry, material properties, structural connections, and impact conditions to determine component stresses, strains, accelerations, and connection loads. The NRC staff finds that the use of a nonlinear dynamic finite element analysis is an acceptable method for evaluating the FTC response to the postulated drop events and provides an adequate basis for assessing compliance with the established structural acceptance criteria.

The staff reviewed the results against the acceptance criteria established by the licensee, which require that: (1) the FTC lid remain attached to the base assembly, (2) the lid-to-base bolts not experience gross failure, (3) the end plates and shield gates remain attached to the FTC, (4) the FTC retain and confine the fuel assembly without deformation or damage that could compromise confinement, and (5) the FTC not experience gross structural failure that could adversely affect shielding integrity.

The analysis results show that the FTC maintained its overall structural configuration during all evaluated impact events. The lid and base assemblies remained engaged during impact and rebound, calculated strains in the primary FTC components remained below the licensee's established plastic strain limit, and the welds connecting the end plates to the lid and base assemblies remained structurally adequate. In addition, the NRC staff verified during the audit that the lid-to-base bolts remained structurally adequate, with safety factors greater than 1.0 and combined loading interaction ratios less than 1.0.

Based on the analysis results, the NRC staff finds that the FTC does not experience gross structural failure, the lid remains attached to the base assembly, the lid-to-base bolts do not experience gross failure, and the end plates and shield gates remain attached following the postulated drop events. The staff further finds that the FTC maintains its fuel retention, confinement, and shielding functions. Therefore, the NRC staff concludes that the licensee has demonstrated that the FTC satisfies the acceptance criteria for the evaluated drop accident and will continue to perform its intended safety functions under the postulated drop conditions.

3.5 Fuel Bundle DD41 Integrity During Planned Move

The NRC staff requested confirmation from the licensee that the integrity of fuel assembly DD41 had been assessed to ensure that the planned lift from the reactor vessel could be performed without compromising fuel rod bundle integrity or causing additional damage. In its October 2, 2026, supplement, the licensee stated that a qualitative assessment of the assembly's condition was performed using underwater camera systems. This provided direct visual verification of the assembly's overall structural condition, including any observable deformation. The licensee further indicated that reactor coolant chemistry sampling was completed and demonstrated no indications of fuel rod cladding failure. The NRC staff finds that chemistry sampling is an appropriate method for detecting cladding breach, and in combination with the visual examination, provides reasonable assurance that the assembly's condition prior to movement, while damaged, remains structurally intact.

In the LAR, the licensee described the design of the fuel movement device intended for use in lifting assembly DD41. In its October 2, 2026, supplement, the licensee stated that the device was fabricated taking into account information from their qualitative assessment as well as the observed bowing and deflection of the upper tie plate. The device is designed to provide positive control of the assembly via the structural guide bars of the assembly in alignment with the grid straps. The staff notes that freeing DD41 from its current pinned location could change conditions with respect to the degree of bowing and potentially other geometric or structural properties of the fuel assembly.

The NRC staff requested clarification regarding the potential for further damage to assembly DD41 when placed into the fuel transfer canister, given the significant bowing identified, and given that the FTC drawings, as examined on the licensee portal, appeared not to accommodate the observed degree of deformation. One staff concern was the potential for the canister lid installation to force the assembly into a straighter configuration, potentially damaging fuel rods, grid straps, or guide bar weld connections. In its October 8, 2026, supplement, the licensee explained that, once DD41 is positioned on the FTC pedestals, the extent of bowing and plastic deformation will be confirmed using cameras before lowering the canister lid. The staff finds that this visual confirmation step provides an additional verification point to ensure the assembly is within the acceptable envelope for canister loading. In the October 12, 2026, supplement, the licensee clarified the acceptance criteria based on the visual observations. First, assembly DD41 will be confirmed to be within the bounds of the FTC alignment tool, which ensures that the assembly is aligned in the horizontal plane. Second, none of the gaps between the assembly and the FTC pedestals shall be greater than 2 inches. Assembly DD41 seating on the FTC pedestals will be verified from a horizontal view using markings on the alignment tool that represent 2 inches from the pedestals. Based on an evaluation by the fuel vendor, the predicted plastic deformation should not exceed 3 inches from the horizontal and the largest gaps between DD41 and the pedestals are expected to exist at the top end of the assembly. The licensee stated that, if any gaps are determined to be greater than 2 inches, work will stop and the site will re-evaluate.

The licensee stated that their fuel vendor evaluated the expected response of DD41 under the mechanical constraints of the FTC and concluded that the assembly would straighten during lid installation without risk of cladding failure. The NRC staff notes that the licensee acknowledged the potential for grid-to-guide bar welds to experience localized axial tension or compression as the guide bars realign. According to the vendor evaluation, any resulting grid cell distortion would not challenge fuel rod integrity, provided the rods remain within the envelope defined by the inner surface of the FTC lid. In the October 12, 2026, supplement, the licensee stated that

based on the fuel vendor evaluation, during lowering of the FTC lid, it is expected that the non-fuel hardware (upper tie plate, guide bars, and spacer grids) will absorb any undesired forces and strain externally placed on the assembly.

The staff accessed and considered the licensee-provided fuel handling procedure (FHSO-19A, Recovery of Fuel Assembly DD41, Revision 2) on their electronic portal to gain an understanding of this first of a kind fuel movement process. Regarding contingencies, in the October 8, 2026, supplement, the licensee stated that the site's ALARA plan includes immediate post-closure water chemistry sampling and radiological measurements, as well as stop-work criteria should adverse conditions be identified. In the October 12, 2026, supplement, the licensee clarified that portable continuous air monitors may alarm during the lift phase as the FTC is moved from below the water surface to above the water surface and that RP personnel will monitor detector response throughout canister operations and evaluate alarm conditions based on detector behavior. The NRC staff finds that these measures constitute appropriate controls to identify and respond to any unexpected degradation that might occur during FTC lid installation. Additional details are provided above in section 3.3 of this SE.

In addition, as described in Section 3.2.6 of the LAR, additional degradation of the fuel assembly is one of the radiological contingencies considered for the entirety of the fuel movement from vessel to spent fuel pool, with recovery actions governed by applicable engineering limits and approved procedures, with priority placed on maintaining fuel cooling, radiation shielding, contamination control, and personnel protection.

Based on its review of the information above, the NRC staff concludes that the licensee has implemented reasonable precautions to address the potential for additional damage to fuel assembly DD41 during this first of a kind evolution. The staff finds that the licensee has established adequate contingency measures to address any such damage.

4.0 EXIGENT SITUATION

Background

The NRC's regulations contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. These provisions are applicable under exigent circumstances. Consistent with the requirements in 10 CFR 50.91(a)(6), exigent circumstances exist when: (1) a licensee and the NRC must act quickly; (2) time does not permit the NRC to publish a Federal Register notice allowing 30 days for prior public comment; and (3) the NRC determines that the amendment involves no significant hazards consideration.

Under the provisions in 10 CFR 50.91 (a)(6), the NRC notifies the public in one of two ways: (1) by issuing a Federal Register notice providing an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comments; or (2) by using local media to provide reasonable notice to the public in the area surrounding the licensee's facility. In this case, the NRC published notices in the Grand Rapids Press and the Kalamazoo Gazette (October 6 and 8, 2026), and the Herald Palladium (October 7 and 8, 2026).

The licensee provided the following basis to support the finding:

Tilting of fuel assembly DD41 was not an intended outcome of core loading and therefore PNP could not have foreseen the need for a license amendment to revise the UFSAR-described fuel-handling process and associated fuel-handling accident

assumptions that would support a routine licensing amendment request submission schedule.

PNP is requesting emergency processing of this license amendment request as a delay in approval of the proposed changes would result in a delay in the resumption of activities necessary to reach the plant conditions required to restore the plant to power operation.

10 CFR 50.91(a)(7) states "Where the Commission finds that significant hazards considerations are involved, it will issue a Federal Register notice providing an opportunity for a prior hearing even in an emergency situation, unless it finds an imminent danger to the health or safety of the public." PNP concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified, as discussed in Section 4.2 [of the LAR].

NRC Staff Conclusion

As discussed in the licensee's application dated September 30, 2026, the licensee requested that the proposed amendment be processed by the NRC on an emergency basis. Although the licensee requested that the NRC process the license amendment request under emergency circumstances under 10 CFR 50.91(a)(5), which allows the NRC to issue a license amendment involving no significant hazards considerations (NSHC) without prior notice and opportunity for a hearing or for public comment, the NRC staff determined that there was sufficient time to provide the public a reasonable opportunity for the public to comment and issue the amendment based on exigent circumstances under 10 CFR 50.91(a)(6). The NRC staff found that it must act quickly on the amendment request because PNP is currently loading fuel as part of its efforts to return the plant to operation, and failure to approve the amendment in an expedited manner, would prevent continuation of core loading activities and consequently preclude the unit's timely return to power operation. The NRC staff further finds that the licensee used its best efforts to make a timely application. As stated by licensee, the tilting of fuel assembly DD41 was not an intended outcome of core loading, so the licensee could not have foreseen the need and basis for this amendment.

Based on these findings, and the determination that the amendment involves no significant hazards consideration as discussed below, the NRC staff concludes that a valid need exists to issue the license amendment under the exigent provisions of 10 CFR 50.91(a)(6).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

Under the Atomic Energy Act of 1954, as amended, and the NRC's regulations, the NRC staff may issue and make an amendment immediately effective, notwithstanding the pendency before the Commission of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has made a final determination that no significant hazards consideration is involved.

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously

evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the NRC staff proposed its analysis of the issue of no significant hazards consideration, which is presented below:

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

Response: No.

Operation of PNP in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Although the DD41 proposed fuel assembly restoration path differs from routine fuel handling practices due to the development of the Fuel Transfer Canister (FTC) and lifting system, the licensee evaluated the use of the Fuel Movement Device (FMD) and the FTC and determined the fuel handling methods and administrative controls provide equivalent protection to the routine methods. There is no significant increase in the probability of an accident. In addition, the licensee evaluated the potential dose consequences of an accident during DD41 recovery and found there is no significant increase in the consequences of an accident as compared to existing safety analyses.

The licensee's evaluation provided with the September 30, 2026, application documents the radiological consequences of a postulated Fuel Handling Accident (FHA) in accordance with Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Designs Basis Accidents at Nuclear Power Reactors," Revision 0 (ML003716792), pursuant to Section 50.67, "Accident source term," of Title 10 of the Code of Federal Regulations (10 CFR). The analysis concluded the radiological dose to the Control Room, Exclusion Area Boundary, and Low Population Zone would be within regulatory limits.

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

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

Response: No.

The proposed amendment will not create the possibility of a new or different kind of accident from any previously analyzed. An accident previously evaluated in the UFSAR is a fuel handling accident where a single fuel assembly is dropped during handling. The recovery of DD41 does not create a new or different kind of accident because, although the process and travel path are different from routine fuel handling evolutions, the overall accident progression previously evaluated remains the same. The proposed activity does not introduce potential damage to more than a single fuel assembly, and the accident remains bound by the existing FHA described in UFSAR Section 14.19.

Similarly, the loaded FTC movements within the fuel handling building follow

previously approved designated safe load paths using a single failure proof crane.

The fuel handling activity was evaluated for the use of a FTC to move DD41 in a dry condition. The evaluation determined that moving the fuel dry did not create the possibility of a new or different kind of accident, and the accident scenario remains bounded by the existing UFSAR Section 14.19 FHA.

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

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

Response: No.

The proposed amendment will not involve a significant reduction in a margin of safety. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed amendment would modify the PNP licensing basis to allow for a one-time use fuel handling activity for recovery of DD41. A postulated accident involving DD41 remains bounded by the UFSAR Section 14.19 FHA analysis of record.

Based on results of the thermal and fuel oxidation evaluations, the analyzed fuel transfer operation maintains the fuel cladding temperature below the applicable limit and restricts uranium dioxide oxidation to acceptable levels that maintain fuel cladding integrity during the recovery of DD41.

There will be no effect on plant systems necessary to perform protection functions. No instrument setpoints or system response times are affected and none of the acceptance criteria for any accident analysis will be changed. Consequently, the proposed change will have no impact on the radiological consequences of a design basis accident.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration determination. Based on this review, and the staff's evaluation of the underlying LAR, as discussed above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 DISPOSITION OF PUBLIC COMMENTS

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways: (1) by issuing a *Federal Register* notice providing an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comments; or (2) by using local media to provide reasonable notice to the public to provide comments in the area surrounding the licensee's facility. In this case, the NRC published notices in the Grand Rapids Press and the Kalamazoo Gazette (October 6 and 8, 2026), and the Herald Palladium (October 7 and 8,

2026). The notice provided a 3-day period for public comment on the proposed no significant hazards consideration (NSHC) determination.

The NRC staff received sets of comments from 20 individuals with one individual providing three sets of comments during this period, with most sets containing more than one comment. Only three comments, both from the same individual, directly challenged the staff's proposed NSHC determination. For the first two, the individual summarized their comments at the end of the document by stating:

Taken together, these [two] comments question whether the NRC staff has established an adequate basis for its preliminary finding of No Significant Hazards Consideration.

Comment 1 remains the overriding concern: an abnormal event occurred during routine fuel handling, and the proposed recovery now relies upon equipment, personnel, procedures, and administrative controls whose applicability and adequacy for this substantially different operation must be independently established.

.....

Comment [2] establishes that acceptable calculated radiological consequences do not, by themselves, resolve questions concerning Control Room protection, recovery-personnel exposure, or the adequacy of accident-prevention measures.

The NRC should not interpret compliance with bounding radiological accident-dose criteria as resolving the distinct questions of accident probability, equipment and administrative-control reliability, the possibility of new or different accidents, or margins of safety.

Nor should the desire to avoid restart delays displace the engineering rigor, personnel preparation, and regulatory oversight appropriate to a first-of-a-kind recovery involving damaged irradiated fuel.

The central issue is not simply whether the consequences of a postulated accident remain within regulatory limits. It is whether the NRC has independently established that the proposed recovery can be performed with adequate controls to prevent the accident in the first place.

I therefore request that the NRC address these [two] comments in its final evaluation, document the technical and regulatory bases for its conclusions, and determine whether all three findings required by 10 CFR 50.92(c) are adequately supported before issuing the proposed license amendment.

In addressing both comments, the NRC staff has performed an independent review of the licensee's plan to recover fuel assembly DD41 and return to the SFP, as described in the LAR and supplements and found the proposed changes acceptable, as documented in this safety evaluation. This review included reviewing the equipment being used, as well as procedures to perform the evolution. This review also included a very detailed review of the dose consequences associated with the licensee's proposed plan, including performing the use of independent models and confirmatory calculations, and found the potential dose consequences from a fuel handling accident to be within regulatory limits and the characterization of the

radiological condition expected for the workers during the evolution comparable to those developed by the NRC staff.

The third comment by the individual states that the NRC's RAI-3, issued on October 7, 2026, identifies a potential new mechanical failure mechanism. The individual further states that the licensee's response to the question acknowledges the deformation but provides no supporting engineering analysis and that the NRC should not accept this statement alone as sufficient. On October 9, 2026, the NRC staff issued additional questions on this topic. By letter dated October 12, 2026, the licensee provided its response. Section 3.5 of the staff's safety evaluation evaluates the issues raised in this comment.

The remaining comments were out of scope, such as requesting more time to comment, stated that this should not be done as an emergency amendment, feels the NRC is rushing the review, a lack of public awareness, general disagreement with the application, or Palisades starting up in general.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment on October 9, 2026. The Michigan State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

This action relates to authorizations under 10 CFR Part 50 with respect to installation or use of a facility component. The NRC staff has determined that any ground disturbance is limited to previously disturbed areas. Additionally, the NRC staff has determined that the action involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, no significant increase in individual or cumulative public or occupational radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. Finally, the NRC staff has determined that a categorical exclusion applies and that special circumstances under 10 CFR 51.22, "Categorical exclusions," are not present that would preclude reliance on the categorical exclusion. Accordingly, this action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(d)(8). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared for this action in connection with the action.

9.0 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.

Date of issuance: October 13, 2026