



PNP 2026-052

September 30, 2026

10 CFR 50.90

10 CFR 50.91

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

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

Subject: Emergency License Amendment Request: One-Time-Use Amendment to Allow
Alternative Fuel Handling Method for the Recovery of Fuel Assembly DD41

Pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.90 and 10 CFR 50.91(a)(5) Palisades Energy, LLC (Palisades Energy) hereby requests a one-time emergency change to the Renewed Facility Operating License (RFOL) for Palisades Nuclear Plant (PNP). The license amendment request (LAR) proposes to revise UFSAR Sections 9.11 *Fuel Handling and Storage Systems*, and 14.19 *Fuel Handling Accident*, to facilitate transport of a fuel assembly from the reactor utilizing different plant equipment and following a different travel path than is described within the UFSAR.

Palisades entered Mode 6 (Refueling) and began loading fuel into the reactor vessel on August 30, 2026. During core reload, fuel assembly DD41 (DD41) was placed into core location H-1 and tilted over after being released by the refueling machine. 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.

The proposed UFSAR changes would be effective for the duration of the recovery of the DD41 fuel assembly including its safe retrieval from its current orientation in the reactor vessel, packaging it within an engineered transfer canister, transporting the canister from the containment building to the spent fuel handling building, and placement of the canister and fuel assembly into the cask loading area within the spent fuel pool.

Emergency approval of the proposed licensing basis change is justified to restore the DD41 fuel assembly to an appropriate location in the spent fuel pool in an expedited manner. Therefore, Palisades Energy requests an approval date of October 2, 2026, for this LAR.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a), *Notice for public comment*, subparagraph (5), using the standards in 10 CFR 50.92, *Issuance of amendment*, paragraph (c), and it has been determined that the changes involve no significant hazards consideration. The basis for this determination is included in the Enclosure.

In accordance with 10 CFR 50.91(b), *State consultation*, Holtec Palisades is notifying the State of Michigan of this request by transmitting a copy of this letter, with its enclosure, to the designated State of Michigan official.

This letter contains no new regulatory commitments and no revisions to existing regulatory commitments.

If you have any questions regarding this submittal, please contact Kyle Steffic, Regulatory Assurance Manager, at (269) 290-8915.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Michael
Schultheis



Digitally signed by Michael Schultheis
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Date: 2026.09.30 23:00:53 -0400

Michael Schultheis
Site Vice President
Holtec Palisades

Enclosure: Description and Evaluation of Proposed Changes

Enclosure Attachment: Radiological Consequences of DD41 FHA at Palisades

cc: NRC Region III Regional Administrator
NRC Senior Resident Inspector - Palisades Nuclear Plant
NRC Project Manager - Palisades Nuclear Plant
Designated Michigan State Official

Enclosure

PNP 2026-052

Description and Evaluation of Proposed Changes

BACKGROUND

During core reload, fuel assembly DD41 (DD41) was placed into core location H-1 and tilted over after being released by the refueling machine. DD41 is currently oriented at an approximate 30-degree angle from the core plate, with the lower tie plate (LTP) in core location H-1 and the upper tie plate (UTP) leaning against the core shroud on the opposite side of the core. The fuel rods in DD41 retained their integrity as primary fission product barrier. None of the other fuel assemblies in the reactor core were impacted by DD41.

DD41 is currently stable and no further movement has been detected. Compensatory measures have been taken to stabilize the assembly to keep it from further movement, and the refueling team has DD41 under continuous camera surveillance. The recovery evaluation determined that damage to the DD41 UTP prevents use of the refueling machine to grapple DD41 in the manner described in Updated Final Safety Analysis Report (UFSAR) Section 9.11, *Fuel Handling and Storage Systems (Section 9.11)*.

1 SUMMARY DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), 50.90, *Application for amendment of license, construction permit, or early site permit*, Palisades Energy, LLC (Palisades Energy), hereby requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) for a one-time use emergency revision to the Palisades Nuclear Plant (PNP) licensing basis to facilitate removal of the DD41 fuel assembly from the reactor vessel. This LAR proposes no Technical Specification (TS) changes; however, Attachment 1 contains proposed revisions to UFSAR Section 9.11, and UFSAR Section 14.19, *Fuel Handling Accident (Section 14.19)*, to define an alternative fuel handling method.

This change would only be effective for the duration of the recovery of DD41 including its safe retrieval from its current orientation, packaging it within an engineered Fuel Transfer Canister (FTC), transporting the canister from containment, and placement of the canister into the cask loading area within the spent fuel pool.

Emergency approval of the proposed licensing basis change is justified to restore DD41 to an appropriate location in the spent fuel pool in an expedited manner. Therefore, Palisades Energy requests an approval date of October 2, 2026, for this license amendment request.

To facilitate recovery of DD41, Holtec Manufacturing engineered and fabricated a custom recovery tool capable of engaging the fuel assembly and an engineered FTC to support and enclose DD41 during transit from the reactor vessel to the spent fuel pool.

A detailed description and technical evaluation of the licensing basis requirements and the proposed changes are provided in Sections 2 and 3 of this LAR.

The LAR addresses the applicable evaluation in Section 4, including applicable regulatory requirements, and the No Significant Hazards consideration.

Environmental considerations are discussed in Section 5. It was determined that no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

2 DETAILED DESCRIPTION

Palisades Energy proposes, on a one-time basis to revise the UFSAR Sections 9.11 and 14.19 to describe the DD41 recovery process due to the deviation from the approved licensing basis that covers routine fuel movements. The approach and considerations for this process are described in further detail in Section 2.3 of the LAR.

The UFSAR proposed changes are described in this Enclosure. The Technical Specifications (TS) and TS Bases are not affected and do not require revision to support DD41 recovery actions.

2.1 Current Licensing Basis Requirements

2.1.1 UFSAR Section 9.11, *Fuel Handling and Storage Systems*

UFSAR Section 9.11 of the PNP UFSAR describes the operation and requirements of the handling and storage system for new and spent fuel. This section 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 which will remain unchanged by this proposed amendment.

UFSAR Section 9.11.3.5, *Radiological Considerations*, discusses radiological conditions during refueling or fuel inspection activities and of the SFP. Personnel performing refueling or fuel inspection activities are protected from excessive radiation levels by water shielding. Fuel assemblies are maintained under at least 10 feet of water and individual fuel rods are maintained under at least 8 feet of water. Equipment is designed and operated such that it is not possible to lift irradiated fuel above the 10 feet submergence limit. Protection against accidental radiation release from irradiated fuel is provided by the containment ventilation system and isolation capability, if required, of the SFP and auxiliary building ventilation system. The submergence of an assembly in 10 feet of water results in the dilution and partial retention of released fission products by the pool water.

UFSAR Section 9.11.4, *Fuel Handling System*, describes the typical refueling process where the fuel handling system transfers the fuel assemblies between the refueling cavity and the fuel storage pool on fixed rails through a transfer tube. 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.

2.1.2 UFSAR Section 14.19 *Fuel Handling Accident*

UFSAR Section 14.19 describes the remote possibility of an accident with significant consequences during refueling. Physical limitations imposed upon refueling operations and administrative restrictions on refueling procedures reduce the possibility of an accident.

Fuel handling hoists and manipulators are designed so that it is not possible to raise fuel bundles above a position which provides the minimum water shield requirements. This constraint applies in fuel handling areas inside containment and in the SFP area. In addition to these safeguards, direct radiation monitors at fuel handling areas give operating personnel

audible and visual warning of high radiation levels. Interlocks are also provided to prevent tilt machine rotation during insertion and removal of spent fuel assemblies. Fuel storage pool integrity is assured by designing the pool and storage racks as Class 1 structures.

Per Section 14.19.3, *Radiological Consequences*, PNP is currently licensed for a fuel handling accident of a dropped assembly during handling. The analysis postulates the assembly is damaged to the extent that a number of fuel rods fail. A part of the licensing basis is an independent calculation and assessment performed by the NRC which assumed all fuel rods equivalent of an entire assembly failure of 216 rods occurred in comparison of the Consumer Power Company's analysis assumption that only 13 rods would fail in an outer row. The results of the analysis of a dropped fuel assembly in the reactor core provides data that the energy absorption capability of the fuel assemblies indicate that an assembly dropped in such manner is capable of absorbing the kinetic energy of the drop without causing any fuel rod failures.

2.1.3 Technical Specifications

The PNP TS and associated Bases were reviewed to determine whether the proposed alternative fuel handling process requires changes. The review considered TS applicability during MODE 6, CORE ALTERATIONS, movement of irradiated fuel assemblies, movement of a fuel cask in or over the SFP, and storage of irradiated fuel in the SFP.

The review determined that no changes to the TS requirements are necessary. All applicable Limiting Conditions for Operation, Required Actions, Surveillance Requirements, Design Features, and Administrative Controls will be satisfied during the proposed alternative fuel handling evolution.

The proposed activity does, however, change certain assumptions and descriptions associated with the existing fuel handling safety analysis in UFSAR Sections 9.11 and 14.19. In particular, the current licensing basis describes irradiated fuel as remaining underwater during transfer and credits the surrounding water for shielding, cooling, and retention of released iodine. The proposed alternative fuel handling process uses an engineered FTC with associated approved procedure controls during portions of the proposed activity in which the damaged fuel assembly is not submerged. The corresponding FTC safety risk analysis demonstrates that applicable radiological, criticality, thermal, structural, and fuel handling safety criteria continue to be satisfied for the alternative method.

TS reviewed for applicability and determined to be impacted by the proposed changes to UFSAR Sections 9.11 and 14.19 included the following TS:

- 3.3.6, "Refueling Containment High Radiation (CHR) Instrumentation"
- 3.7.10, "Control Room Ventilation (CRV) Filtration"
- 3.7.11, "Control Room Ventilation (CRV) Cooling"
- 3.7.12, "Fuel Handling Area Ventilation System"
- 3.7.14, "Spent Fuel Pool (SFP) Water Level"
- 3.7.15, "Spent Fuel Pool (SFP) Boron Concentration"
- 3.7.16, "Spent Fuel Pool Storage"

- 3.9.3, "Containment Penetrations"
- 3.9.6, "Refueling Cavity Water Level"
- 4.3, "Fuel Storage"

The revised safety analysis does not alter the required operability, applicability, or actions associated with these TS. Conforming changes to affected TS Bases are not appropriate due to this being one-time operation limited to recovery of DD41.

UFSAR Section 9.11.3.2 currently identifies the Region 2 cask laydown area as an area that may be used for placement of spent fuel shipping casks, dry fuel storage equipment, and fuel inspection and repair equipment. The FTC containing DD41 will be temporarily staged in this area as part of the approved recovery evolution. The revised UFSAR will specifically describe this temporary configuration and the associated criticality, thermal, structural, radiological, and load handling analyses.

Following transfer to the SFP, DD41 will be removed from the FTC and placed into an approved storage configuration meeting the applicable requirements of TS 3.7.16 and TS 4.3 before subsequent dry fuel storage activities.

2.2 Reason for the Proposed Change

Due to the conditions described in the Background section, DD41 cannot be recovered using the refueling machine, and relocating DD41 from the reactor vessel to the spent fuel pool will require a process not described in the UFSAR. PNP proposes a revision to UFSAR Sections 9.11 and 14.19 to implement a one-time alternative process that would relocate DD41 from the reactor vessel to the SFP, where subsequent handling and storage can be performed using the fuel-handling process described in the UFSAR.

The proposed activity changes the UFSAR-described fuel-handling process and associated fuel-handling accident assumptions. UFSAR Section 9.11.3.5.1 describes personnel performing refueling or fuel inspection activities as protected from excessive radiation levels by maintaining fuel assemblies under at least 10 feet of water. UFSAR Section 9.11.4.1 describes fuel assemblies as being handled underwater at all times using the refueling machine, tilt machines, transfer carriage and rails, and spent fuel handling machine. UFSAR Section 14.19 assumes adequate fuel cooling is provided by the surrounding water and states that fuel is not removed from the water during transfer from the reactor core to its storage location. Section 14.19 further assumes that fuel-gap activity from damaged fuel is released to refueling water and credits partial iodine retention in that water. UFSAR Table 14.19-1 assumes a minimum of 22.5 feet of water above the damaged fuel assembly. Section 14.19.1 further states that fuel-handling hoists and manipulators are designed such that fuel assemblies cannot be raised above the position necessary to maintain the minimum water-shield requirement.

The alternative recovery process departs from these assumptions by transferring DD41 above water within a drained FTC using the containment polar crane, purpose-designed rigging, a rail transfer system through the equipment hatch, and the SFP building crane. The revised safety analysis therefore establishes the safety basis for the alternative recovery process by evaluating

radiological consequences, decay-heat removal and fuel temperatures, criticality, structural integrity, shielding, and postulated load drops and lifting-system failures.

PNP applied the 10 CFR 50.59 process to evaluate the proposed process for DD41 recovery and determined that prior NRC approval is required to proceed with the proposed use of the FTC alternative fuel-handling process.

2.3 Description of the Proposed Change

PNP proposes 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 DD41. Once recovery of DD41 is completed, the changes would transition to historical UFSAR content.

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 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.

2.3.1 Summary of UFSAR Revised Text

All revisions to the UFSAR include a note that designates the language as historical UFSAR content following the recovery of fuel assembly DD41.

Section 9.11.4 would be revised to include the following paragraphs:

The Fuel Handling System has an additional transfer method for transferring damaged fuel assemblies that cannot be transferred using the normal process through the transfer tube. The additional transfer method is composed of an engineered Fuel Movement Device (FMD) used to engage and remove a damaged fuel assembly from the reactor core by lifting it and placing it into an engineered Fuel Transfer Canister (FTC) staged at the bottom of the refueling cavity away from the reactor flange. The FMD is attached to the polar crane with rigging components along with dynamometers to measure the physical force of the suspended load that provides live accurate measurements during the lift to operators. Once the FTC is assembled and ready to be lifted out of the water, the FTC with the assembly inside is lifted to the containment trolley system at the equipment hatch. Once loaded on the trolley, the FTC is moved out of the containment building through the equipment hatch into the spent fuel pool building.

8. *Fuel Movement Device*

The Fuel Movement Device (FMD) is designed and fabricated to be submerged and operated remotely to safely grasp and lift the tilted fuel assembly from the reactor core. The FMD is rigged to the containment polar crane, L-1, for retrieval and subsequent movement to the engineered transfer canister. The FMD is classified as a non-safety related component to be used only for tilted fuel assembly DD41 recovery. The lifting operation utilizing the FMD, with the fuel assembly attached, occurs entirely underwater always maintaining at least 10 feet of water above the assembly. The allowable stress limits used in the design and analysis are per ASME Section III, Subsection NF for Class 3 linear supports. The calculated stresses in the FMD under normal lifting and handling conditions are compared against applicable Level A allowable stress limits. [Reference 1]

The FMD has a total of 12 lift arms. Calculations conservatively assume that only 8 of 12 lift arms are in contact with the fuel assembly during the lift (i.e., minimum of 4 lift arms at top and bottom ends with 2 lift arms in contact on each side of the fuel assembly).

For purposes of defining the upper limit of the radiological consequences of a fuel-handling accident, Section 14.19 assumed that all fuel rods in the affected fuel assembly fail. Accordingly, the radiological consequences of fuel rod damage resulting from FMD fuel-handling accidents are bounded by the existing Section 14.19 fuel-handling accident analysis.

9. *Fuel Transfer Canister*

The Fuel Transfer Canister (FTC) is an engineered Holtec designed enclosure fabricated to accommodate fuel assembly DD41. The enclosure fully protects the damaged assembly during transport. The enclosure has been analyzed for dropped load impact and radiation risks. [Reference 2]

A shielding analysis calculation concludes that the enclosure will maintain dose rates below target rates of 500 mrem/hr on contact for the most limiting surface of the FTC.

Section 14.19.1 would be revised to include the following paragraphs:

Due to the need to recover fuel assembly DD41 which tilted over after it was released by the fuel-handling machine grapple during refueling activities, additional consideration associated with activities related to tilted fuel assemblies is necessary to restore Palisades Unit 1 back to normal plant operations.

Visual inspections of the tilted assembly concluded that damage sustained by the fuel assembly prevents retrieval using the grapple on the refueling machine. The specific conditions of the DD41 fuel assembly are that only assembly frame damage was sustained and that the primary fission product barrier of fuel cladding is still intact. The described condition remains bounded by the existing fuel handling accident described in Section 14.19.3. A retrieval process was developed to use specially designed and engineered recovery components described in Section 9.11 to retrieve the tilted fuel assembly.

The retrieval process includes steps where the fuel assembly is transported out of the shielding water barrier protection. Transporting the assembly in an engineered fuel transfer canister has been analyzed against existing design and regulatory requirements by an engineering design package that addresses the risks and safety concerns. The various analyses performed to support the transport conclude that transporting through the containment hatch and into the fuel handling building is safe and does not violate any existing dose, seismic, Technical Specifications, UFSAR, Technical Specifications Bases, or any other design basis documents that currently make up the regulatory structure of the Palisades plant that are part of or referenced in the Renewed Facility Operating License (RFOL) or any of its conditions.

Therefore, the recovery scheme to mitigate consequences and recover a damaged fuel assembly, that cannot be handled with the refueling machine grapple, has been determined to be within the regulatory framework of the current RFOL.

2.4 Basis for Emergency Processing

In 10 CFR 50.91(a)(5), the NRC refers to “emergency situations” as those in which failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant’s licensed power level. Under such a situation, the Commission may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or public comment.

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.

3 TECHNICAL EVALUATION

3.1 Recovery Equipment Description

3.1.1 Fuel Movement Device

PNP intends to attach an FMD to DD41 for stabilization and provide an alternative rigging attachment point. The FMD enables use of L-1 in combination with two independent chain falls to lift the FMD holding DD41 for removal from the reactor cavity and placement into the FTC.

The FMD clasps around and reinforces DD41 at the grid straps, in contact with the guide bars, forming a structural spine along the assembly. End clasps secure the top and bottom of DD41 to prevent it from slipping from the FMD during lifting and handling.

The FMD lifts a single fuel assembly and is considered a "heavy load" per the definition in Section 1.2 of NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* because the combined weight of the FMD and DD41 is greater than the combined weight of a fuel assembly and its associated handling tool at PNP (1300 lbs., Operating Requirements Manual Section 3.21). The lifting operation using the FMD would be performed underwater, with DD41 maintained at least 10 feet below the water surface, consistent with the radiation shielding provisions described in UFSAR Section 9.11.3, *Spent Fuel Storage*.

The purpose of this calculation is to evaluate the structural integrity of the Fuel Movement Device (FMD). The primary input data for stress analysis are dimensional data, weight data, and construction materials, as well as fuel assembly geometry and weights. All calculated safety factors are greater than 1.0, indicating that the stress criteria for ASME III Subsection NF for Class 3 linear supports are satisfied.

3.1.2 Fuel Transfer Canister

Structural Evaluation

The applicable 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. Calculated stresses include the specified dynamic load factor and are compared with the allowable stresses applicable to the material, temperature, component, and stress category. The ultimate-strength design factor of 10 specified in NUREG-0612, Section 5.1.6(3)(b), for interfacing lift points was adopted for the combined static and dynamic lifting load.

Associated analysis and calculations performed are documented in HOLTEC Report HI-2261319 Rev. 1, *Structural Evaluation of Fuel Transfer Canister for Palisades Nuclear Plant*. [Reference 3] Per HI-2261319 Rev. 1, the stress criteria are satisfied, and therefore the FTC is determined to be structurally adequate for its intended purpose.

Reference 3 evaluated the structural integrity of the FTC and demonstrated compliance with the specified allowable stress limits. The stress analysis used static equilibrium and strength of materials principles to evaluate potential failure modes under normal lifting and handling conditions. The calculation used dimensions and weights listed in the FTC drawing as design inputs, and concludes that all safety factors are greater than 1.0.

SFP Seismic Stability and Floor Loading Analysis

The FTC will be supported on a vertical support stand within the cask loading area of the SFP. This configuration has been evaluated for seismic stability under the safe shutdown earthquake (SSE) loading condition. The evaluation addressed the potential for overturning and sliding, and the structural adequacy of the SFP floor slab. This evaluation is documented in HOLTEC Report HI-2261282 Rev. 2, *Seismic Stability and Floor Loading Analysis of the Fuel Transfer Canister in the Spent Fuel Pool at Palisades* [Reference 2].

The seismic stability evaluation analyzed the initial overturning and sliding stability checks using force and moment equilibrium principles. Prior to performing the stability evaluation, the

dynamic characteristics of the combined FTC, vertical stand, and supporting spent fuel pool (SFP) slab are evaluated to establish the applicable spectral accelerations. The natural frequency is determined by using Roark's formulas for stress and strain, and the inputs to these calculations are existing plant floor loading evaluations and weight of the fuel, FTC, and vertical support stand. Sliding is identified as a credible failure mode, but the calculated sliding displacement remains less than the available clearance within the Cask Loading Area (CLA) [Reference 2].

Based on the results of the evaluation, the FTC and vertical stand remain stable against overturning under the SSE loading condition. Furthermore, the structural evaluation demonstrates that the calculated floor slab demands do not exceed the applicable capacities and the SFP floor slab is demonstrated to be structurally adequate to support the configuration under evaluated loading conditions.

Thermal Evaluation

Per HOLTEC Report HI-2261312 Rev. 0, *Thermal Evaluation of Fuel Transfer Operations using Fuel Transfer Canister at Palisades NPP* [Reference 4], a steady-state 3-D Computational Fluid Dynamics (CFD) model was used to perform thermal evaluations. The allowable fuel exposure duration in air was determined using the cumulative oxidation usage methodology described in the HI-STAR 80 SAR. Methodology is based on the temperature-dependent oxidation kinetics of UO_2 documented in NUREG-1565 and limits the cumulative oxidation usage to $U \leq 1.0$. The bounding fuel temperature calculated by the computational fluid dynamics analysis was used to determine the permissible duration of fuel exposure to air.

Accordingly, the fuel transfer operation is acceptable provided that the total duration for which the fuel is exposed to an oxidizing atmosphere during removal of the FTC from water, gravity drainage, transfer through air, and placement in the spent fuel pool does not exceed the prescribed duration of 133 hours. Based on the results of the thermal and fuel oxidation evaluations, the analyzed fuel transfer operation maintains the fuel cladding temperature below the applicable limit and restricts UO_2 oxidation to acceptable levels that maintain fuel cladding integrity during the recovery of DD41.

This evaluation demonstrates that the canister retains its integrity while carrying DD41. The analysis was performed using FLUENT code and included the effects of conduction, natural convection, and thermal radiation heat transfer. The methodology incorporated conservative modeling assumptions for the fuel assembly, FTC configuration, material properties, and heat rejection mechanisms. It has been concluded that the fuel cladding temperatures do not exceed 517°F, which is under the 752°F acceptance criteria.

Drop Analysis

Per HI-2261346 Rev. 0, *Fuel Transfer Canister Drop Analysis* [Reference 5], the structural response of the FTC was evaluated for three bounding horizontal drop conditions using LS-DYNA analyses which included: FTC pure horizontal impact, FTC horizontal drop with the top end impacting first, FTC horizontal drop with the bottom end impacting first. 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. Therefore, the bolts satisfy the applicable strength and interaction criteria.

The results from all three simulations indicate that the FTC remains structurally adequate and functionally intact under the evaluated bounding drop conditions. No calculated response was identified that would compromise the FTC structure, fuel confinement boundary, closing-gate assembly, or safe retention of the spent fuel assembly.

3.2 Assessment of Radiological Conditions

The radiological assessment considered the differences between routine underwater fuel movement and the proposed alternate recovery and transfer of fuel assembly DD41. The assessment addressed direct gamma and neutron radiation, occupational dose, airborne radioactivity, contamination, installed and supplemental radiation monitoring, ventilation and release controls, and radiological contingencies.

Detailed implementation requirements, including specific instrument locations, alarm or action levels, personnel stay times, hold points, protective clothing, respiratory protection, and contingency actions, are established in approved recovery procedures, Radiation Work Permits, and the activity-specific ALARA plan.

3.2.1 Dose Comparisons

During routine fuel movement, the fuel assembly remains underwater, and the water provides radiation shielding, cooling, neutron absorption, and retention of radioactive material. The proposed alternate recovery involves placement of DD41 in an engineered FTC and movement above the water surface for a controlled period.

The FTC provides engineered gamma and neutron shielding for the loaded fuel assembly. The shielding evaluation considers the fuel source term, burnup and decay history, canister geometry, fuel orientation, shielding thickness, penetrations, end effects, and potential radiation streaming paths.

Engineering shielding calculations used MCNP5 code to predict local radiation dose rates during transfer of the canister. 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. The design inputs for this calculation are characteristics of the DD41 fuel and non-fuel hardware, and the calculation estimates dose rates of approximately 110 mrem/hr on contact with the side of the canister, approximately 60 mrem/hr at 1 foot from the side, and less than 500 mrem/hr on contact at both ends of the canister. [Reference 8]

The activity-specific ALARA review provides the framework for evaluating projected personnel dose for the recovery activity relative to routine fuel-handling activities, including loading, draining, lifting, transfer, placement into the spent fuel pool, and credible delays or interruptions. Personnel positioning, remote handling, shielding, access control, and time near the loaded canister are addressed through the station's ALARA program and applicable radiological work controls to maintain occupational radiation exposure as low as reasonably achievable.

3.2.2 Personnel Monitoring and ALARA Controls

The activity is subject to the station's Radiation Protection and ALARA programs and is controlled through an approved Radiation Work Permit and activity-specific ALARA plan. These work controls address Radiological Protection coverage and management oversight during designated phases of the evolution.

Personnel monitoring under the station's dosimetry program includes electronic dosimetry appropriate for the expected gamma radiation field. Neutron-sensitive instrumentation and personnel neutron monitoring are included, as appropriate, to address the significant neutron contribution to the expected dose rate.

Extremity monitoring requirements are determined through the radiological work-planning process based on task geometry, personnel proximity, and the potential for localized radiation exposure. Changes in work scope or radiological conditions are evaluated for their impact on these requirements through the station's radiological work-control process.

The applicable radiological work controls address, as appropriate:

- Pre-job radiological surveys and briefings;
- Establishment of radiologically controlled boundaries;
- Personnel positioning and stay-time controls;
- Remote monitoring and remote handling, where practical;
- Verification of the transfer canister configuration, including credited shielding components, before movement;
- Defined radiological hold points;
- Continuous or frequent dose-rate monitoring during critical phases;
- Stop-work criteria for unexpected radiation levels or changing conditions; and
- Post-job dose review and documentation.

The ALARA review establishes activity-specific administrative dose limits, stay times, survey frequencies, and action levels for implementation through applicable Radiation Work Permits and work-control documents.

Access controls restrict entry to affected areas to personnel necessary to support the evolution, based on expected and measured radiological conditions.

3.2.3 Airborne Radioactivity and Contamination Controls

Available inspection and chemistry information does not indicate that DD41 currently contains failed fuel rods. The assembly's known structural damage and proposed handling outside the normal underwater fuel-handling configuration are addressed through the station's Radiation Protection and ALARA programs in evaluating the potential for airborne radioactivity and contamination.

The activity-specific radiological work-planning process addresses transfer canister draining, residual contaminated water, movement of a wet canister, leakage or spillage, external surface contamination, and immersion of the heated fuel and canister into the spent fuel pool. This evaluation includes the potential transport of radioactive material by steam or aerosols generated during immersion.

Applicable radiological work controls include, as appropriate:

- Baseline airborne-radioactivity and contamination surveys;
- Air sampling or continuous air monitoring during designated phases;
- Surveys of the transfer canister and handling equipment;
- Controls for draining, collecting, and processing contaminated water;
- Spill and leakage controls;
- Protective clothing and respiratory-protection decision criteria;
- Post-movement airborne-radioactivity and contamination surveys; and
- Defined actions for an unexpected increase in airborne activity or contamination.

Engineering evaluations of fuel temperature, allowable air-exposure duration, canister draining time, steam generation, and fuel-integrity response during immersion provide inputs to the development of activity-specific radiological controls. The approved ALARA plan and applicable Radiation Work Permits document these controls as part of pre-evolution work authorization. Changes in work scope or conditions are evaluated through the station's radiological work-control process to determine necessary revisions to the controls.

3.2.4 Area and Effluent Radiation Monitoring

Plant-installed radiation monitoring for the evolution includes the Refueling Containment area radiation monitors, RIA-2316 and RIA-2317, and the spent fuel pool area/criticality monitors, RIA-2313 and RIA-5709. Verification of monitoring coverage for the transfer path and expected radiation fields is part of the pre-evolution work-planning and authorization process.

Applicable Technical Specification requirements govern monitor operability, required surveillance, and actions for inoperable instrumentation. The activity is subject to these requirements based on the plant configuration and the monitoring functions required for the evolution.

The review addresses:

- Suitability of monitor locations and coverage for the transfer path;
- Expected radiation levels relative to the monitors' usable ranges;
- Availability of alarm functions and required automatic protective actions;
- Potential for the expected canister radiation field to produce an unintended or unanalyzed monitor response and associated resolution;
- Operability of required Technical Specification functions; and
- Supplemental temporary or portable monitoring where installed coverage is insufficient.

Portable gamma and neutron survey instruments supplement installed monitoring during the activity, with instrument selection and placement addressed in the approved radiological work controls.

Evaluation of applicable process and effluent radiation-monitoring pathways is part of the work-planning process and addresses detection and assessment of potential releases from containment or the spent fuel pool area in accordance with approved station programs. This evaluation considers the expected radionuclides, release pathways, and monitoring capabilities.

3.2.5 Ventilation and Release Controls

The recovery activity is subject to applicable requirements for containment and spent fuel pool ventilation and release controls. Verification of the required ventilation configuration, filtration capability, release pathway, and radiation-monitoring functions is part of pre-movement work authorization. Applicable Technical Specifications and approved procedures govern maintenance of these functions throughout the evolution.

The configuration review addresses the relationship between the Refueling Containment high-radiation instrumentation and associated automatic protective actions, including containment isolation and ventilation responses. Applicable Technical Specifications and approved procedures govern required system and monitoring function operability and actions for loss of required functions, including suspension of the activity and placement of the fuel in a safe position, as applicable, under the direction of Operations.

Establishment and approval of an acceptable ventilation and release-control configuration is a prerequisite to initiation of the activity.

3.2.6 Radiological Contingency Considerations

Response actions for credible abnormal radiological conditions are addressed through the station's radiological work-control and recovery planning processes. Documentation of these actions in approved procedures and work-control documents is part of pre-evolution authorization. Conditions addressed include:

- Dose rates greater than calculated or expected;
- Unexpected neutron radiation levels;
- Damage to or loss of credited transfer-canister shielding;
- Loss of required area, airborne, or effluent monitoring;
- Loss of the required ventilation or filtration configuration;
- Unexpected airborne radioactivity or contamination;
- Transfer-canister leakage or contaminated-water spillage;
- A delay or interruption with the fuel assembly above water;
- Approaching the analyzed maximum allowable air-exposure duration;
- Steam or aerosol generation greater than expected;
- Additional degradation of the fuel assembly; and
- Inability to complete the planned transfer.

Approved procedures and work-control documents identify safe holding or placement locations, required hold points, stop-work criteria, communications, supplemental monitoring, and the authority for resuming the evolution. Recovery actions are governed by applicable engineering limits and approved procedures, with priority given to maintaining fuel cooling, shielding, contamination control, and protection of personnel.

3.2.7 Offsite Dose Evaluation

This evaluation documents the conservative analysis of the radiological consequences of a postulated FHA in accordance with Regulatory Guide 1.183 Revision 0, pursuant to 10 CFR 50.67. The design inputs used are an isotropic inventory of DD41 and schematics of Palisades ventilation from the fuel handling area of the Auxiliary building to the Stack. It has been concluded that tabulated radiological dose to the Control Room, Exclusion Area Boundary

(EAB), and Low Population Zone (LPZ) are within regulatory limits as described in HI-2261348 Rev. 1, *Radiological Consequences of DD41 FHA at Palisades* [Reference 6].

3.3 Heavy Lifts Assessment

3.3.1 Lifting and Rigging Considerations

Fuel Movement Device

To ensure proper use of the FMD, the supporting analysis establishes the following administrative controls:

1. Maximum Fuel Assembly Angle

The maximum angle subtended by the longitudinal axis of the fuel and the horizontal plane shall not exceed 40 degrees maximum at any time while DD41 is being lifted or handled by the FMD.

2. Maximum Pull Force

The maximum permissible pull force that can be applied to the lifting bail on the FMD upper assembly, while attempting to free fuel assembly DD41 from its current orientation in the reactor core, shall not exceed 1,800 lbf. This limit is based on the structural capacity of the FMD, not the fuel assembly.

Once DD41 is loaded into the FTC, the containment polar crane will lift the canister along the designated safe load path described in FHSO-19A, *Recovery of FA DD41* [Reference 9] and place it onto the existing rail system for transfer through the equipment hatch to the spent fuel pool building.

Fuel Transfer Canister

The structural evaluation performed in HI-2261332 Rev. 1, *Structural Analysis of the Transfer Canister Lift Rig* [Reference 7], demonstrates that the FTC lift rig satisfies the applicable allowable stress and structural acceptance criteria for its specified normal operating conditions. The evaluated load-bearing components have adequate capacity for the intended in-plant handling of the loaded FTC.

A structural analysis of the transfer canister lift rig calculation demonstrates the load bearing capacity of the FTC lift rig. The design inputs are component material properties and maximum applied load. The evaluation demonstrated that the rig meets stress limits set by ANSI N14.6 [Reference 3].

3.3.2 Safe Load Path

The DD41 recovery procedure establishes a controlled and defined load path for movement of the FTC from the reactor cavity to the SFP building, FHSO-19A, *Recovery of FA DD41* [Reference 9]

Following recovery of DD41 from its tilted position, DD41 is maintained a minimum of 10 feet underwater while being lifted from the reactor vessel and transferred laterally to the FTC. The FMD and DD41 are moved until they are no longer directly over any portion of the reactor vessel, after which the assembly is leveled and lowered onto the bottom section of the FTC. The

procedure requires verification, using load-cell indication, that DD41 is seated and supported on the canister bottom before the FMD is disengaged.

After DD41 is positioned on the FTC bottom, the top section is lowered onto the bottom section, aligned using locating pins and corresponding lugs, and secured. The assembled and loaded FTC has an approximate lift weight of 49,000 pounds.

Operations will use the Electronic Shift Operations Management System (eSOMS) to monitor and document FTC out-of-water time in support of compliance with the analyzed 133 hour exposure limit. The out-of-water clock will begin when the FTC first breaks the water surface and end when the FTC is fully submerged in the SFP.

For transfer out of containment, the assembled FTC is raised vertically from the refueling cavity while maintained in a horizontal orientation. Following release by Radiation Protection, the FTC is maintained in horizontal orientation and transported along the defined transit path to the containment equipment hatch. The canister is then placed onto the equipment hatch trolley and transferred through the equipment hatch into the SFP building.

Once inside the SFP building, the horizontal rigging is removed, and the FTC is re-rigged for a vertical lift. The FTC is then upended to the vertical orientation and moved to the cask laydown area of the SFP and placed in the FTC vertical stand.

The safe load path therefore consists of the following segments:

1. Reactor vessel to FTC: DD41 is lifted and transferred using the FMD for placement in the FTC while maintaining at least 10 feet of water over DD41.
2. FTC assembly: DD41 is seated in the lower section of the FTC, top section is installed and secured, and the assembled FTC is prepared for transfer.
3. Containment transfer: The loaded FTC is lifted from the cavity, moved horizontally along the defined path to the equipment hatch, and placed on the equipment hatch trolley.
4. Equipment hatch transfer: The trolley is used to move the FTC through the equipment hatch into the SFP building.
5. SFP building transfer: The FTC is re-rigged, upended from a horizontal to vertical orientation, and moved to the cask laydown area of the SFP where it is secured prior to removal of rigging.

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Regulatory Requirements

10 CFR 50.91(a)(5) provides the requirements to be met to allow the NRC to perform expedited approval of a license amendment under emergency circumstances. As discussed in Section 1, Palisades Energy is requesting emergency processing of this license amendment request to restore the DD41 fuel assembly to an appropriate location in the spent fuel pool in an expedited manner. Accordingly, this license amendment request satisfies the criteria for the Commission to issue a license amendment under the emergency provisions of 10 CFR 50.91(a)(5).

The applicable regulatory requirement for radioactivity control during fuel handling evolutions is specified in 10 CFR 50, Appendix A, GDC 61, "*Fuel Storage and Handling and Radioactivity Control*", which states that *the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed:*

- (1) with a capability to permit appropriate periodic inspection and testing of components important to safety,*
- (2) with suitable shielding for radiation protection,*
- (3) with appropriate containment, confinement, and filtering systems,*
- (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and*
- (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.*

As discussed in Section 3.2, the recovery process developed by Holtec Palisades contains sufficient controls to maintain compliance with the GDC 61 requirements.

4.2 No Significant Hazards Consideration

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90 and 10 CFR 50.91(a)(5), Holtec Palisades requests a one-time emergency change to the Renewed Facility Operating License (RFOL) for Palisades Nuclear Plant Unit 1 Number DPR-20. The amendment request proposes to revise Updated Final Safety Analysis Report Sections 9.11 "Fuel Handling and Storage System," and Section 14.19 "Fuel Handling Accident" to facilitate transport of a fuel assembly from the reactor vessel through a different travel path than is currently described.

Holtec Palisades has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92(c), "Issuance of amendment," as discussed below:

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

Response: No.

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Although the proposed fuel assembly DD41 (DD41) restoration path differs from routine fuel handling practices due to the development of the FTC and lifting system, Palisades Nuclear Plant (PNP) evaluated the use of the Fuel Movement Device (FMD) and Fuel Transfer Canister (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, PNP 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.

This evaluation documents the radiological consequences of a postulated Fuel Handling Accident (FHA) in accordance with Regulatory Guide 1.183 Revision 0, pursuant to 10 CFR 50.67. Analysis concluded the radiological dose to the Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) are within regulatory limits.

Therefore, the proposed amendment does 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 previously evaluated?

Response: No.

The 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 remains the same. The proposed activity does not introduce potential damage to more than a single fuel assembly and the accident remains bounded 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; the accident scenario remains bounded by the existing UFSAR Section 14.19 FHA.

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

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

Response: No.

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 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 UO₂ 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, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, Holtec Palisades 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.

5 ENVIRONMENTAL CONSIDERATION

A review has determined the proposed change qualifies for a categorical exclusion from environmental review in accordance with 10 CFR 51.22. The proposed action is an amendment regarding the installation or use of a facility component as described in 10 CFR 51.22(d)(8). The proposed action does not disturb any previously undisturbed ground and there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, does not result in a significant increase in individual or cumulative public or occupational radiation exposure, and does not result in a significant increase in the potential for or consequences from radiological accidents. Therefore, pursuant to 10 CFR 51.22, neither an environmental assessment nor an environmental impact statement is required.

6 REFERENCES

1. HI-2261277, Stress Analysis of Fuel Movement Device for Palisades Nuclear Plant
2. HI-2261282, Seismic Stability and Floor Loading Analysis of the Fuel Transfer Canister in the Spent Fuel Pool at Palisades
3. HI-2261319, Structural Evaluation of Fuel Transfer Canister for Palisades Nuclear Plant
4. HI-2261312, Thermal Evaluation of Fuel Transfer Operations Using Fuel Transfer Canister at Palisades NPP
5. HI-2261346, Fuel Transfer Canister Drop Analysis
6. HI-2261348, Radiological Consequences of DD41 FHA at Palisades
7. HI-2261332, Structural Analysis of the Transfer Canister Lift Rig
8. HI-2261303, Dose Evaluation for the Palisades Transfer Canister
9. FHSO-19A, Recovery of FA DD41

Enclosure Attachment

PNP 2026-052

Radiological Consequences of DD41 FHA at Palisades

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- All significant assumptions are stated.
- For any analysis, the methodology is consistent with the physics of the problem.
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Revision Log

Revision	Description of Changes
0	Initial Issue.
1	Revised assumption on aerosol removal efficiency, since actual data is referenced. Credited 20-minute delay for Control Room Emergency HVAC switchover. Added discussion on comparison of consequences from existing design basis FHA to the DD41 FHA. Revision bars are used to denote changes from Rev 0



Executive Summary

The purpose of this report is to evaluate the radiological consequences of the DD41 FHA for the Palisades Nuclear Plant. The overturned fuel assembly DD41 will be recovered and lifted from the reactor core and moved to a specially designed fuel transfer canister. The evaluation addresses the period during which the fuel assembly loaded fuel transfer canister is removed from the water, drained by gravity, and transferred through the air onto a special rail cart. The purpose of this analysis is to determine the dose received by the most limiting receptors at the EAB, LPZ and Control Room due to unlikely event the fuel transfer canister is dropped during lifting process, to ensure regulatory compliance stated in 10 CFR 50.67 [1].

This assessment follows the regulatory guidance in US NRC RG 1.183 [2] and NUREG-0800 Section 15.0.3 [3]. The results of this assessment demonstrate that, in the unlikely event of an FHA, both onsite and offsite doses would remain within acceptable limits, thereby supporting the LAR submittal.

Statement of Limitations

This document has been prepared in accordance with the HPP-8002-0002 SMR-300 Program Quality Plan [4] and HI-2230815 [5].

Unverified Assumptions

There are no Unverified Assumptions used within this analysis document.

Analysis Administrative Controls

There are no Analysis Administrative Controls required from this analysis document.



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1.0 INTRODUCTION

1.1 Purpose

The purpose of this report is to evaluate the radiological consequences of the DD41 FHA for the Palisades Nuclear Plant. The overturned fuel assembly DD41 will be recovered and lifted from the reactor core and moved to a specially designed fuel transfer canister. The evaluation addresses the period during which the fuel assembly loaded fuel transfer canister is removed from the water, drained by gravity, and transferred through the air onto a special rail cart. The purpose of this analysis is to determine the dose received by the most limiting receptors at the EAB, LPZ and Control Room due to unlikely event the fuel transfer canister is dropped during lifting process, to ensure regulatory compliance stated in 10 CFR 50.67 [1].

1.2 Discussion

This analysis evaluates the radiological consequences associated with postulated DD41 FHA event in support of the LAR submittal. The analysis provides the technical basis for characterizing the potential radiological consequences of the event and supports the corresponding conclusions and claims documented in LAR.

The analysis considers the relevant characteristics and functions of SSCs that affect the transport and release of radioactive material during the FHA event. The resulting radiological consequence evaluation provides supporting evidence for the treatment of the FHA event within the LAR.

2.0 DEFINITIONS

Terms, definitions, and acronyms used in this document conform to those listed in the Company's QA Terms database, ASME NQA-1 "Quality Assurance Requirements for Nuclear Facility Applications" [6] definitions, and HI-2240905 "SMR-300 Acronyms and Glossary of Terms" [7]. Should duplicate terms or acronyms exist, ASME NQA-1 shall govern.

The following additional terms and abbreviations are included herein to facilitate the understanding of this analysis.

- **DCF** – Dose Conversion Factors
- **DE** – Dose Equivalent
- **EAB** – Exclusion Area Boundary
- **FHA** – Fuel Handling Accident
- **FHB** – Fuel Handling Building
- **HEPA** – High-Efficiency Particulate Air
- **LAR** – License Amendment Request
- **SNAP** – Symbolic Nuclear Analysis Package

3.0 METHODOLOGY

3.1 General

The analysis uses the calculation methods and assumptions presented in RG 1.183 [2] which the US NRC considers acceptable in complying with regulations for accident dose consequence analysis.



This analysis uses the RADTRAD (also known as RADTRAD-AC analytical code) computer program and the SNAP editor to prepare the model files and execute RADTRAD [8].

3.2 Accident Propagation

The evaluation addresses the period during which the fuel assembly loaded fuel transfer canister is removed from the reactor cavity water, drained by gravity, and transferred through the air onto a special rail cart. Since all of the release within the FHB would be filtered, the most limiting FHA will be a release inside the containment (with the assumption that the containment equipment hatch is open). In case of closed containment hatch, release within containment would also be filtered via pre-filters and HEPA filters prior exiting the stack, so this analysis bounds a release in containment with equipment hatch closed.

The event propagation for release path via containment equipment hatch is described step by step below.

- The DD41 fuel assembly is placed in the transport canister and lifted.
- At the time of fuel handling, the canister is postulated to drop on the containment floor.
- The drop is postulated to damage all fuel rods of the transported fuel assembly mechanically, allowing radionuclides in the fuel-clad gap to be released.
- All the gap activity in the damaged rods is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment.
- The radioactive material is released to FHB through the open containment equipment hatch.
- Radioactive material is postulated to be released into the environment via the FHB ventilation system which exhausts via the plant stack.
- The released materials from the stack become part of a plume.
- Plume traverses based on the wind speed, direction, and stability.
- The most limiting receptor at the location at the outer boundaries of EAB and LPZ absorbs radioactive material because of external exposure due to immersion in the plume and internal exposure because of inhalation of radioactive material from the plume.
- The most limiting receptors in the Control Room absorb radioactivity by internal exposure and direct shine from the atmospheric plume and filters.

3.3 RADTRAD Model

To model the radiological consequences of FHA conservatively, a RADTRAD model is developed. The simulation of the model is utilized to find TEDE for receptors at various locations. The physical representation of the RADTRAD model in the SNAP editor can be seen in Figure 3-1.

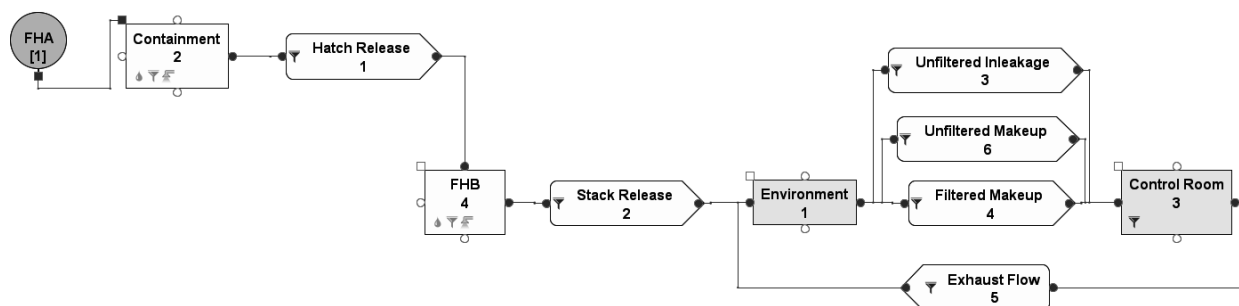


Figure 3-1 Fuel Handling Accident RADTRAD Model

Containment, FHB, Control Room and environment are modeled as compartments as shown in Figure 3-1. The transport of fission products between these compartments is modeled using pathways.

The methodology to prepare the model is described in this section. RADTRAD estimates radiological consequences at the specified dose location based on the amount, form, and species (source term) of the radioactive material released during the postulated accident. As radioactive material is transported through the compartments, the user can account for sprays, natural deposition and other retention mechanisms that may reduce the quantity of radioactive material. Material can flow between buildings, from buildings to the environment, or into control rooms through HEPA filters, piping, or other connectors. An accounting of the amount of radioactive material retained due to these tortuous pathways is maintained.

The source term input processing is described in Section 3.4. The radionuclide transport modeling is described in Section 3.5.

Using the models described, the dose analysis has been performed as described in the sections below.

3.3.1 Offsite Dose Consequences

3.3.1.1 EAB

The dose calculations have determined the TEDE received by the most limiting receptor at the most limiting EAB location using RADTRAD.

The maximum 2-hour TEDE has been determined by calculating the postulated dose for a series of small, time increments and performing a “sliding” sum over the increments for successive 2-hour periods.

3.3.1.2 LPZ

The TEDE is determined for the most limiting receptor at the outer boundary of the LPZ for the duration of the accident.

3.3.2 Onsite Dose Consequences

The TEDE is determined for the most limiting receptors in the Control Room for the duration of the accident.



3.4 Source Term Modeling

The determination of the source term for evaluating the radiological consequences of an FHA accident was performed in accordance with requirements listed in Appendix B of RG 1.183 [2]. This appendix provides assumptions which are acceptable to the staff of the US NRC.

3.4.1 Assembly Inventory

The inventory of fuel assembly DD41 provided by Palisades staff (see Ref. [9]) was analyzed and activities of isotopes of interest were derived.

Since only radionuclides in the gap are available for release, only noble gases, halogens and alkali metal isotopes are considered in the analysis. This is as per RG 1.183 Appendix B, regulatory position 1.2 [2]. The resulting inventory can be found in Section 9.1 and Appendix A.

3.4.2 Extent of Fuel Damage

The total fission product inventory available for release is determined by multiplying the radionuclide inventory in assembly with the total number (or fraction) of affected rods.

As per RG 1.183 [2] Appendix B, the number of fuel rods assumed to be damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. However, given the uncertain condition of the fuel assembly DD41, this type of analysis is not feasible. Thus, conservatively, it is assumed that all fuel rods will be damaged and the total inventory of DD41 is used for the FHA analysis.

3.4.3 Release Fractions and Timings

The radionuclide inventory for the single assembly is subsequently modified in RADTRAD through the application of radionuclide-specific release fractions and timings. Release fractions determine the portion of the release of the radionuclide groups during the accident. The release timings, as the name suggests, determine the timing of release for each radionuclide groups.

For FHA, only the radionuclides in the fuel rod void volume (plenum and pellet-to-cladding gap) are available for release due to cladding breach. Table 3 of RG 1.183 [2] lists the radionuclide groups residing in the fuel rod void volume that are available for release upon a cladding breach.

All radionuclides are assumed to be released instantaneously after initiation of the accident. This is a conservative assumption.

3.4.4 Chemical Form

The chemical form of radioiodine released from the fuel to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, as per RG 1.183 Appendix B, regulatory position 1.3 [2].

3.5 Radionuclides Transport Modeling

The transport of fission products between compartments is modeled by using flow pathways with corresponding flow rates. The mitigation of the radioactivity during the transport has been modeled by accounting for filtered pathways.



3.5.1 Filtered Pathways Modeling

Each pathway is modeled using a filter. They are described below.

3.5.1.1 Hatch Release

The filter for this pathway models the open containment equipment hatch. No mitigation is assumed for the pathway; hence filter efficiency is 0% for all chemical forms.

3.5.1.2 Stack Release

The filters for this pathway models filtered release that exits via the FHB filtration system exhausting via the plant stack. The exhaust air flows at 75000 cfm rate through the HEPA filter and a charcoal filter [10]; hence filter efficiency is 94% for elemental and organic forms as per Ref. [11], [12]. For the aerosol form – 99% removal efficiency is conservatively applied per Palisades Filter Specification [13] (See Assumption 2 in Section 6.0).

3.5.1.1 Control Room Unfiltered Makeup

The initial unfiltered makeup flow rate is 660 cfm. After control room isolation at 20 minutes, the flow is 0 cfm [10]. No mitigation is assumed for the pathway; hence filter efficiency is 0% for all chemical forms.

3.5.1.2 Control Room Filtered Makeup

The control room filtered makeup flow of 1413.6 cfm starts at 20 minutes (time of control room isolation) [10]. 99% filter efficiency is applied to all release fractions.

3.5.1.3 Control Room Unfiltered Inleakage

The unfiltered inleakage was set to 100 cfm [10]. No mitigation is assumed for the pathway; hence filter efficiency is 0% for all chemical forms.

3.5.1.4 Control Room Exhaust Flow

The control room exhaust is equal to the sum of the control room intake and inleakage flows. No mitigation is assumed for the pathway; hence filter efficiency is 0% for all chemical forms.

3.5.1.5 Control Room Recirculation Flow

The control room HVAC system is assumed to be operating in normal mode at the beginning of the event. The control room enters emergency mode at 20 minutes (manual operator action). So, 1413.6 cfm flow is applied to this filter after 20 minutes [10]. 99% filter efficiency is applied to all release fractions.

3.5.2 Atmospheric Dispersion Modeling

Atmospheric dispersion (from release point to receptor points) analysis has been performed [14]. The Atmospheric Dispersion Coefficients (X/Qs) have been determined for the receptors:

- Individual at the most limiting EAB location
- Most limiting receptor at the outer boundary of the LPZ
- Hypothetical maximum exposed individual in Control Room

The factors input can be found in Appendix A.



3.5.3 Receptors' Modeling

The critical receptors characteristics such as breathing rate and occupancy factors are input to the RADTRAD models (see Appendix A). They are chosen as per RG 1.183 [2].

4.0 ACCEPTANCE CRITERIA

To address the US NRC regulatory requirements stated in 10 CFR 50.67 [1], the calculation methodology follows RG 1.183 [2] and NUREG-0800 Section 15.0.3 [3].

The dose acceptance criteria, as per the guides, are as follows:

- EAB: Maximum dose in any 2-hour period in 720 hours (30 days) following the start of the radioactivity release ≤ 6.3 rem TEDE
- LPZ: Maximum dose in 720 hours (30 days) following the start of the radioactivity release ≤ 6.3 rem TEDE
- Control Room: Maximum dose in 720 hours (30 days) following the start of the radioactivity release ≤ 5 rem TEDE

The above requirements bound the requirements stated in 10 CFR 50.67 [1].

5.0 DESIGN INPUTS

The input data for the RADTRAD model is provided in this section.

Table 5-1 Design Inputs

Input #	Input	Reference(s)
1	Iodine Species Chemical Form: Cesium Iodide - 95% Elemental Iodine – 4.85% Organic Iodine – 0.15%	Regulatory Position 1-3, Appendix B of RG 1.183 [2]
2	Release Fractions and Timings: I-131 – 0.08 Kr-85 – 0.1 Other Noble Gases – 0.05 Other Halogens – 0.05 Alkali Metals – 0.12	Table 3 of RG 1.183 [2]
3	Breathing rates for the most limiting receptors at EAB and LPZ, and in Control Room	Sections 4.1.3 and 4.2.6 of RG 1.183 [2]
4	Control Room Occupancy Factors: 0-24 hrs - 1.0 24-96 hrs - 0.6 96-720 hrs - 0.4	Section 4.2.6, RG 1.183 [2]
5	Atmospheric Dispersion Factors	Ref. [14]
6	DD41 Fuel Assembly Inventory	Ref. [9]
7	Containment and FHB Volumes	Table 14.16-2 and Figure 1-6 of Ref. [10]
8	Control Room HVAC Parameters	Table 14.24-1 of Ref. [10]



6.0 ASSUMPTIONS

The following table identifies assumptions categorized as Reasonable (R), and Conservative (C). Reasonable and Conservative assumptions include justification explaining the basis for their categorization.

Table 6-1 Assumptions

Item (Section #)	Assumption	Type	Justification
1	It is assumed that all fuel rods will be damaged and the total inventory of DD41 is used for the FHA analysis.	C	This assumption is conservative as it maximizes amount of radionuclides available to be released
2	Aerosol removal efficiency of FHB HVAC exhaust pathway HEPA filter is limited to 99%.	C	FHB HVAC exhaust consists of two staged filtering system – first the exhaust air is diverted through a prefilter, HEPA filter and a charcoal adsorber bed, then through a filter package which contains a bank of roughing filters and a bank of HEPA filters [10]. The HEPA filters installed have aerosol removal efficiency of 99.97% per Palisades Filter Specification [13], thus this assumption is conservative.

7.0 COMPUTER PROGRAMS

Holtec International maintains an active list of QA-validated computer codes on the Company's network that are approved for use in Safety-significant projects.

Table 7-1 Computer Programs

Program	RADTRAD
Program Version	5.0.3
Generic Report #	HI-2220870-R3.0
Computer ID	HTC-NPD-1989, HTC-L-SMR-10149
Operating System & Version	Windows 11 Enterprise, Version 25H2



8.0 COMPUTER FILES

Table 8-1 Computer Files

Num.	Description	File Type	File Name	File Location
1	Excel spreadsheet containing calculation of the source term	.xlsx	DD41_isotopics	M:\Users\hhovhannisyan\Palisades FHA\Rev. 1
2	RADTRAD model for FHA	.med	Palisades_FHA	M:\Users\hhovhannisyan\Palisades FHA\Rev. 1
3	RADTRAD output file for FHA	.out	radtrad_FHA	M:\Users\hhovhannisyan\Palisades FHA\Rev. 1

9.0 ANALYSIS

9.1 Inventory

The inventory is derived from the Ref. [9]. As mentioned in Section 3.4, the isotopes that reside in fuel-clad gap are considered for release. The selection of radionuclides is based on Table 3 and Table 5 in RG 1.183 [2] and recommended nuclides in regulatory position 1.2 of RG 1.183 Appendix B [2].

The selected radionuclides list and their activity are provided in Table 9-1.

Table 9-1 DD41 Fuel Assembly Inventory

Isotope	Activity (Bq)	Activity (Ci)
Kr-85	1.38138E+14	3733.460551
I-129	474327596	0.012819665
Cs-134	7.11815E+14	19238.24497
Cs-135	10708729840	0.289425131
Cs-137	1.83948E+15	49715.712

9.2 Release Fractions and Timings

The release fractions from the fuel-clad gap are taken from Table 3 of RG 1.183 [2]. A value of 1.0E-4 hrs was input as a release duration to simulate instant release from damaged fuel rods.

Table 9-2 Release Fractions

Group	Fraction
I-131	0.08
Kr-85	0.1



Group	Fraction
Other noble gases	0.05
Other halogens	0.05
Alkali metals	0.12

9.3 TEDE Calculations

Based on the methodology described in Section 3.3, the RADTRAD calculation has been performed to find TEDE for:

- Individual at the most limiting EAB location
- Most limiting receptor at the outer boundary of the LPZ
- Hypothetical maximum exposed individual in Control Room

10.0 RESULTS AND CONCLUSIONS

Table 10-1 provides the TEDE at EAB, LPZ, and Control Room comparison to the regulatory limits. The direct shine dose contribution is the sum of the cloud and filter shine doses documented in Table 5 of Reference [15]. The DBA LOCA shine analysis is considered bounding for the FHA because the DBA LOCA model assumes release from the entire reactor core, whereas the FHA model considers damage to only one fuel assembly with a reduced radionuclide inventory. In addition, the DBA LOCA analysis includes a broader range of radionuclides than the FHA analysis. Since the DBA LOCA source term is based on a larger affected fuel inventory and includes all radionuclides considered in the FHA evaluation, the resulting DBA LOCA shine dose bounds the corresponding FHA shine dose.

Table 10-1 TEDE Results

Location	Fuel Damage TEDE (rem)	Shine TEDE (rem)	Total (rem)	Regulatory Limit (rem)
EAB	0.56437	-	0.56437	6.3
LPZ	0.069735	-	0.069735	6.3
Control Room	2.7462	0.294	3.0402	5

The results of this report's TEDE calculations seen in Table 10-1 meet the NRC regulatory requirements stated in 10 CFR 50.67 [1]. The comparison of doses expected during this particular FHA with existing worst-case design-basis FHA, documented in Table 14.1-6 of Palisades FSAR [10], demonstrate that the expected doses at EAB, LPZ and Control Room are lower (Table 10-2), even though current analysis employs two additional conservative modeling assumptions:

- No credit is given to radionuclides retention in the water in the fuel pool or reactor cavity since the drop is postulated to occur during most limiting period of evolution, when the canister is transferred through the air onto a special rail cart.
- All the gap activity in the damaged rods is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment, contrary to design-basis FHA, where the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

The decrease of expected doses for offsite (EAB and LPZ) and onsite (Control Room) receptors is driven mainly by reduced inventory of DD41 fuel assembly due to longer cooling period,



compared to a reference fuel assembly considered in design-basis FHA which assumed 2-day cooling period.

Table 10-2 Comparison with Design-Basis FHA

Location	DD41 FHA (rem)	Design-Basis FHA (rem)	Difference (%)
EAB	0.56437	2.2	-74.4
LPZ	0.069735	0.28	-75.1
Control Room	3.0402	4.04	-24.8

11.0 REFERENCES

- [1] US NRC, "10 CFR 50.67 Accident Source Term," March 24, 2021.
- [2] US NRC, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, Revision 0.
- [3] US NRC, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors," NUREG-0800, Section 15.0.3, Initial Issuance.
- [4] Holtec International, "SMR-300 Program Quality Plan," HPP-8002-0002, Revision 4.
- [5] Holtec International, "Topical Report On The Quality Assurance Program for Holtec International's Small Modular Reactor (SMR) Design and Construction," HI-2230815, Revision 2.
- [6] The American Society of Mechanical Engineers, "NQA-1 Quality Assurance Requirements for Nuclear Facility Applications," ASME, New York, NY, February 20, 2015.
- [7] Holtec International, "Acronyms, Abbreviations, and Glossary of Terms," HI-2240905, Revision 1.
- [8] Holtec International, "Commercial Grade Dedication of Computer Code RADTRAD," HI-2220870, Revision 3.
- [9] E-Mail from Angelo Camargo to Hovhannes Hovhannisyan, *Subject: Palisades DD41 Isotropics*, 9/26/2026.
- [10] PALISADES NUCLEAR PLANT, "UPDATED FINAL SAFETY ANALYSIS REPORT, Rev. 38".
- [11] NAI-E03-211 Project Memo 2004-07, "Remaining Event Inputs," July 8 2004.



- [12] Email from Jeffery Voskuil (NMC) to Jim Harrell and Steve Thomasson (NAI), *Subject: Response to Input Questions from NAI*, 10/8/2004.
- [13] Palisades Plant, Consumers Power Company, "Specification For Ventilating System Filters," No. 5935-M-52, April 19, 1968.
- [14] NAI Calculation Number NAI-1149-002 Rev. 0, "Determination of Atmospheric Dispersion Factors for Palisades".
- [15] NAI Calculation Number NAI-1149-024 Rev. 4, "Determination of Direct Shine Doses for a Design Basis LOCA for Palisades".
- [16] U.S. Environmental Protection Agency, "Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhalation, Submersion, And Ingestion," Federal Guidance Report No. 11, September 1988.
- [17] U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, September 1993.

12.0 LIST OF APPENDICES

Appendix A – RADTRAD Model Input Data



Appendix A RADTRAD Model Input Data

Model Options:

Plant Power Level = 2703 MWth
Decay = Decay and daughtering
Onset of Gap Release = 0.0 hours
Start of Accident = 720.0 hours
Dose Conversion type = FGR 11& 12 [16], [17]

Source term:

Inventory: Palisades Single FA DD41

Table A-1 Palisades Single FA DD41

Isotope	Activity (Ci)
Kr-85	3733.460551
I-129	0.012819665
Cs-134	19238.24497
Cs-135	0.289425131
Cs-137	49715.712

Release Durations:

Gap = 1E-4 hrs
Early = 0.0 hrs

Release Fractions:

Group	Fraction
Noble Gases	0.1
Halogens	0.05
Alkali Metals	0.12
Tellurium	0.0
Alkaline Earth Metals	0.0
Noble Metals	0.0
Cerium	0.0
Lanthanides	0.0
Others	0.0
Non-radioactive Aerosols	0.0

Iodine Chemical Form:

Aerosol Fraction	0.95
Elemental Fraction	0.0485
Organic Fraction	0.0015

Compartments:

Containment:



Type: Normal

Volume: 1.64E6 ft³ (Reference [10])

Source Connection: FHA

FHB:

Type: Normal

Volume: 9.3E4 ft³ (Reference [10])

Environment:

Type: Environment

Dose Locations: Exclusion Area Boundary and Low Population Zone

Onsite X/Q Tables:

Release to the environment	Intake from environment
Stack Release	Unfiltered Inleakage (Control Room Normal Intake, X/Q Table 3)
	Filtered Makeup (Control Room Emergency Intake, X/Q Table 4)
	Unfiltered Makeup (Control Room Normal Intake, X/Q Table 3)
Exhaust Flow	-

Control Room:

Type: Control Room Dose

Volume: 3.5923E4 ft³ (Reference [10])

Dose Locations: Control Room

Filter: Filter 6 (Control Room Recirculation Flow)

Pathways:

Hatch Release:

From Compartment: Containment

To Compartment: FHB

Pathway Type: Filtered Pathway

Filter: Filter 1 (Hatch Release)

Stack Release:

From Compartment: FHB

To Compartment: Environment



Pathway Type: Filtered Pathway

Filter: Filter 2 (Stack Release)

Unfiltered Inleakage:

From Compartment: Environment

To Compartment: Control Room

Pathway Type: Filtered Pathway

Filter: Filter 3 (Unfiltered Inleakage)

Filtered Makeup:

From Compartment: Environment

To Compartment: Control Room

Pathway Type: Filtered Pathway

Filter: Filter 4 (Filtered Makeup)

Exhaust Flow:

From Compartment: Control Room

To Compartment: Environment

Pathway Type: Filtered Pathway

Filter: Filter 5 (Exhaust Flow)

Unfiltered Makeup:

From Compartment: Environment

To Compartment: Control Room

Pathway Type: Filtered Pathway

Filter: Filter 7 (Unfiltered Makeup)

Dose Locations:

Exclusion Area Boundary:

Breathing rates:

Duration (hours)	Breathing Rates (m ³ /s)	Breathing Rates (cfm)
0	3.5E-4	0.741608
8.0	1.8E-4	0.3813984
24.0	2.3E-4	0.4873424

X/Q Table: X/Q Table 1 (Exclusion Area Boundary)

Compartment: Environment

Low Population Zone:



Breathing rates:

Duration (hours)	Breathing Rates (m ³ /s)	Breathing Rates (cfm)
0	3.5E-4	0.741608
8.0	1.8E-4	0.3813984
24.0	2.3E-4	0.4873424

X/Q Table: X/Q Table 2 (Low Population Zone)

Compartment: Environment

Control Room:

Breathing rates:

Duration (hours)	Breathing Rates (m ³ /s)	Breathing Rates (cfm)
0	3.5E-4	0.741608
720	3.5E-4	0.741608

X/Q Table: X/Q Table 3 (Control Room Normal Intake)

Occupancy Factors:

Time (Hours)	Occupancy Factor
0.0	1.0
24.0	0.6
96.0	0.4
720.0	0.4

Compartment: Control Room

Filters:

Filter 1 (Hatch Release):

Filter table:

Time (Hours)	Flow rate (cfm)	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	1.0E6	0.0	0.0	0.0
720.0	1.0E6	0.0	0.0	0.0

Filter 2 (Stack Release):

Filter table:

Time (Hours)	Flow rate (cfm)	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	7.5E4	99.0	94.0	94.0
720.0	7.5E4	99.0	94.0	94.0

Filter 3 (Unfiltered Inleakage):



Filter table:

Time (Hours)	Flow (cfm)	rate	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	0.0		0.0	0.0	0.0
0.3333	100.0		0.0	0.0	0.0
720.0	100.0		0.0	0.0	0.0

Filter 4 (Filtered Makeup):

Filter table:

Time (Hours)	Flow (cfm)	rate	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	0.0		99.0	99.0	99.0
0.3333	1413.6		99.0	99.0	99.0
720.0	1413.6		99.0	99.0	99.0

Filter 5 (Exhaust Flow):

Filter table:

Time (Hours)	Flow (cfm)	rate	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	660.0		0.0	0.0	0.0
0.33333	1513.6		0.0	0.0	0.0
720.0	1513.6		0.0	0.0	0.0

Filter 6 (Control Room Recirculation Flow):

Filter table:

Time (Hours)	Flow (cfm)	rate	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	0.0		99.0	99.0	99.0
0.33333	1413.6		99.0	99.0	99.0
720.0	1413.6		99.0	99.0	99.0

Filter 7 (Unfiltered Makeup):

Filter table:

Time (Hours)	Flow (cfm)	rate	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.0	660.0		0.0	0.0	0.0



Time (Hours)	Flow rate (cfm)	Aerosol Removal Efficiency (%)	Elemental Removal Efficiency (%)	Organic Removal Efficiency (%)
0.33333	0.0	0.0	0.0	0.0
720.0	0.0	0.0	0.0	0.0

X/Q Tables:

Exclusion Area Boundary:

Time (hours)	X/Q (s/m ³)
0.0	5.39E-4
720.0	5.39E-4

Low Population Zone:

Time (hours)	X/Q (s/m ³)
0.0	6.66E-5
2.0	3.03E-5
8.0	2.04E-5
24.0	8.67E-6
96.0	2.54E-6
720.0	2.54E-6

Control Room Normal Intake:

Time (hours)	X/Q (s/m ³)
0.0	6.1E-3
2.0	4.32E-3
8.0	1.73E-3
24.0	1.27E-3
96.0	9.79E-4
720.0	9.79E-4

Control Room Emergency Intake:

Time (hours)	X/Q (s/m ³)
0.0	8.32E-4
2.0	7.69E-4
8.0	2.83E-4
24.0	2.15E-4
96.0	1.57E-4
720.0	1.57E-4