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10 CFR 50.12
10 CFR 50.47
10 CFR 50.47, Appendix E

HDI PNP 2022-017

July 11, 2022

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

Subject: Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E

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

References:
1. Letter from Entergy Nuclear Operations, Inc. to U.S. Nuclear Regulatory Commission, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated June 13, 2022 (ADAMS Accession No. ML22164A067)
2. U.S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Nuclear Plant – Issuance of Amendment Re: Changes to the Emergency Plan for Permanently Defueled Condition (CAC No. MG0198; EPID L-2017-LLA-0305)," dated September 24, 2018 (ADAMS Accession No. ML18170A219)
3. Final Safety Evaluation by the Office of Nuclear Reactor Regulation for the Topical Report HI-2200750, Revision 0, “Holtec Spent Fuel Pool Heat Up Calculation Methodology,” (Docket: 99902086) (EPID L-2020-TOP-0056) dated March 25, 2022 (ADAMS Accession No. ML22075A308)

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.12, *Specific exemptions,* Holtec Decommissioning International, LLC. (HDI), on behalf of Holtec Palisades, LLC, hereby requests exemptions from portions of 10 CFR 50.47, *Emergency Plans*, paragraphs (b) and (c)(2); and 10 CFR Part 50, Appendix E, *Emergency Planning and Preparedness for Production and Utilization Facilities*, for the Palisades Nuclear Plant (PNP) license. The requested exemptions would allow for a reduction in scope of the PNP Post-Shutdown Emergency Plan (PSEP) consistent with the permanently shutdown and defueled condition of PNP.

By letter dated June 13, 2022, Entergy certified to the U.S. Nuclear Regulatory Commission (NRC) that power operations ceased at PNP on May 20, 2022, and that the fuel was permanently removed from the reactor vessel and placed in the PNP spent fuel pool (SFP) on

June 10, 2022, in accordance with 10 CFR 50.82, *Termination of License*, paragraphs (a)(1)(i) and (a)(1)(ii) (Reference 1). It is understood and acknowledged that upon the NRC's docketing of these certifications, in accordance with 10 CFR 50.82(a)(2), the license for PNP no longer authorizes operation of the reactor, nor emplacement or retention of fuel into the reactor vessel. The irradiated fuel will be stored in the SFP and in dry cask storage at an onsite independent spent fuel storage installation (ISFSI) until it is shipped offsite.

To address the transition from an operating plant to a permanently defueled facility, by letter dated September 24, 2018, the NRC issued Amendment No. 267 (PSEP) (Reference 2), approving changes to the PNP emergency plan to support the planned permanent cessation of operations and permanent removal of fuel from the reactor vessel. Upon implementation of the PSEP on June 15, 2022, the PNP emergency response organization (ERO) on-shift and augmented staffing were revised commensurate with the reduced spectrum of credible accidents for a permanently shutdown and defueled nuclear power reactor facility.

The exemptions requested herein would allow HDI to make changes to the PSEP as the risk and radiological consequences of credible accidents at PNP continue to decrease. The requested exemptions are permissible under 10 CFR 50.12 because they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and present special circumstances.

More specifically, application of the portions of the regulations from which exemptions are sought is not necessary to ensure adequate emergency response capability for PNP and to achieve the underlying purpose of the regulations. Furthermore, continued application of these portions of the regulations from which exemptions are sought would result in an undue hardship or other costs to the licensee and the PNP Decommissioning Trust Fund by requiring continued implementation of unnecessary emergency response capabilities. Finally, granting the requested exemptions would result in benefit to the public health and safety and would not result in a decrease in safety, because the exemptions would enhance the ability of the PNP ERO to respond to credible scenarios.

The specific exemptions being requested are contained in the Enclosure to this letter.

Site-specific calculations have been developed to determine the end of the zirconium fire period for PNP utilizing the methodology approved by the NRC in Reference 3. The PNP site-specific calculations indicate that the requested exemptions may be made effective with an optimized SFP layout at approximately 12 months after power operations cease at PNP. PNP shutdown occurred on May 20, 2022 (Reference 1), and approximately 12 months after shutdown will be May 31, 2023.

Attachment 1 to the Enclosure contains a proprietary version of the calculations used to develop the request for exemptions included in the Enclosure. HDI requests that the NRC withhold this information in accordance with 10 CFR 2.390, *Public inspections, exemption, requests for withholding*.

Attachment 2 to the Enclosure contains a non-proprietary, redacted version of the calculations.

Attachment 3 contains an affidavit supporting withholding of the proprietary information described above.

Attachment 4 contains an analysis demonstrating that there is a High Confidence of Low Probability of Failure (HCLPF) of the SFP.

HDI plans to submit a permanently defueled emergency plan (PDEP) amendment request, containing a PDEP and a Permanently Defueled Emergency Action Level (EAL) scheme, for NRC review and approval in accordance with to 10 CFR 50.54, *Conditions of licenses*, paragraph (q)(4) and 10 CFR 50, Appendix E, Section IV, *Content of Emergency Plans*, paragraph B.2. The proposed PDEP will be based on the exemptions requested herein.

In support of the requested exemptions, PNP personnel have had discussions with cognizant state and local response organizations regarding the regulatory exemption requests to be submitted to the NRC. PNP personnel will continue to meet with representatives from the State of Michigan; local emergency preparedness personnel with Berrien County, Allegan County, and Van Buren County.

HDI requests review and approval of this exemption request by April 29, 2023, with the approved exemptions to become effective on May 31, 2023 (i.e., approximately 12 months following permanent shutdown of PNP). Approval of these exemptions by April 29, 2023, will allow adequate time to prepare changes to the PSEP and ERO prior to the effective date.

In accordance with 10 CFR 50.91, *Notice for public comment; State consultation*, paragraph (b), a copy of this exemption request, with enclosure, is being provided to the designated State Officials.

This letter contains no new regulatory commitments or updates to existing commitments.

If you have any questions regarding this submittal, please contact Jim Miksa, Regulatory Assurance Engineer, at (269) 764-2945.

Respectfully,

Jean A. Fleming

Digitally signed by Jean A. Fleming
Date: 2022.07.11 13:11:53 -04'00'

Jean A. Fleming
Vice President, of Licensing, Regulatory Affairs & PSA
Holtec International

Enclosure: Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50, Appendix E

Attachments to Enclosure:

1. Holtec Spent Fuel Pool Calculations (Holtec Proprietary, Withhold Information from Public Disclosure pursuant to 10 CFR 2.390)

2. Holtec Spent Fuel Pool Calculations (Non-Proprietary)

3. Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure

4. Palisades Spent Fuel Pool HCLPF Evaluation

cc:

U.S. NRC Regional Administrator (Region III)
NRC Senior Resident Inspector – PNP
NRC Project Manager - PNP
Designated Michigan State Official

Enclosure

HDI PNP 2022-017

Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50, Appendix E

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Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50, Appendix E

1.0 SUMMARY DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.12, *Specific exemptions*, Holtec Decommissioning International, LLC. (HDI), on behalf of Holtec Palisades, LLC, requests exemptions from certain requirements of the following regulations for the Palisades Nuclear Plant (PNP) license:

- 10 CFR 50.47, *Emergency Plans*, paragraph (b) regarding onsite and offsite emergency response plans for nuclear power reactors;
- 10 CFR 50.47(c)(2) to establish plume exposure and ingestion pathway emergency planning zones (EPZs) for nuclear power plants; and
- 10 CFR Part 50, Appendix E, *Emergency Planning and Preparedness for Production and Utilization Facilities*, which establishes the elements that make up the content of emergency plans.

The underlying purpose of the requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E are to ensure that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency; to establish plume exposure and ingestion pathway EPZs for nuclear power plants; and to ensure that licensees maintain effective onsite and offsite emergency plans, with the cooperation and assistance of State and local authorities. These requirements continue to apply to a nuclear power reactor licensee after permanent cessation of power operations and permanent removal of fuel from the reactor vessel because there are no explicit regulatory provisions distinguishing emergency planning requirements for a power reactor that has been permanently shutdown from those for an operating power reactor. However, once a power reactor is permanently shutdown and defueled, and a sufficient decay of the spent fuel has occurred, the risk of an offsite radiological release is significantly lower, and the types of possible accidents are significantly fewer. The requested exemptions would allow HDI to reduce emergency planning requirements and subsequently revise the PNP Post-Shutdown Emergency Plan (PSEP) to reflect the permanently shutdown and defueled condition.

The requested exemptions and justification for each are based on, and consistent with, Interim Staff Guidance (ISG) NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, issued May 11, 2015 (Reference 1).

2.0 BACKGROUND

By letter dated June 13, 2022, Entergy certified to the NRC that power operations ceased at PNP on May 20, 2022, and that the fuel was permanently removed from the reactor vessel and placed in the PNP spent fuel pool (SFP) on June 10, 2022, in accordance with 10 CFR 50.82, *Termination of License*, paragraphs (a)(1)(i) and (a)(1)(ii) (Reference 2). Upon the NRC's docketing of these certifications, in accordance with 10 CFR 50.82(a)(2), the license for PNP no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel. The irradiated fuel will be stored in the SFP and in dry cask storage at an onsite independent spent fuel storage installation (ISFSI) until it is shipped offsite.

To address the transition from an operating plant to a permanently defueled facility, by letter dated September 24, 2018, the NRC issued Amendment No. 267 (PSEP) (Reference 3), approving changes to the PNP emergency plan to support the planned permanent cessation of operations and permanent removal of fuel from the reactor vessel. Upon implementation of the PSEP on June 15, 2022, the PNP emergency response organization (ERO) on-shift and augmented staffing were revised commensurate with the reduced spectrum of credible accidents for a permanently shutdown and defueled nuclear power reactor facility.

When PNP was licensed for power operations, Chapter 14 of the PNP Updated Final Safety Analysis Report (UFSAR) describes accident analyses for postulated design basis accidents (DBAs) and transient scenarios under which PNP is licensed. The most severe postulated DBA involves damage to the nuclear reactor core and the release of large quantities of fission products. Many of these accident scenarios involve failures or malfunctions of systems, which could affect the fuel in the reactor vessel. With the termination of reactor operations and the permanent removal of fuel from the reactor vessel, such accidents are no longer possible. Therefore, the postulated accidents involving failure or malfunction of the reactor, reactor coolant system, steam system, or turbine generator, are no longer applicable. The only remaining DBAs will be the Fuel Handling Accident (FHA) in the SFP, the liquid waste incident, the waste gas incident, and the postulated cask drop accident. Because PNP is permanently shut down and the reactor is defueled, an FHA in the reactor cavity is no longer applicable because all irradiated spent fuel is stored in the SFP or an ISFSI. Therefore, because an FHA can only occur during movement of spent fuel in the SFP, the FHA event will be limited to the SFP.

The analyses of the potential radiological impact of accidents while the facility is in a permanently defueled condition indicate that no DBA or reasonably conceivable beyond design basis accident would result in radioactive releases that exceed U.S. Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs) (Reference 4) beyond the exclusion area boundary (EAB).

The exemptions requested herein would allow HDI to make changes to the emergency plan as the risk and radiological consequences of credible accidents at PNP continue to decrease. The requested exemptions are permissible under 10 CFR 50.12 because they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and present special circumstances.

3.0 BASIS FOR EXEMPTION REQUEST

To allow for a reduction in emergency planning requirements commensurate with the hazards associated with PNP's permanently defueled condition, exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E, are needed.

Calculations related to beyond design basis events for the PNP SFP were performed. The site-specific calculations demonstrate that at approximately 12 months after permanent cessation of power operations of the PNP reactor, a minimum of 10 hours is available before the fuel cladding temperature of the hottest fuel assembly in the SFP reaches the zirconium fire temperature of 900 degrees Celsius (°C) with a complete loss of SFP water inventory.

Attachment 1 to this Enclosure contains a proprietary version of the calculations used to develop this request for exemptions. Because Attachment 1 to this Enclosure contains proprietary information, it is supported by an affidavit provided in Attachment 3 to this Enclosure.

The calculations contained in Attachment 1 to this Enclosure include appropriate markings identifying those portions that are proprietary. Attachment 2 to this Enclosure contains a non-proprietary, redacted version of the calculation.

Based on the shutdown date of May 20, 2022, approximately 12 months following permanent cessation of power operations of PNP would occur on May 31, 2023. Based on the results of the analysis, in the unlikely event of a beyond design basis event, 10 hours is available to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel and, if governmental officials deem warranted, for authorities to implement offsite protective actions using a comprehensive approach to emergency planning to protect the health and safety of the public before the hottest fuel assembly reaches the zirconium fire temperature. Because of the time it would take for the adiabatic heatup to occur, there is ample time to respond to any drain down event that might cause such an occurrence by restoring cooling or makeup, or providing spray to the PNP SFP. As a result, the likelihood that such a scenario would progress to a zirconium fire is deemed not credible.

Based on the analyses detailed in Section 5.0 of this Enclosure, it has been concluded that the portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E identified in Tables 1 and 2 in Section 4.0 of this Enclosure, are not necessary to protect the health and safety of the public because the PNP reactor is in the permanently shutdown and defueled condition, and continued applicability of the regulations would be unduly burdensome. Approval of the exemptions requested in Tables 1 and 2 of this Enclosure would not present an undue risk to the public or prevent appropriate response in the event of an emergency at PNP.

HDI plans to submit a license amendment request, containing a Permanently Defueled Emergency Plan (PDEP) and an associated emergency action level (EAL) scheme, for NRC review and approval in accordance with the requirements of 10 CFR 50.54, *Conditions of licenses*, paragraph (q)(4) and 10 CFR 50, Appendix E, Section IV.B.2. The proposed PDEP and EAL scheme will be based on the exemptions requested herein.

4.0 REQUESTED EXEMPTIONS FROM EMERGENCY PLANNING REQUIREMENTS

HDI requests exemptions from portions of 10 CFR 50.47(b) and (c)(2) and 10 CFR Part 50 Appendix E to the extent that these regulations apply to specific provisions of onsite and offsite emergency planning that will no longer be applicable approximately 12 months after permanent cessation of power operations of the PNP reactor, because the certifications required by 10 CFR 50.82(a)(1)(i) and (ii) have been submitted to the NRC, and sufficient decay of the PNP spent fuel will have occurred.

The specific portions of 10 CFR 50.47 and 10 CFR Part 50 Appendix E, from which exemptions are being requested are identified using strikethrough text in Table 1 (Exemptions Requested from 10 CFR 50.47(b) and (c)(2)) and Table 2 (Exemptions Requested from 10 CFR Part 50, Appendix E), below. The portions of the regulations that are not identified using strikethrough text (i.e., those portions for which an exemption is not being requested), remain applicable to PNP. Details related to specific exemption requests are provided in the Basis for Exemption column in the Tables 1 and 2 of this Enclosure.

The requested exemptions and the associated justification for each are based on, and consistent with, NSIR/DPR-ISG-02 (Reference 1).

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
1	10 CFR 50.47(b): The onsite ~~and, except as provided in paragraph (d) of this section, offsite~~ emergency response plans for nuclear power reactors must meet the following standards:	In the Statement of Considerations for the Final Rule for Emergency Planning requirements for Independent Spent Fuel Storage Installations (ISFSIs) and for monitored retrievable storage (MRS) facilities (60 FR 32430; June 22, 1995) (Reference 5), the Commission responded to comments concerning offsite emergency planning for ISFSIs or an MRS and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS [monitored retrievable storage installation] would not warrant establishing Emergency Planning Zones." As discussed in ISG-02 (Reference 1), in a nuclear power reactor's permanently defueled state, the accident risks are more like an ISFSI or MRS than an operating nuclear power plant. The draft proposed rulemaking in SECY-00-0145 (Reference 6) suggested that after at least one year of spent fuel decay time, the decommissioning licensee would be able to reduce its emergency planning program to one similar to that required for an MRS under 10 CFR 72.32, *Emergency Plan*, paragraph (b) and additional emergency planning reductions would occur when: (1) approximately five years of spent fuel decay time has elapsed; or (2) a licensee has demonstrated that the decay heat level of spent fuel in the pool is low enough that the fuel would not be susceptible to a zirconium fire for all spent fuel configurations. The EP program would be like that required for an ISFSI under 10 CFR 72.32(a) when fuel stored in the SFP has more than five years of decay time and would not change substantially when all the fuel is transferred from the SFP to an onsite ISFSI. Because of the slow progression of the event scenarios postulated in the DBA and postulated beyond DBA analyses and because the

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
		duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor, significant time is available to complete actions necessary to mitigate an emergency without impeding timely performance of emergency plan functions. Exemptions from offsite emergency planning requirements have previously been approved when site-specific analyses indicate that at least 10 hours is available from a partial drain down event where cooling of the spent fuel is not effective until the hottest fuel assembly reaches 900°C. The technical basis that underlie the approval of the exemption request is based partly on the analysis of a period that spent fuel stored in the SFP is unlikely to reach the zirconium ignition temperature in less than 10 hours. This period is based on a heatup calculation which uses several simplifying assumptions. Some of these assumptions are conservative (adiabatic conditions), while others are non-conservative (no oxidation below 900°C). Weighing the conservatisms and non-conservatisms, the NRC staff has judged that this calculation reasonably represents conditions which may occur in the event of a SFP accident. The NRC staff concluded that if 10 hours were available to initiate mitigative actions, or if needed, offsite protective actions using a Comprehensive Emergency Management Plan (CEMP), formal offsite radiological emergency plans would not be necessary for a permanently defueled nuclear power reactor licensee. Within 90 days of the permanent cessation of power operations of the PNP reactor, the radiological consequences of the DBAs that remain applicable to PNP will not exceed small fractions of the U.S. Environmental Protection Agency's (EPA) Protective Action Guides (PAGs) at the Exclusion Area Boundary (EAB). The radiological

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
		consequences of the remaining applicable DBAs are discussed in Section 5.2 of this Enclosure. An analysis for the PNP SFP for beyond design basis events has been performed, which demonstrates that at approximately 12 months after permanent cessation of power operations of the PNP reactor, a minimum of 10 hours is available before fuel cladding temperature of the hottest fuel assembly in the SFP reaches 900°C with a complete loss of SFP water inventory, assuming no heat loss (adiabatic heatup). HDI maintains procedures and strategies for the movement of any necessary portable equipment that will be relied upon for mitigating the loss of SFP water. These mitigative strategies, addressing events involving a loss of SFP cooling and/or water inventory, include implementation of SFP inventory makeup strategies required under 10 CFR 50.155, *Mitigation of beyond-design-basis events*, for a permanently shut down and defueled facility, which will continue to be maintained in accordance with Renewed License No. DPR-20, License Condition 6.b.7. These diverse strategies provide defense-in-depth and ample time to provide makeup water or spray to the SFP prior to the onset of zirconium cladding ignition when considering very low probability beyond design basis events affecting the SFP. These diverse strategies are described in more detail in the PNP response to SDA-4 in Table 5 of this Enclosure. Two (2) trained on-shift individuals at PNP can implement necessary actions to supply makeup water to the PNP SFP in approximately two (2) hours. The two (2) on-shift individuals are assigned to perform this task and they do not have other assigned required emergency preparedness activities during the performance of this

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
		task that would inhibit timely performance. Direction and selection of the tasks related to adding makeup water to the PNP SFP will continue to be the responsibility of the Emergency Director.
2	10 CFR 50.47(b)(1): Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations ~~within the Emergency Planning Zones~~ have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
3	10 CFR 50.47(b)(2): On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.	No exemption is requested.
4	10 CFR 50.47(b)(3): Arrangements for requesting and effectively using assistance resources have been made, ~~arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made,~~ and other organizations capable of augmenting the planned response have been identified.	Discontinuing offsite emergency planning activities and reducing the scope of onsite emergency planning is acceptable given the significantly reduced offsite consequences when PNP is in the permanently defueled condition. The PDEP will continue to maintain arrangements for requesting and using assistance resources from offsite support organizations. Decommissioning power reactors present a low likelihood of any credible accident resulting in radiological releases requiring offsite

Table 1

Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
		protective measures because of the permanently shut down and defueled status of the reactor. An emergency operations facility (EOF) is not required. The control room or another location can provide for the communication and coordination with offsite organizations for the level of support required. Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services as appropriate. Also refer to the basis for 10 CFR 50.47(b).
5	10 CFR 50.47(b)(4): A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee~~, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures~~.	PNP will adopt the Permanently Defueled Emergency Action Levels (EALs) detailed in Appendix C of Nuclear Energy Institute (NEI) 99-01, "Development of EALs for Non-Passive Reactors," Revision 6 (Reference 7), endorsed by the NRC in a letter dated March 28, 2013. No offsite protective actions are anticipated to be necessary. Therefore, classification above the Alert level (e.g., Site Area Emergency or General Emergency) will no longer be required. Also refer to the basis for 10 CFR 50.47(b).
6	10 CFR 50.47(b)(5): Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations ~~and the public~~ has been established~~; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established~~.	SECY-00-0145 (Reference 6) indicates that after approximately 1 year of spent fuel decay time (and as supported by the adiabatic heatup analysis), the NRC staff believes an exception to the offsite EPA PAG standard is justified for a zirconium fire scenario considering the low likelihood of this event together with time available to take mitigative or protective actions between the initiating event and before the onset of a postulated zirconium fire. SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)**		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
		Water Reactor," (Reference 8) provides that depending on the size of the pool liner leak, releases could start anywhere from eight hours to several days after the leak starts, assuming that mitigation measures are unsuccessful. If 10 CFR 50.155(b)(2) (formerly 10 CFR 50.54(hh)(2)) - type mitigation measures are successful, releases could only occur during the first several days after the fuel was removed from the reactor. Therefore, offsite emergency plans are not necessary for permanently defueled nuclear power plants. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
7	10 CFR 50.47(b)(6): Provisions exist for prompt communications among principal response organizations to emergency personnel ~~and to the public.~~	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
8	10 CFR 50.47(b)(7): ~~Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors),~~ [T]he principal points of contact with the news media for dissemination of information during an emergency ~~(including the physical location or locations)~~ are established in advance, and procedures for coordinated dissemination of information to the public are established.	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
9	10 CFR 50.47(b)(8): Adequate emergency facilities and equipment to support the emergency response are provided and maintained.	No exemption is requested.

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)**		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
10	10 CFR 50.47(b)(9): Adequate methods, systems, and equipment for assessing and monitoring actual or potential ~~offsite~~ consequences of a radiological emergency condition are in use.	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
11	10 CFR 50.47(b)(10): A range of protective actions has been developed for ~~the plume exposure pathway EPZ for~~ emergency workers and the public. ~~In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~	In the unlikely event of a SFP accident, the iodine isotopes which contribute to an offsite dose from an operating reactor accident are not present, so potassium iodide (KI) distribution offsite would no longer serve as an effective or necessary supplemental protective action. Protective actions will be maintained for emergency workers and any offsite emergency responders who would respond to the site. The Commission responded to comments in its Statement of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and MRS facilities (60 FR 32435) (Reference 5), and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS would not warrant establishing Emergency Planning Zones." Additionally, in the Statement of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 5), the Commission responded to comments concerning site-specific emergency planning that includes evacuation of surrounding population for an ISFSI not at a reactor site, and concluded that, "[T]he Commission does not agree that as a general matter emergency plans for an ISFSI must include evacuation planning." Because the NRC concludes that evacuation planning is not needed for a decommissioning reactor site that meets the criteria for an

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)**		
Item #	**Regulation in 10 CFR 50.47**	**Basis for Exemption**
		exemption from offsite EP requirements as discussed in the exemption from 10 CFR 50.47(b) (Item 1 of Table), evacuation time estimates are also not needed. Also refer to the basis for 10 CFR 50.47(b) (Item 1 of Table 1) detailing the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures and basis for Appendix E, Section IV.1 (Item 2 in Table 2) exemptions for discussion of the similarity between a permanently defueled reactor and a non-power reactor.
12	10 CFR 50.47(b)(11): Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.	No exemption is requested.
13	10 CFR 50.47(b)(12): Arrangements are made for medical services for contaminated injured individuals.	No exemption is requested.
14	10 CFR 50.47(b)(13): General plans for recovery and reentry are developed.	No exemption is requested.
15	10 CFR 50.47(b)(14): Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.	No exemption is requested.

<u>Table 1</u>

Exemptions Requested from 10 CFR 50.47(b) and 10 CFR 50.47(c)(2)

Item #	Regulation in 10 CFR 50.47	Basis for Exemption
16	10 CFR 50.47(b)(15): Radiological emergency response training is provided to those who may be called on to assist in an emergency.	No exemption is requested.
17	10 CFR 50.47(b)(16): Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.	No exemption is requested.
18	10 CFR 50.47(c)(2)~~: Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.~~ The size of the EPZs ~~also~~ may be determined on a case-by-case basis for gas cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal.~~ The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.~~	Federal guidance provided in the EPA's "Protective Action Guides and Planning Guidance for Radiological Incidents, EPA-400/R-17/001," dated January 2017 (EPA PAG Manual) states that the EPZ is based on the maximum distance at which a PAG might be exceeded (Reference 4). Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
1	10 CFR 50 Appendix E III. The Final Safety Analysis Report; Site Safety Analysis Report The final safety analysis report or the site safety analysis report for an early site permit that includes complete and integrated emergency plans under § 52.17(b)(2)(ii) of this chapter shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee. The site safety analysis report for an early site permit which proposes major features must address the relevant provisions of 10 CFR 50.47 and 10 CFR part 50, appendix E, within the scope of emergency preparedness matters addressed in the major features. The plans submitted must include a description of the elements set out in Section IV for the emergency planning zones (EPZs) to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.	No exemption is requested.

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
2	10 CFR 50 Appendix E IV. Content of Emergency Plans 1. The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, *i.e.*, organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery~~, and onsite protective actions during hostile action~~. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license under this part, or for an early site permit (as applicable) or combined license under 10 CFR part 52, shall contain information needed to demonstrate compliance with the standards described in § 50.47(b), and they will be evaluated against those standards.	Upon docketing of its "Certification of Permanent Removal of Fuel from the Reactor Vessel," in accordance with 10 CFR 50.82(a)(1)(i) and (ii), PNP is a permanently shut down facility with spent fuel stored in the PNP SFP and ISFSI. In the EP Final Rule (76 FR 72596, Nov. 23, 2011) (Reference 9), the NRC defined "hostile action" as, in part, an act directed toward a nuclear power plant or its personnel. This definition is based on the definition of "hostile action" provided in NRC Bulletin 2005-02 (Reference 10). NRC Bulletin 2005-02 was not applicable to nuclear power reactors that have permanently ceased operations and have certified that fuel has been removed from the reactor vessel. The NRC excluded non-power reactors (NPRs) from the definition of "hostile action" at that time because an NPR is not a nuclear power plant and a regulatory basis had not been developed to support the inclusion of NPR in that definition. Similarly, a decommissioning power reactor or ISFSI is not a "nuclear reactor" as defined in the NRC's regulations. The following similarities between PNP and NPRs demonstrate that the PNP facility should be treated in a similar fashion as an NPR. Similar to NPRs, PNP poses lower radiological risks to the public from accidents than do operating power reactors because: (1) PNP is a permanently shut down facility (with fuel stored in the SFP and ISFSI) and can no longer generate fission products; 2) Fuel stored in the PNP SFP and ISFSI has lower decay heat resulting in lower risk of fission product release in the event of a non-credible boil off or drain down event; and 3) no credible accident at PNP can result in radiological releases requiring offsite protective actions.

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
		Although, the analysis described above provides a justification for exempting PNP from "hostile action"-related requirements, some emergency planning requirements for security-based events will be maintained. The classification of security-based events, notification of offsite authorities, and coordination with offsite agencies under a CEMP concept will still be required.
3	IV.2 ~~This nuclear power reactor license applicant shall also provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, using the most recent U.S. Census Bureau data as of the date the applicant submits its application to the NRC.~~	Refer to the basis for 10 CFR 50.47(b)(10) (Item 11 in Table 1).
4	IV.3 ~~Nuclear power reactor licensees shall use NRC approved evacuation time estimates (ETEs) and updates to the ETEs in the formulation of protective action recommendations and shall provide the ETEs and ETE updates to State and local governmental authorities for use in developing offsite protective action strategies.~~	Refer to the basis for 10 CFR 50.47(b)(10) (Item 11 in Table 1).
5	IV.4 ~~Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and~~	Refer to the basis for 10 CFR 50.47(b)(10) (Item 11 in Table 1).

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~	
6	IV.5 ~~During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.~~	Refer to the basis for 10 CFR 50.47(b)(10) (Item 11 in Table 1).
7	IV.6 ~~If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC under § 50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and~~	Refer to the basis for 10 CFR 50.47(b)(10) (Item 11 in Table 1).

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	~~providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~	
8	IV.7 After an applicant for a combined license under part 52 of this chapter receives its license, the licensee shall conduct at least one review of any changes in the population of its EPZ at least 365 days prior to its scheduled fuel load. The licensee shall estimate EPZ permanent resident population changes using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. If the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ, to increase by 25 percent or 30 minutes, whichever is less, from the licensee's currently approved ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC for review under § 50.4 of this chapter no later than 365 days before the licensee's scheduled fuel load.	No exemption is requested. HDI is not an applicant for a combined license. Therefore, this regulation is not applicable to PNP, and an exemption is not necessary.
9	A. Organization The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for	No exemption is requested.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	notification of such individuals in the event of an emergency. Specifically, the following shall be included:	
10	A.1. A description of the normal plant ~~operating~~ organization.	Upon docketing of the certifications required by 10 CFR 50.82(a)(1), PNP is not a facility that can be operated to generate electrical power. Therefore, PNP will not have a "plant operating organization." Rather, the facility will be maintained by a defueled on-shift staff.
11	A.2. A description of the onsite emergency response organization (ERO) with a detailed discussion of: a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency; b. Plant staff emergency assignments; c. Authorities, responsibilities, and duties of an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.	No exemption is requested.
12	A.3. ~~A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization~~.	The number of staff at PNP during the decommissioning process will be small but commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. HDI will maintain a level of emergency response that does not require response by headquarters personnel. The on-shift and emergency response positions will be defined in the Permanently Defueled Emergency Plan (PDEP).

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
13	A.4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making ~~offsite~~ dose projections and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.	Analyses been developed indicating that, within approximately 12 months after shutdown, no credible accident at PNP can result in radiological releases requiring offsite protective actions. HDI will maintain the capability to determine if a radiological release is occurring and perform dose projections. If a release is occurring, HDI will communicate release and dose projection information to offsite authorities for their consideration. The offsite organizations are responsible for deciding what, if any, protective actions should be taken. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
14	A.5. ~~Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.~~	As indicated by the adiabatic heatup analysis, the time available to initiate compensatory actions in the event of a loss of SFP cooling or inventory precludes the need to identify and describe the special qualification of these individuals in the emergency plan. The number of staff at PNP when it is in the permanently defueled state will be small but will be commensurate with the need to maintain the facility in a manner that is protective of public health and safety. The on-shift individuals described in the PDEP will be able to implement SFP mitigation strategies.
15	A.6. A description of the local offsite services to be provided in support of the licensee's emergency organization.	No exemption is requested.
16	A.7. ~~By June 23, 2014,~~ identification of~~, and a description of the~~ assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with	A decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For this reason and those described in the basis for

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	emergencies, including ~~hostile action at the site. For purposes of this appendix, "hostile action" is defined as~~ an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force.	Section IV.1 of 10 CFR Part 50, Appendix E (Item 2 of Table 2), a decommissioning power reactor is not a facility that falls within the definition of "hostile action." Similarly, for security, risk insights can be used to determine which targets are important to protect against sabotage. A level of security commensurate with the consequences of a sabotage event is required and is evaluated on a site-specific basis. The severity of the consequences declines as fuel ages, and over time, the underlying concern that a sabotage attack could cause offsite radiological consequences is removed. Although the analysis provided above and in the basis for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 of Table 2) provides a justification for exempting PNP from "hostile action"-related requirements, some emergency planning requirements for security-based events will be maintained. The classification of security-based events, notification of offsite authorities, and coordination with offsite agencies under a CEMP concept will still be required. HDI will maintain appropriate actions for the protection of onsite personnel in a security-based event. The scope of protective actions will be appropriate for the defueled plant status, but will not be the same as actions necessary for an operating power plant. Although the NRC has previously exempted decommissioning power reactors from "hostile action" considerations, the PNP Physical Security Plan will continue to provide high assurance against a potential security event impacting a designated target set. Therefore, some emergency planning requirements for security-based events are maintained. Protective actions are maintained for onsite personnel through the classification of security-based events,

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
		notification of offsite authorities, and coordination of offsite response organizations (i.e., local law enforcement, firefighting, medical assistance) onsite under a CEMP concept.
17	A.8. ~~Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.~~	Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services as appropriate. Because analyses have been developed indicating that approximately 12 months after shutdown, no credible accident at PNP can result in radiological releases in excess of EPA PAGs requiring offsite protective actions, protective actions such as evacuation should not be required, but could be implemented at the discretion of offsite authorities under a CEMP concept. Also refer to the basis for 10 CFR 50.47(b) (Item 1 in Table 1).
18	A.9. ~~By December 24, 2012, for nuclear power reactor licensees, a detailed analysis demonstrating that on shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.~~	Responsibilities for on-shift and emergency response personnel are detailed in the PDEP and implementing procedures, and will be regularly tested through drills and exercises, audited, and inspected by HDI and the NRC. The duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating power reactor. In the EP Final Rule (Reference 9), the NRC acknowledged that the staffing analysis requirement was not necessary for non-power reactor licensees because staffing at non-power reactors is generally small, which is commensurate with operating the facility in a manner that is protective of the public health and safety. The minimal systems and equipment needed to maintain the spent nuclear fuel in the spent fuel pool (SFP) or in a dry cask storage system in a safe condition requires minimal personnel and is governed by Technical Specifications. Because of the slow

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
		progression of the event scenarios postulated in the DBA and postulated beyond DBA analyses and because the duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor, significant time is available to complete actions necessary to mitigate an emergency without impeding timely performance of emergency plan functions. Additionally, the duties of the on-shift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor. For these reasons, it can be concluded that a decommissioning NPP is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.A.9.
19	B. Assessment Actions B.1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within ~~and outside~~ the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite ~~and offsite~~ monitoring. ~~By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant.~~ The initial emergency action levels shall be	Based on the exemptions from 10 CFR 50.47(b), the PDEP will state that the annual EAL review will include the contiguous State and local offsite agencies; specifically, the State of Michigan and Van Buren County. However, based upon the reduced scope of EALs for the permanently defueled PNP reactor, the scope of the annual review of EALs is expected to be limited (i.e., informal mailings, etc.). HDI will develop EALs consistent with the guidance on Permanently Defueled EALs detailed in Appendix C of NEI 99-01, Revision 6 (Reference 7). Also, refer to the basis for exemption for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 of Table 2), for the justification from the regarding "hostile action."

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.	
20	B.2. A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in § 50.54(q) for all other emergency action level changes.	No exemption is requested.
21	C. Activation of Emergency Organization C.1. The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite ~~and offsite~~ radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, ~~such as the pressure in containment and the response of the Emergency Core Cooling System)~~ for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) Notification of unusual events, (2) alert, ~~(3) site area emergency, and (4) general~~	The Permanently Defueled EALs, developed consistent with Appendix C of NEI 99-01, Revision 6 (Reference 7), will be adopted as previously described. This scheme eliminates the Site Area Emergency and General Emergency event classifications. Additionally, the need to base EALs on containment parameters and the response of the Emergency Core Cooling System is no longer appropriate. The guidance presented in NEI 99-01, Rev.6 was endorsed by the NRC in a letter dated March 28, 2013 (ADAMS Accession No. ML 12346A463) (Reference 11). No offsite protective actions are anticipated to be necessary, so classification above the Alert level is no longer required. In the event of an accident at a defueled facility that meets the conditions for relaxation of emergency planning requirements, time is available for event mitigation, and if necessary, implementation of offsite protective actions using a comprehensive approach to emergency planning. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) detailing the low likelihood of any credible accident

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	~~emergency~~. These classes are further discussed in NUREG–0654/FEMA–REP–1.	resulting in radiological releases requiring offsite protective measures. In the Statement of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 5), the Commission responded to comments concerning a General Emergency at an ISFSI and MRS, and concluded that, "…an essential element of a General Emergency is that a release can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels off site for more than the immediate site area." The probability of a condition reaching the level above an emergency classification of Alert is very low. In the event of an accident at a defueled facility that meets the conditions for relaxation of EP requirements, time is available to initiate measures to protect the public in accordance with a comprehensive approach to emergency planning. As stated in NUREG-1738 (Reference 12), for instances of small SFP leaks or loss of cooling scenarios, these events evolve very slowly and generally leave many days for recovery efforts. Offsite radiation monitoring will be performed as the need arises. Due to the decreased risks associated with defueled plants, offsite radiation monitoring systems are not required.
22	C.2. ~~By June 20, 2012, nuclear power reactor~~ licensees shall establish and maintain the capability to assess, classify, and declare an emergency condition ~~within 15 minutes~~ after the availability of indications to plant operators that an emergency action level has been exceeded and shall promptly declare the	In the Proposed Rule (74 FR 23254) (Reference 13) to amend certain emergency planning requirements for 10 CFR Part 50, the NRC asked for public comment on whether the NRC should add requirements for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	emergency condition as soon as possible following identification of the appropriate emergency classification level. Licensees shall not construe these criteria as a grace period to attempt to restore plant conditions to avoid declaring an emergency action due to an emergency action level that has been exceeded. Licensees shall not construe these criteria as preventing implementation of response actions deemed by the licensee to be necessary to protect public health and safety provided that any delay in declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.	declare an emergency condition. The NRC received several comments on these issues. The NRC believed there may be a need for the NRC to be aware of security-related events early on so that an assessment of the likelihood that the event is part of a larger coordinated attack can be made. However, the NRC determined that further analysis and stakeholder interactions are needed prior to changing the requirements for non-power reactor licensees. Therefore, the NRC did not include requirements in the 2011 EP Final Rule (Reference 9) for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. HDI proposes to maintain the capability to assess, classify, and declare an emergency condition within 30 minutes after the availability of indications to operators that an EAL threshold has been reached. With the PNP reactor in the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible. The consequences resulting from the only remaining events develop over a significantly longer period. As such, the 15-minute requirement to classify and declare an emergency is unnecessarily restrictive. The proposed changes to the declaration and notification times were presented to the cognizant officials from the offsite response organizations, and no objections to the proposed changes were received. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) detailing the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures and 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 in

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
		Table 2) for discussion on the similarity between a permanently defueled reactor and a non-power reactor.
23	D. Notification Procedures D.1. Administrative and physical means for notifying local, State, and Federal officials and agencies ~~and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary,~~ shall be described. This description shall include identification ~~of the appropriate officials, by title and agency,~~ of the State and local government agencies ~~within the EPZs~~.	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) and 10 CFR 50.47(b)(10) (Item 11 in Table 1).
24	D.2. ~~Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.~~	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) and 10 CFR 50.47(b)(10) (Item 11 in Table 1).
25	D.3. A licensee shall have the capability to notify responsible State and local governmental agencies ~~within 15 minutes~~ after declaring an emergency. ~~The licensee shall demonstrate that the appropriate governmental authorities have the capability to~~	HDI proposes to complete emergency notification within 60 minutes to appropriate state and local government agencies after an emergency declaration or a change in classification. This timeframe is consistent with the 10 CFR 50.72(a)(3) notification to the NRC

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~make a public alerting and notification decision promptly on being informed by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at a site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and initiate notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15-minute design objective for the primary prompt public alert and notification system. When there is a decision to activate the alert and notification system, the appropriate governmental authorities~~	and is appropriate because in the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible and there is no need for State or local response organizations to implement any protective actions. Likewise, there is no need to maintain an alert and notification system. The PNP PDEP includes primary and backup means for conducting the required notifications to the appropriate State and local government agencies. Decommissioning-related emergency plan submittals for PNP have been discussed with cognizant officials from these offsite response organizations. These discussions have addressed changes to onsite and offsite emergency preparedness throughout the decommissioning process, including the proposed changes to those agencies that are provided emergency notifications, the 30-minute emergency declaration time, the 60-minute notification time, those agencies participating in the annual review of EALs, and those agencies invited to participate in drills and exercises. The proposed changes to the declaration and notification times were presented to the cognizant officials from the offsite response organizations, and no objections to the proposed changes were received. Also, refer to the bases for 10 CFR 50.47(b) (Item 1 in Table 1) and 10 CFR 50.47(b)(10) (Item 11 in Table 1).

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~will determine whether to activate the entire alert and notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public alert and notification system shall remain with the appropriate governmental authorities.~~	
26	D.4. ~~If FEMA has approved a nuclear power reactor site's alert and notification design report, including the backup alert and notification capability, as of December 23, 2011, then the backup alert and notification capability requirements in Section IV.D.3 must be implemented by December 24, 2012. If the alert and notification design report does not include a backup alert and notification capability or needs revision to ensure adequate backup alert and notification capability, then a revision of the alert and notification design report must be submitted to FEMA for review by June 24, 2013, and the FEMA-approved backup alert and notification means must be implemented within 365 days after FEMA approval. However, the total time period to implement a FEMA-approved backup alert and notification means must not exceed June 22, 2015.~~	Refer to the basis for exemption for Section IV.D.3 (Item 25 in Table 2) regarding the requirements for an alert and notification system.
27	E. Emergency Facilities and Equipment Adequate provisions shall be made and described for emergency facilities and equipment, including: E.1. Equipment at the site for personnel monitoring;	No exemption is requested.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
28	E.2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;	No exemption is requested.
29	E.3. Facilities and supplies at the site for decontamination of onsite individuals;	No exemption is requested.
30	E.4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;	No exemption is requested.
31	E.5. Arrangements for medical service providers qualified to handle radiological emergencies onsite;	No exemption is requested.
32	E.6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;	No exemption is requested.
33	E.7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;	No exemption is requested.
34	E.8.a(i) A licensee ~~onsite technical support center and an emergency operations~~ facility from which effective direction can be given and effective control can be exercised during an emergency;	Analyses have been performed indicating that, approximately 12 months after shutdown, no credible accident at PNP can result in radiological releases requiring offsite protective actions, or in the event of beyond design basis accidents, 10 hours is available to take mitigative actions, and if needed, implement offsite protective actions using a CEMP approach. Therefore, offsite agency response is not required at an Emergency Operations Facility (EOF) and onsite actions may be directed from the Control Room or another

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
		location, without the requirements imposed on a Technical Support Center (TSC) or EOF. Therefore, there is no need to maintain a TSC or an EOF. An onsite facility will continue to be maintained, from which effective direction can be given and effective control may be exercised during an emergency. The PDEP will continue to maintain arrangements for requesting assistance and using resources from appropriate offsite support organizations. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) and 50.47(b)(3) (Item 4 in Table 1).
35	E.8.a(ii) ~~For nuclear power reactor licensees, a licensee onsite operational support center;~~	NUREG-0696, "Functional Criteria for Emergency Response Facilities," (Reference 14) provides that the operational support center (OSC) is an onsite area separate from the Control Room and the TSC where licensee operations support personnel will assemble in an emergency. For a defueled power plant, an OSC is no longer required to meet its original purpose of an assembly area for plant logistical support during an emergency. A single onsite facility will continue to be maintained at PNP, from which Control Room support, emergency mitigation, radiation monitoring, and effective control may be exercised during an emergency.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
36	E.8.b. ~~For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles from a nuclear power reactor site shall request prior Commission approval by submitting an application for an amendment to its license. For an emergency operations facility located more than 25 miles from a nuclear power reactor site, provisions must be made for locating NRC and offsite responders closer to the nuclear power reactor site so that NRC and offsite responders can interact face-to-face with emergency response personnel entering and leaving the nuclear power reactor site. Provisions for locating NRC and offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:~~	Refer to the basis for exemption for 10 CFR 50.47(b)(3) (Item 4 in Table 1).
37	E.8.b.(1) ~~Space for members of an NRC site team and Federal, State, and local responders~~	
38	E.8.b.(2) ~~Additional space for conducting briefings with emergency response personnel;~~	

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
39	E.8.b.(3) ~~Communication with other licensee and offsite emergency response facilities;~~	
40	E.8.b.(4) ~~Access to plant data and radiological information; and~~	
41	E.8.b.(5) ~~Access to copying equipment and office supplies;~~	
42	E.8.c. ~~By June 20, 2012, for a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, a facility having the following capabilities:~~ (1) ~~The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves;~~	Refer to the basis for exemption for 10 CFR 50.47(b)(3) (Item 4 in Table 1) and 10 CFR Part 50 Appendix E, Section IV.E.8.a(i) (Item 34 in Table 2).
43	E.8.c.(2) ~~The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves; and~~	
44	E.8.c.(3) ~~The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site; and~~	

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
45	E.8.d. ~~For nuclear power reactor licensees, an alternative facility (or facilities) that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and collectively having the following characteristics: the capability for communication with the emergency operations facility, control room, and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation, for use when onsite emergency facilities cannot be safely accessed during hostile action. The requirements in this paragraph 8.d must be implemented no later than December 23, 2014, with the exception of the capability for staging emergency response organization personnel at the alternative facility (or facilities) and the capability for communications with the emergency operations facility, control room, and plant security, which must be implemented no later than June 20, 2012.~~	Refer to the basis for exemption for 10 CFR Part 50 Appendix E, Section IV.1 (Item 2 in Table 2), regarding "hostile action."
46	E.8.e. ~~A licensee shall not be subject to the requirements of paragraph 8.b of this section for an existing emergency operations facility approved as of December 23, 2011;~~	Refer to the basis for exemption for 10 CFR 50.47(b)(3) (Item 4 in Table 1) and Appendix E to 10 CFR Part 50 (item 36 in Table 2).
47	E.9. At least one onsite and one offsite communications system; each system shall have a backup power source. All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where	Refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) and (b)(10) (Item 11 in Table 1). HDI will maintain primary and backup communications capabilities with the State of Michigan, Van Buren County, and the NRC. Because EPZs would be eliminated, the PNP PDEP would no longer describe provisions to communicate with Berrien and Allegan

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	consistent with the function of the governmental agency, these arrangements will include: E.9.a. Provision for communications with contiguous State/local governments ~~within the plume exposure pathway EPZ.~~ Such communications shall be tested monthly.	County. The onsite response facilities will be combined into a single facility, as described in the basis for exemption from 10 CFR 50, Appendix E, IV.E.8.a (i) (Item 34 in Table 2).
48	E.9.b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.	No exemption is requested.
49	E.9.c. ~~Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.~~	Analyses have been performed indicating that, approximately 12 months after shutdown, no credible accident at PNP can result in radiological releases requiring offsite protective actions, or in the event of beyond design basis accidents, 10 hours is available to take mitigative actions, and if needed, implement offsite protective actions using a CEMP approach. Therefore, there is no need for HDI to maintain the TSC, EOF, or field assessment teams. Additionally, there is no need to maintain or test committed provisions for communications between State and local emergency operations centers (EOCs) with these facilities. Also, refer to the basis for exemption for 10 CFR 50.47(b) and 50.47(b)(3) (Item 4 in Table 1). An onsite facility will continue to be maintained for PNP, from which effective command and control can be maintained during an emergency. Communication with State and local response organizations is maintained to coordinate onsite assistance, if required. The provisions remaining in 10 CFR Part 50, Appendix

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
		E, Section IV.E.9.a, b, and d include the necessary requirements for communications systems and testing.
50	E.9.d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the ~~nuclear power reactor control room, the onsite technical support center, and the emergency operations~~ facility. Such communications shall be tested monthly.	The functions of the Control Room, EOF, TSC, and OSC may be combined into one or more locations due to the smaller facility staff and the greatly reduced required interaction with State and local emergency response facilities. An onsite facility will continue to be maintained, from which effective direction can be given and effective control may be exercised during an emergency. HDI will maintain communications capability with the NRC. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
51	F. Training F.1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiological emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:	No exemption is requested.
52	F.1.i. Directors and/or coordinators of the plant emergency organization;	

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
53	F.1.ii. Personnel responsible for accident assessment, including control room shift personnel;	
54	F.1.iii. Radiological monitoring teams;	
55	F.1.iv. Fire control teams (fire brigades);	
56	F.1.v. Repair and damage control teams;	
57	F.1.vi. First aid and rescue teams;	
58	F.1.vii. Medical support personnel;	
59	F.1.viii. ~~Licensee's headquarters support personnel;~~	The number of staff at PNP during the decommissioning process will be small but commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. HDI will maintain a level of emergency response that does not require additional response by headquarters personnel. The on-shift and emergency response positions are defined in the PDEP and will be regularly tested through drills and exercises, audited, and inspected by HDI and the NRC. Therefore, exempting licensee's headquarters personnel from training requirements is considered to be reasonable. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
60	F.1.ix. Security personnel.	No exemption is requested.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
61	F.1. In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services~~/Civil Defense~~, local law enforcement personnel~~, local news media persons.~~	Because of the low probability of DBAs or other credible events to exceed the EPA PAGs, offsite emergency measures are limited to support provided by local police, fire departments and medical services, as appropriate. Local news media personnel no longer need radiological orientation training since they will not be called upon to support the formal Joint Information Center. The term "Civil Defense" is no longer a commonly used term and is no longer applicable as an example in the regulation.
62	F.2. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, ~~test the public alert and notification system,~~ and ensure that emergency organization personnel are familiar with their duties.[3]	Because of the low probability of DBAs or other credible events to exceed the EPA PAGs, and the available time to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to employ their CEMP to take protective actions, the public alert and notification system will not be used, and no testing of the system will be required. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1).
63	F.2.a.~~ A full participation exercise[4] which tests as much of the licensee, State, and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located. Nuclear power reactor licensees shall submit exercise scenarios under § 50.4 at least 60 days before use in a full participation exercise required by this paragraph 2.a.~~	HDI will continue to invite the State of Michigan and Van Buren County to participate in the periodic drills and exercises conducted to assess their ability to perform responsibilities related to an emergency at PNP to the extent defined by the PNP PDEP. Because the need for offsite emergency planning is relaxed due to the low probability of DBAs or other credible events that would be expected to result in an offsite radioactive release that would exceed the limits of EPA PAGs, and the available time for event mitigation, no offsite emergency plans will be in place to test.
64	F.2.a.(i)~~ For an operating license issued under this part, this exercise must be conducted within two years before the~~	

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~issuance of the first operating license for full power (one authorizing operation above 5 percent of rated power) of the first reactor and shall include participation by each State and local government within the plume exposure pathway EPZ and each state within the ingestion exposure pathway EPZ. If the full participation exercise is conducted more than 1 year prior to issuance of an operating licensee for full power, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.~~	The intent of submitting exercise scenarios at power reactors is to verify that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For permanently shutdown and defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning site. HDI considers PNP to be exempt from F.2.a.(i) - (iii) because PNP will be exempt from the umbrella provision of Section IV.F.2.a Item 63 of Table 2).
65	F.2.a.(ii) ~~For a combined license issued under part 52 of this chapter, this exercise must be conducted within two years of the scheduled date for initial loading of fuel. If the first full participation exercise is conducted more than one year before the scheduled date for initial loading of fuel, an exercise which tests the licensee's onsite emergency plans must be conducted within one year before the scheduled date for initial loading of fuel. This exercise need not have State or local government participation. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of the first full participation exercise, or if the Commission finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, the provisions of § 50.54(gg) apply.~~	
66	F.2.a.(iii) ~~For a combined license issued under part 52 of this chapter, if the applicant currently has an operating reactor at~~	

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~the site, an exercise, either full or partial participation,[5] shall be conducted for each subsequent reactor constructed on the site. This exercise may be incorporated in the exercise requirements of Sections IV.F.2.b. and c. in this appendix. If FEMA identifies one or more deficiencies in the state of offsite emergency preparedness as the result of this exercise for the new reactor, or if the Commission finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, the provisions of § 50.54(gg) apply.~~	
67	F.2.b. Each licensee at each site shall conduct a subsequent exercise of its onsite emergency plan every 2 years. ~~Nuclear power reactor licensees shall submit exercise scenarios under § 50.4 at least 60 days before use in an exercise required by this paragraph 2.b. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section.~~ In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, event classification, notification of offsite authorities, assessment of the onsite ~~and offsite~~ impact of radiological releases, ~~protective action recommendation development,~~	Refer to the basis for exemption for 10 CFR Part 50 Appendix E, Section IV.F.2.a (Item 63 in Table 2). The low probability of a design-basis accident or other credible events that would result in an offsite radioactive release that would exceed the EPA PAGs, and the available time for event mitigation at PNP during decommissioning render the TSC, OSC, and EOF unnecessary. The principal functions required by regulation can be performed at a single onsite location that does not meet the requirements of the TSC, OSC, or EOF. The onsite response facilities at PNP will be combined into a single facility. HDI will continue to conduct biennial exercises and will invite the State of Michigan, Van Buren County, and local support organizations (firefighting, law enforcement, and ambulance/medical services), to participate in periodic drills and exercises to assess their ability to perform responsibilities related to an emergency at PNP to the extent defined by the PNP PDEP.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~protective action decision making, plant~~ system repair and mitigative action implementation. During these drills, activation of all of the licensee's emergency response facilities ~~(Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF))~~ would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff in all participating facilities would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills may focus on the onsite exercise training objectives.	
68	F.2.c. ~~Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period. If two different licensees each have licensed facilities located either on the same site or on adjacent, contiguous sites, and share most of the elements defining co-located licensees,[6] then each licensee shall:~~	Refer to the bases for exemption for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 in Table 2) and 10 CFR Part 50, Appendix E, Section IV.F.2.a (Item 63 in Table 2).
69	F.2.c.(1) ~~Conduct an exercise biennially of its onsite emergency plan;~~	
70	F.2.c.(2) ~~Participate quadrennially in an offsite biennial full or partial participation exercise;~~	

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
71	F.2.c.(3) ~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~	
72	F.2.c.(4) ~~Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~	
73	F.2.c.(5) ~~Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~	
74	F.2.d. ~~Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in the ingestion pathway portion of exercises at least once every exercise cycle. In States with more than one nuclear power reactor plume exposure pathway EPZ, the State should rotate this participation from site to site. Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in a hostile action exercise at least once every cycle and should fully participate in one hostile action exercise by December 31, 2015. States with more than one nuclear power reactor plume exposure pathway EPZ should rotate this participation from site to site.~~	Refer to the basis for exemption for 10 CFR Part 50.47(b)(10) (Item 11 in Table 1).

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
75	F.2.e. Licensees shall enable any State or local government ~~located within the plume exposure pathway EPZ~~ to participate in the licensee's drills when requested by such State or local government.	Refer to the basis for exemption for 10 CFR Part 50.47(b)(10) (Item 11 in Table 1).
76	F.2.f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC,~~ in consultation with FEMA,~~ cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response.~~ The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.~~	FEMA is responsible for evaluating the adequacy of an offsite response exercise. No action is expected from State or local government organizations in response to an event at a decommissioning site other than firefighting, law enforcement, and ambulance/medical services. Letters of Agreement will continue to be in place for those services. Offsite response organizations will continue to implement actions to protect the health and safety of the public, as they would at any other industrial site, using a CEMP approach. Therefore consultation with FEMA is no longer necessary.
77	F.2.g. All exercises, drills, and training that provide performance opportunities to develop, maintain, or demonstrate key skills must provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified in a critique of exercises, drills, or training must be corrected.	No exemption is requested.
78	F.2.h. The participation of State and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to § 50.47(c)(1). In such cases, an exercise shall be held with	No exemption is requested.

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.	
79	F.2.i. Licensees shall use drill and exercise scenarios that provide reasonable assurance that anticipatory responses will not result from preconditioning of participants. ~~Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action~~. Exercise and drill scenarios as appropriate must emphasize coordination among onsite and offsite response organizations.	There are no DBAs or credible events that could occur that could result in radiological releases that exceed the EPA PAGs and the previously routine progression to General Emergency in power reactor site scenarios will not be applicable. Therefore, demonstration of a response to a wide spectrum of events is not expected. Also, refer to the basis for exemption for 10 CFR 50.47(b) (Item 1 in Table 1) detailing the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures and the basis for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 in Table 2), regarding "hostile action."
80	F.2.j.(i) ~~The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section.~~ (ii) ~~Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center.~~ (iii) ~~In each 8-calendar-year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to~~	Refer to the basis for exemption for 10 CFR Part 50, Appendix E, Section IV.F.2 (Item 62 in Table 2). Also, refer to the basis for exemption for 10 CFR 50.47(b)(5) (Item 6 in Table 1) regarding 10 CFR 50.155(b)(2) (formerly 10 CFR 50.54(hh)(2)) and 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 in Table 2), regarding "hostile action." Periodic drills and exercises will be completed to demonstrate ERO proficiency in key skills necessary to implement the principal functional areas of emergency response as applicable for the permanently defueled plant status. Critiques will follow each drill or exercise activity. HDI will continue to include the State of Michigan, Van Buren County, and local support organizations in the periodic drills and exercises to assess their ability to perform responsibilities

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	~~provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements:~~ (1) ~~hostile action directed at the plant site;~~ (2) ~~no radiological release or an unplanned minimal radiological release that does not require public protective actions;~~ (3) ~~an initial classification of, or rapid escalation to a Site Area Emergency or General Emergency;~~ (4) ~~Implementation of strategies, procedures, and guidance under § 50.155(b)(2); and~~ (5) ~~integration of offsite resources with onsite response.~~ (iv) ~~The licensee shall maintain a record of exercises conducted during each 8-year exercise cycle that documents the content of scenarios used to comply with the requirements of section IV.F.2.j of this appendix.~~ (v) ~~Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015.~~ (vi) ~~The first 8-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under 10 CFR part 52, the first 8-year exercise cycle begins in the calendar year of the initial exercise required by section IV.F.2.a of this appendix.~~	related to an emergency at PNP to the extent defined by the PNP PDEP.
81	G. Maintaining Emergency Preparedness	No exemption is requested.

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
	Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.	
82	H. Recovery Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.	No exemption is requested.
83	I. Onsite Protective Actions During Hostile Action ~~By June 20, 2012, for nuclear power reactor licensees, a range of protective actions to protect onsite personnel during hostile action must be developed to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.~~	Refer to the basis for exemption for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 in Table 2).
84	10 CFR 50 Appendix E V. Implementing Procedures No less than 180 days before the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, or the scheduled date for initial loading of fuel for a combined license under part 52 of this chapter, the applicant's or licensee's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in § 50.4.	No exemption is requested.

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	Regulation in Part 50, Appendix E	Basis for Exemption
85	10 CFR 50 Appendix E VI. Emergency Response Data System 1. The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance. 2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware shall be provided at each unit by the licensee to interface with the NRC receiving system. Software, which will be made available by the NRC, will assemble the data to be transmitted and transmit data from each unit via an output port on the appropriate data system.	The regulation that identifies the requirement to maintain the Emergency Response Data System (ERDS) is not applicable to nuclear power facilities that are permanently shut down. With the PNP reactor permanently shutdown and defueled, this system is no longer necessary to transmit safety system parameter data. No exemption is requested because this change in the ERDS data requirement is identified in 10 CFR Part 50, Appendix E, Section VI.2.

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
86	10 CFR 50 Appendix E Footnotes 3, 4, 5, and 6 are proposed for exemption. [4] ~~Full participation when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite local and State authorities and licensee personnel physically and actively take part in testing their integrated capability to adequately assess and respond to an accident at a commercial nuclear power plant. Full participation includes testing major observable portions of the onsite and offsite emergency plans and mobilization of State, local and licensee personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.~~ [5] ~~Partial participation when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite authorities shall actively take part in the exercise sufficient to test direction and control functions; *i.e.*, (a) protective action decision making related to emergency action levels, and (b) communication capabilities among affected State and local authorities and the licensee.~~ [6] ~~Co-located licensees are two different licensees whose licensed facilities are located either on the same site or on adjacent, contiguous sites, and that share most of the following emergency planning and siting elements:~~	HDI considers PNP to be exempt from Footnotes 4, 5, and 6 because PNP will be exempt from the umbrella provisions of Section F.2 (Item 62 in Table 2).

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**Regulation in Part 50, Appendix E**	**Basis for Exemption**
	~~a. Plume exposure and ingestion emergency planning zones;~~ ~~b. Offsite governmental authorities;~~ ~~c. Offsite emergency response organizations;~~ ~~d. Public notification system; and/or~~ ~~e. Emergency facilities.~~	

5.0 TECHNICAL EVALUATION

5.1 Accident Analysis Overview

10 CFR 50.82(a)(2) specifies that the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel after docketing the certifications for permanent cessation of power operations and permanent removal of fuel from the reactor vessel, in accordance with 10 CFR 50.82(a)(1). Following the termination of reactor operations at PNP, and the permanent removal of the fuel from the reactor vessel, the postulated accidents involving failure or malfunction of the reactor and supporting structures, systems, and components (SSCs) are no longer applicable. Summaries of the radiological accidents analyzed for the permanently shutdown and defueled condition, and supporting this request for exemptions, are presented below.

Section 5.0 of ISG-02 (Reference 1) indicates that site-specific analyses should demonstrate that: (1) the radiological consequences of remaining applicable DBAs would not exceed the limits of the EPA PAGs at the EAB; (2) in the event of a beyond design basis event resulting in the partial drain down of the SFP to the point that cooling is not effective, there is at least 10 hours (assuming an adiabatic heat up) from the time that the fuel is no longer being cooled until the hottest fuel assembly reaches 900°C; (3) adequate physical security is in place to assure implementation of security strategies that protect against spent fuel sabotage; and (4) in the unlikely event of a beyond design basis event resulting in a loss of all SFP cooling, there is sufficient time to implement pre-planned mitigation measures to provide makeup or spray to the SFP before the onset of a zirconium cladding ignition.

Table 3 contains a listing of seven analyses described in ISG-02 (Reference 1) that are expected to be evaluated by a decommissioning power reactor licensee requesting an exemption from emergency planning requirements. The table also contains a description of how HDI has addressed each of these analyses.

Table 3 **Interim Staff Guidance-02 Comparison**		
ISG-02 Analysis	**ISG-02 Description**	**IPEC Response**
1	Applicable design DBAs (i.e., fuel handling accident in the spent fuel storage facility, waste gas system release, and cask handling accident if the cask handling system is not licensed as single-failure-proof) (Indicates that any radiological release would not exceed the limits of EPA PAGs at EAB);	The postulated DBAs that remain applicable to PNP upon implementation of the requested exemptions are the FHA in the SFP, the liquid waste incident, the waste gas incident, and the postulated cask drop accidents. The results of these analyses indicate that within 90 days of the permanent cessation of power operations of the PNP reactor, the radiological consequences of the DBAs that remain applicable to PNP cannot exceed small fractions of the EPA PAG criterion of 1 rem Total Effective Dose Equivalent (TEDE) at the EAB. These analyses are described in Section 5.2 of this Enclosure.
2	Complete loss of SFP water inventory with no heat loss (adiabatic heatup) demonstrating a minimum of 10 hours is available before any fuel cladding temperature reaches 900 degrees Celsius from the time all cooling is lost (Demonstrates sufficient time to mitigate events that could lead to a zirconium cladding fire);	An analysis of the PNP SFP has been performed, which demonstrates that approximately 12 months after permanent cessation of power operations of the associated unit, a minimum of 10 hours is available before fuel cladding temperature of the hottest fuel assembly in the SFP reaches 900°C with a complete loss of SFP water inventory, assuming no heat loss (adiabatic heatup). Based on the results of the analysis, in the unlikely event of a beyond design basis event, 10 hours is available to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel and, if governmental officials deem warranted, for authorities to implement offsite protective actions using a comprehensive approach to emergency planning to protect the health and safety of the public before the hottest fuel assembly reaches the rapid oxidation temperature. This analysis is described in Section 5.3.1 of this Enclosure and is included in the Attachment to this Enclosure.
3	Loss of SFP water inventory resulting in radiation exposure at the EAB and control room; (Indicates that any release is less than EPA PAGs at EAB); and	An analysis of the PNP SFP has been performed to determine the radiological impacts of a complete loss of SFP water. It was determined that the gamma radiation dose rates at the EAB and the PNP Control Room would be less than the regulatory defined limits at one year after permanent cessation of power operations. This analysis is described in Section 5.3.2 of this Enclosure.
4	Considering the site-specific seismic hazard, either an evaluation demonstrating a	An analysis has been developed demonstrating that there is a High Confidence of Low Probability of Failure

Table 3

Interim Staff Guidance-02 Comparison

ISG-02 Analysis	ISG-02 Description	IPEC Response
	high confidence of a low-probability (less than 1 x 10^{-5} per year) of seismic failure of the spent fuel storage pool structure or an analysis demonstrating the fuel has decayed sufficiently that natural air flow in a completely drained pool would maintain peak cladding temperature below 565 degrees Celsius (the point of incipient cladding damage) (Indicates that any release is less than EPA PAGs at EAB).	(HCLPF) of the SFP. This analysis is provided as Attachment 4 to this Enclosure.
5	The analyses and conclusions described in NUREG-1738 are predicated on the risk reduction measures identified in the study as Industry Decommissioning Commitments (IDC) and Staff Decommissioning Assumptions (SDA), listed in Tables 4.1-1 and 4.1-2 of that document. The staff should ensure that the licensee has addressed these IDCs and SDAs for the decommissioning site if they are storing fuel in an SFP.	HDI has addressed the IDCs and SDAs. The IDCs and SDAs are addressed in Section 5.4 and Tables 4 (IDCs) and 5 (SDAs) of this Enclosure.
6	Verify that the licensee presents a determination that there is sufficient resources and adequately trained personnel available on-shift to initiate mitigative actions within the 10-hour minimum time period that will prevent an offsite radiological release that exceeds the EPA PAGs at the EAB.	The onsite mitigative actions in response to a loss of SFP cooling and to provide makeup water to the PNP SFP are incorporated into the PNP procedures and utilize adequately trained on-shift resources for implementation. There are multiple ways to initiate mitigative actions and add makeup water to the SFP within the 10-hour minimum period with or without entry to the SFP floor. Additionally, although the number of staff at PNP will be small with the PNP reactor permanently shutdown and defueled, the staffing level will be commensurate with the need to maintain the facility in a manner that is protective of public health and safety. The on-shift individuals described in the PDEP will be able to implement the necessary mitigation actions in approximately 2 hours.

Table 3 Interim Staff Guidance-02 Comparison		
ISG-02 Analysis	**ISG-02 Description**	**IPEC Response**
		Refer to SDA-2 in Table 5 of this Enclosure.
7	Verify that mitigation strategies are consistent with that required by the Permanently Defueled Technical Specifications or by retained license conditions.	HDI maintains procedures and strategies for the movement of any necessary portable equipment that will be relied upon for mitigating the loss of SFP water. These mitigative strategies were developed in response to 10 CFR 50.155(b)(2) (formerly 10 CFR 50.54(hh)(2)), and are maintained in accordance with License Condition 6.b of PNP Renewed License No. DPR-20. These diverse strategies provide defense-in-depth and ample time to provide makeup water or spray to PNP SFP prior to the onset of zirconium cladding ignition when considering very low probability beyond design basis events affecting the SFP. These diverse strategies are described in more detail in the HDI response to SDA-4 in Table 5 of this Enclosure.

5.2 Consequences of Design Basis Events

As described in Amendment No. 272, issued by the NRC on May 13, 2022 (PNP- Issuance of Amendment No. 272 Re: Permanently Defueled Technical Specifications), the applicable remaining DBAs are (1) the FHA in the SFP, (2) the Liquid Waste Incident, (3) a Waste Gas Incident, and (4) a Postulated Cask Drop Accident (Reference 15).

The DBAs that remain applicable to PNP are discussed in the following paragraphs.

5.2.1 Fuel Handling Accident

Following permanent cessation of power operations and permanent removal of fuel from the PNP reactor, an FHA in the reactor cavity is no longer applicable because all irradiated spent fuel will either be stored in the PNP SFP or an ISFSI. Therefore, because an FHA can only occur during movement of spent fuel in the SFP, the FHA event is limited to the SFP.

The FHA analysis assumed 22.5 feet of water above the stored fuel, which resulted in an effective decontamination factor of 183.07 and an overall decontamination factor for elemental iodine of 252 (Reference 16). The FHA utilizes the Alternate Source Term (AST) methodology described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 17).

The analysis demonstrates that after a decay time of 60 days following permanent cessation of power operations of the PNP reactor, with no credit taken for the operability of mitigating structures, systems, or components (SSCs), the FHA in the SFP results in a dose of 0.014 rem TEDE at the EAB (Reference 18). This is less than the EPA PAG criterion of 1 rem TEDE and below the 10% EPA PAG threshold for declaration of a Site Area Emergency (SAE), based on the NRC guidance provided in NEI 99-01, Rev.6 (Reference 7).

5.2.2 Liquid Waste Incident

A liquid tank failure remains a viable accident following the reactor being permanently defueled since liquid tanks may continue to store radioactive liquid. The accidents discussed in the DSAR include an accidental discharge to the circulating water discharge canal, or failure of the primary system makeup storage tank or the utility water storage tank. The primary makeup storage tank and the utility water storage tank have administrative controls that maintain tank activity concentration such that 10 CFR Part 20, *Standards for Protection Against Radiation*, dose limits would not be exceeded in the event of a tank failure. These concentration limits will be maintained in the permanently defueled condition.

HDI has concluded that the PNP design and administrative controls ensure that radioactive liquid leakage or spillage will be retained within the facility or within 10 CFR Part 20 dose limits. Also, administrative controls and automatic interlocks, together with the fail-safe design of the instrumentation and control devices, provide assurance against any discharge of liquid wastes to the environs in excess of 10 CFR Part 20 limits and would not approach the EPA PAG criteria of 1 rem TEDE after a 90-day fuel decay period.

5.2.3 Waste Gas Incident

The PNP DSAR evaluates the accidental release of waste gas. The atmospheric dispersion coefficients and the source term for the FHA, discussed in Section 5.2.1 of this Enclosure, bound those of the design basis gas decay tank rupture (GDTR).

The volume control tank rupture accident is no longer applicable in the permanently defueled condition because primary coolant letdown will no longer be required to support primary coolant system operation. In addition, inputs into the volume control tank rupture accident discussed in UFSAR Section 14.21.2, such as letdown flow and dose equivalent iodine-131 requirements will no longer be applicable in the permanently defueled condition. In the event that the volume control tank continues to hold reactor coolant fluid in the permanently defueled condition, the source term would be lower than during normal operation due to radioactive decay. In addition, the primary coolant iodine and noble gas concentrations released to the atmosphere from the volume control tank after 17 days of decay would be significantly less than the source term from the FHA with 17 days of decay and the CR doses from the FHA.

Therefore, it can be concluded that the dose consequences of the FHA bound the dose consequences of the GDTR with the same decay period.

5.2.4 Postulated Cask Drop Accident

The PNP DSAR evaluates the postulated cask drop accidents. The analysis included a scenario in which a cask is dropped onto spent fuel which has decayed for 90 days. The scenario assumes the Fuel Handling Building (FHB) charcoal filter is not operating and all radiation is released unfiltered from the FHB. The accident results in a dose of 0.08 rem at the EAB 90 days following permanent cessation of power operations of the PNP reactor. This is less than the EPA PAG criterion of 1 rem TEDE and below the 10% EPA PAG threshold for declaration of a SAE, based on the NRC guidance provided in NEI 99-01, Rev.6 (Reference 7).

5.3 Consequences of a Beyond Design Basis Event

5.3.1 Hottest Fuel Assembly Adiabatic Heat Up

An adiabatic heatup analysis was performed comparing the heat load limits for the hottest fuel assembly and for a 2X2 group of assemblies stored in the PNP SFP to a criterion proposed in Commission Paper SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants," (Reference 19) that is applicable to offsite emergency response for nuclear power reactors in the decommissioning process. This criterion considers the time for the hottest assembly to heat up from 30°C to 900°C adiabatically. A heat up time of 10 hours from the time the spent fuel is uncovered, was determined to be sufficient time to take mitigating actions and, if necessary, offsite protective measures without offsite emergency preplanning addressing the facility.

The analysis for the PNP SFP for beyond design basis events demonstrates that approximately 12 months after shutdown, a minimum of 10 hours is available before the fuel cladding temperature of the hottest fuel assembly in the SFP reaches 900°C with a complete loss of SFP water inventory. As stated in NUREG-1738 (Reference 12), 900°C is an acceptable temperature to use for assessing the onset of fission product release under transient conditions (to establish the critical decay time for determining availability of 10 hours to evacuate) if fuel and cladding oxidation occurs in air. Based on the results of the analysis, in the unlikely event of a beyond design basis event, 10 hours is available to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel and, if governmental officials deem warranted, for authorities to implement offsite protective actions using a comprehensive approach to emergency planning to protect the health and safety of the public before the hottest fuel assembly reaches the rapid oxidation temperature.

Because of the time it would take for the adiabatic heat up to occur, there is ample time to respond to any partial drain down event that might cause such an occurrence by restoring cooling or makeup or providing spray. As a result, the likelihood that such a scenario would progress to a zirconium fire is not deemed credible.

Attachment 1 to the Enclosure contains a proprietary version of this analysis.

Attachment 2 to the Enclosure contains a non-proprietary, redacted version of the analysis.

5.3.2 Spent Fuel Pool Drain Down Event

NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (Reference 20) Supplement 1, Section 4.3.9.2, identifies that a SFP drain down event is a beyond design basis event. The analyses provided in Attachments 1 (proprietary version) and 2 (non-proprietary version) to this Enclosure demonstrated that, under adiabatic conditions, a significant release of radioactive material from the spent fuel is not possible within 10 hours from the time the spent fuel is uncovered after approximately 12 months following the permanent cessation of power operation. However, the potential exists for radiation exposure if shielding of the spent fuel in the PNP SFP is lost.

The SFP water and the concrete SFP structure serve as radiation shielding. Therefore, a loss of water shielding above the fuel could increase the offsite radiation levels because of the gamma rays streaming up out of the pool being scattered back to a receptor at the site boundary.

In preparation for the originally planned October 2018 shutdown of the PNP reactor, the radiological consequences of a postulated complete loss of SFP water at the EAB and Control Room were analyzed. It was determined that the gamma radiation dose rate at the EAB would be limited to small fractions of the EPA PAG exposure levels. Based on the analysis, the dose rate to a receptor at the EAB and the limiting dose rate in the PNP Control Room one year after shutdown are less than 0.20 mrem/hour (hr) and 2.5 mrem/hr, respectively. It was determined that this analysis is applicable to the May 2022 shutdown of the PNP reactor (Reference 21).

The EPA PAGs were developed to respond to a mobile airborne plume that could transport and deposit radioactive material over a large area. In contrast, the radiation field formed by scatter from a drained SFP would be stationary rather than moving and would not cause transport or deposition of radioactive materials. The extended period required to exceed the integrated EPA PAG limit of 1 rem TEDE would allow sufficient time to develop and implement onsite mitigative actions and provide confidence that additional offsite measures could be taken without planning if efforts to reestablish shielding over the fuel are delayed.

5.4 Design and Operational Characteristics of the Spent Fuel Pool

Although the limited scope of DBAs and the associated dose consequences, and the significant time available to complete actions necessary to mitigate a beyond design basis accident that remain applicable to PNP justify a reduction in the necessary scope of emergency response capabilities, the Industry Decommissioning Commitments (IDCs) and Staff Decommissioning Assumptions (SDAs) contained in NUREG-1738 (Reference 12) were also evaluated.

NUREG-1738 contains the results of the NRC staff’s evaluation of the potential accident risk in SFPs at decommissioning plants in the United States. As stated therein, the study was undertaken to support development of a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking. The NRC staff performed analyses and sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Reference 22). The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis.

The study found that the risk of a potential SFP accident at decommissioning plants is low and well within the Commission's Safety Goals. The risk is low because of the very low likelihood of a zirconium fire (resulting from a postulated irrecoverable loss of SFP cooling water inventory.

NUREG-1738 provided the following assessment:

> *"The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments, this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a 'small change' in risk, consistent with the guidance of RG 1.174. Figures ES-1 and ES-2 [in NUREG-1738] illustrate this finding. The change in risk due to relaxation of offsite EP is small because the overall risk is*

low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors, additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1x10^{-5}$ per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at $2x10^{-7}$ per year)."

The Executive Summary in NUREG-1738 states, in part,

"The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These characteristics are identified in the study as IDCs and SDAs. Provisions for confirmation of these characteristics would need to be an integral part of rulemaking."

The IDCs and SDAs are listed in Tables 4.1-1 and 4.1-2, respectively, of NUREG-1738 (Reference 12). The following tables indicate how the PNP SFP meets or compares with each of these IDCs (Table 4) and SDAs (Table 5).

5.5 Consequences of a Beyond-Design Basis Earthquake

NUREG-1738 (Reference 12) identifies beyond design basis seismic events as the dominant contributor to events that could result in a loss of SFP coolant that uncovers fuel for plants in the Central and Eastern United States. Additionally, NUREG-1738 identifies a zirconium cladding fire resulting from substantial loss-of-water inventory from the SFP, as the only postulated scenario at a decommissioning plant that could result in significant offsite radiological release. The scenarios that lead to this condition have very low frequencies of occurrence (i.e., on the order of one to tens of times in a million years) and are considered beyond design basis events because the SFP and attached systems are designed to prevent a substantial loss of coolant inventory under accident conditions. However, the consequences of such accidents could potentially lead to an offsite radiological dose in excess of the EPA PAGs (Reference 4) at the EAB.

The risk associated with zirconium cladding fire events decreases as the spent fuel ages because, as the spent fuel ages, the decay time increases, the decay heat decreases, and the short-lived radionuclides decay away. As the decay time increases, the overall risk of zirconium cladding fire continues to decrease due to two factors: (1) the amount of time available for preventative actions increases, which reduces the probability that the actions would not be successful; and (2) the increased likelihood that the fuel is able to be cooled by air, which decreases the reliance on actions to prevent a zirconium fire. The results of the research conducted for NUREG-1738 and NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," (September 2014) (Reference 23) suggests that, while other radiological consequences can be extensive, a postulated accident scenario leading to a SFP zirconium fire, where the fuel has had significant decay time, has little potential to cause offsite early fatalities due to dose, regardless of the type of offsite response (i.e., formal offsite radiological emergency preparedness plan or Comprehensive Emergency Management Plan).

The purpose of NUREG-2161 (Reference 23) was to determine if accelerated transfer of older, colder spent fuel from the SFP at a reference plant to dry cask storage significantly reduces the risks to public health and safety. The study states:

> *"This study's results are consistent with earlier research studies' conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering spent fuel."*

NUREG-2161 also states:

> *"The study shows the likelihood of a radiological release from the spent fuel pool after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower."*

NUREG-2161 also states:

> *"If a leak and radiological release were to occur, this study shows that individuals cancer fatality risk for a member of the public is several orders of magnitude lower than the Commission's Quantitative Health Objective of two in one million (2 x 10-6 /year). For such a radiological release, this study shows public and environmental effects are generally the same or smaller than earlier studies."*

Additionally, the study evaluated the potential benefits of strategies required by 10 CFR 50.54(hh)(2) [relocated to 10 CFR 50.155(b)(2)] following the September 11, 2001, attacks. The study demonstrates that successful implementation of mitigation strategies significantly reduces the likelihood of a release from the SFP in the event of a loss of cooling water. The likelihood of a SFP release was equally low for both high- and low-density fuel loading. This is because high- and low-density fuel loading contains the same amount of new, hotter spent fuel recently moved from the reactor to the SFP. In the unlikely event of an earthquake-induced SFP leak, the likelihood of fuel heatup leading to a release was more strongly affected by the fuel loading pattern rather than the total amount of fuel in the SFP.

The results of NUREG-2161 are consistent with earlier research conclusions that SFPs are robust structures that are likely to withstand severe earthquakes without leaking.

As described in Item 4 of Table 3 of this Enclosure, an analysis has been developed demonstrating a HCLPF of the SFP. Based on this analysis, the probability of seismically induced structural failure of the SFP and rapid loss of inventory is less than the generic bounding value of 1×10^{-5} per year. This analysis is provided as Attachment 4 to this Enclosure.

6.0 CONCLUSION

HDI has concluded, based on the analyses and actions described above, that the health and safety of the public are protected with PNP in the permanently shutdown and defueled condition. Approval of the exemptions requested above would not present an undue risk to the public or prevent appropriate response in the event of an emergency at PNP.

Based on the above, it has been demonstrated that approximately 12 months after permanent cessation of power operations of the PNP reactor, no credible DBA or beyond design basis accident can result in radiological releases requiring offsite protective actions. Additionally,

there is sufficient time, resources, and personnel available to initiate mitigative actions to prevent a radiological release that exceeds EPA PAG doses offsite.

TABLE 4

Industry Decommissioning Commitments (IDCs) Comparison

IDC	Industry Commitments	Response
1	Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).	The Palisades crane design is consistent with this commitment. Heavy load lifts in and around the area of the SFP are performed by the Fuel Pool Building Crane (L-3). The design of this crane is single-failure-proof, as the main hoist meets the single failure criteria in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Reference 24), and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" (Reference 25). Because the L-3 main hoist is single-failure-proof, the likelihood of dropping the spent fuel casks in and around the SFP is extremely low, and an accidental load drop is not considered to be a credible event, such that condition 5.1.2(1) of NUREG-0612 is satisfied and analysis of cask drop accidents in accordance with condition 5.1.2(4) of NUREG-0612 is not required. The PNP procedures provide instructions for lifting activities to meet the guidance provided in NUREG-0612. Although the main hoist of the spent fuel crane is designed and operated in accordance with single-failure-proof criteria for cask handling activities, there may be situations in which lifting devices used with the main hook do not meet these requirements or single-failure-proof features of the main hoist become disabled. In these situations, the crane would no longer meet single-failure-proof requirements, and load drops could be postulated. Therefore, cask drop analyses were performed to document the consequences of postulated fuel transfer cask drop accidents in the fuel handling area at PNP. This analysis is described in Section 5.2.4 of this Enclosure.
2	Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.	Palisades procedures are in place to ensure onsite and offsite resources can be brought to bear during an event, including the following: • Palisades Site Emergency Plan • Emergency Management Guidelines • EI-1 – Emergency Classification and Actions • EI-2.2 – Emergency Staff Augmentation • EI-3 – Communications and Notifications • FSG-100 – Beyond Design Bases External Event (BDBEE) With an Extended Loss of Offsite Power (ELAP) and Emergency Response • FSG-101 – BDBEE / EP Communications • FIG-1 – FLEX Generator Staging and Operation • FIG-2 – FLEX Pump Staging and Operation

TABLE 4

Industry Decommissioning Commitments (IDCs) Comparison

IDC	Industry Commitments	Response
		The procedures listed above (or equivalent) and associated training will be updated as necessary to reflect the permanently shut down and defueled condition. Prior to permanent shut down and permanent removal of fuel from the PNP reactor vessel, on-shift operations personnel, including Certified Fuel Handlers (CFHs) and Non-Certified Operators (NCOs), will be appropriately trained on the relevant procedures and on the various actions needed to provide makeup to the SFP. Following permanent cessation of power operations, maintaining SFP cooling and inventory would be the highest priority activity. Therefore, the personnel needed to perform these actions will be always available. Finally, periodic Emergency Plan drills are conducted with opportunities for offsite response organization participation to maintain proficiency in response to a plant event.
3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.	Palisades procedures are in place to establish and maintain communications between onsite and offsite organizations during severe weather and seismic events, including the following: • Palisades Site Emergency Plan • Emergency Management Guidelines • EI-1 – Emergency Classification and Actions • EI-3 – Communications and Notifications • EI-6.7 – Plant Site Meteorological System • EI-6.8 – Backup and Supplemental Meteorology • EI-17 – Compensating Measures for OOS EAL Equipment and Listing of Non-EAL Equipment Important for Emergency Preparedness • AOP-38 – Acts of Nature • FSG-101 – BDBEE / EP Communications The procedures listed above (or equivalent) will be updated as necessary to reflect the permanently shut down and defueled condition.
4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The	The following procedures provide guidance for communication with offsite resources which may be used to support mitigation strategies for SFP damage and water supply: • Palisades Site Emergency Plan • Emergency Management Guidelines

TABLE 4 Industry Decommissioning Commitments (IDCs) Comparison		
IDC	**Industry Commitments**	**Response**
	plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.	• EI-1 – Emergency Classification and Actions • EI-3 – Communications and Notifications • EI-17 – Compensating Measures for OOS EAL Equipment and Listing of Non-EAL Equipment Important for Emergency Preparedness • AOP-38 – Acts of Nature • FSG-11 – Alternate SFP Makeup and Cooling • FIG-2 – Flex Pump Staging and Operation • FIG-3 – Alternate SFP Makeup and Cooling The procedures listed above (or equivalent) will be updated as necessary to reflect the permanently shut down and defueled condition.
5	SFP instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for SFP temperature, water level, and area radiation levels.	Palisades maintains a Technical Specification that the Spent Fuel Pool (SFP) water level be maintained ≥ 647 ft elevation. SFP level is monitored by a local level indicator (ruler), which provides indication in the Control Room via a camera/monitor. A Control Room low-level alarm at 646' is provided via Spent Fuel Pool Level Switch LS-0924. SFP temperature is monitored by temperature indicating alarms TIA-0925, Spent Fuel Pool Temp Alarm Indicator, and TIA-0926, Spent Fuel Pool Temp Alarm Indicator, located in the Spent Fuel Pool area. Both TIA-0925 and TIA-0926 provide input to the PPC with workstations in the Control Room and include urgent PPC alarms at 140°F with audible annunciation as well as a PPC warning alarms at 125 °F. Additionally, temperature indicating alarm TIA-0917, Discharge From Spent Fuel Pool Heat Exchanger, also located in the SFP area provides an alarm at 115°F on the C-40 Panel in the Auxiliary Building which also results in a Radwaste Panel C-40 Off Normal alarm in the Control Room. Area radiation monitors RIA-5709 and RIA-2313, Spent Fuel Pool Criticality Monitors, monitor the SFP area with readouts and a common alarm in the Control Room. Palisades currently has instrumentation in the SFP that meets the intent of this IDC.
6	SFP seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self-limiting to	The PNP SFP seal around the tilt pit gate meets the intent of the IDC. Seal failure would result in a self-limiting loss of inventory of approximately 6-7 feet, well above the top of the active fuel. In

TABLE 4

Industry Decommissioning Commitments (IDCs) Comparison

IDC	Industry Commitments	Response
	leakage or otherwise engineered so that drainage cannot occur.	addition, the bottom elevation of the gate seal is above the top of the active fuel, therefore, leakage by the gates could not lead to fuel uncovery.
7	Procedures or administrative controls to reduce the likelihood of rapid drain down events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.	PNP System Operation Procedure (SOP-27), "Fuel Pool System," provides the directions for filling and draining the SFP and includes limits on the SFP level. The suction, discharge, skimmer and tilt pit fill lines enter the SFP at elevations greater than 644' 5" to assure that failures of downstream piping cannot result in unacceptable drainage of SFP water inventory. Failure of the outlet piping system would result in draining of the SFP to the outlet level which still maintains an adequate level of water for shielding and cooling requirements. Such a failure could occur because of a wind or tornado generated missile striking a portion of the SFP cooling pump P-51B discharge piping that extends above the SFP building floor. Failure of the inlet piping would result in no loss of water from the SFP as there is no downcomer by which a siphon could be started. [FSAR Section 9.4.3.1] Margins to the Top of Fuel for Piping Failures: • SFP Cooling Discharge Line: 23.5 ft • SFP Cooling Suction Line: 20.4 feet • SFP Cooling Tilt Pit Fill Line: 24.0 feet • SFP Cooling Skimmer Line: 23.5 feet • SFP Bulkhead Gate Failure: 17.0 feet SFP structure integrity, including the concrete SFP and the SFP steel liner is required for maintaining coolant inventory. Detection of gross leakage during normal operation can be accomplished with the SFP level alarm. Leakage detection for SFP stainless steel liners is a manual process of observing flow from channels between the liner and concrete through telltales.
8	An onsite restoration plan will be in place to provide repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote alignment of the	The following procedures provide guidance for providing makeup water to the SFP: • EMG – Emergency Management Guidelines • FSG-11 – Alternate SFP Makeup and Cooling • FIG-2 – Flex Pump Staging and Operation • FIG-3 – Alternate SFP Makeup and Cooling • SOP-27 – Fuel Pool System

TABLE 4

Industry Decommissioning Commitments (IDCs) Comparison

IDC	Industry Commitments	Response
	makeup source to the SFP without requiring entry to the refuel floor.	• AOP-26 – Loss of Spent Fuel Pool Cooling • SAG-1 – Initial Response • SAG-2 – TSC Recommending Strategies • SAG-3 – Site Capabilities • SAG-10 – Control SFP Level • TSG-1 – Instrumentation (Att. M for SFP Level Indication) • TSG-3 – Site Capabilities • TSG-4 – Benefit Consequence (Att. H for SFP) • TSG-5 – Computational Aids (Att. G for SFP) PNP procedure AOP-26, Loss of SFP Level would likely be the initial procedure entered for this scenario. This procedure has direction for: • Loss of SFP Inventory (Att. 1); • Loss of SFP Cooling (Att. 2); • Loss of SFP Inventory Due to SFP Structural Failure (Att. 3); and • Severe Accident Guideline SAG-10, Control SFP Level. The inventory loss sections direct the user to use: • Primary Makeup Water (PMW) using SOP-27, Fuel Pool System; • Fire Water using local firehoses or emergency fill per SOP-27; and • Service Water via PCSO-5, Alternate Source for Charging to the Primary Coolant System (PCS). This has a procedure section 5.3 which provides a source of makeup water to the SFP from the Service Water System (SWS) given the loss of normal makeup water sources following a seismic event. This requires access to the SFP floor. If access to the SFP floor is available, one method is via the normal method of filling from PMW via hose. Additionally, an emergency fill option from the Fire Protection System (FPS) via a spool piece in the SFP Hx Room as described section 7.2.3. There are multiple ways to add makeup water to the SFP with or without entry to the refuel floor. PNP procedure FSG-11 provides actions to make up and cool the SFP by alternate means using FLEX strategies during an extended loss of all AC power (ELAP). Further guidance is within FSG-3, which provides alternate SFP makeup and cooling. SFP makeup water is provided via the portable FLEX Pump Manifold staged at the Turbine Building 590' elevation. SFP makeup

TABLE 4

Industry Decommissioning Commitments (IDCs) Comparison

IDC	Industry Commitments	Response
		can be accomplished through one of two methods, including direct fill from a hose routed to the SFP or through a hard pipe connection in SFP Heat Exchanger Room. Further, FSG-3 specifies that the installation of the SFP spray monitor nozzles and direct fill should be given priority over the hard pipe fill connection due to expected SFP area high radiation levels if SFP level cannot be maintained. However, this method requires access to the SFP floor area; the hard pipe method does not. For a loss of SFP cooling event, AOP-26 Attachment 2 provides cooling options and the makeup options described above. In worst-case scenarios HDI employs the Severe Accident Management Guidelines (SAMGs) and use of the Emergency Management Guidelines (EMGs). Severe Accident Guideline SAG-1, Initial Response step 17 provides guidance if the SFP is not intact and/or level is low or lowering due to an event. All response actions to restore SFP level are the options described above (includes plans for both an available access to the SFP floor as well as cases where access is not available). TSG-3, Site Capabilities provides options if all the normal means are unavailable. Two installed diesel-driven fire pumps and one motor-driven fire pump are available and can provide 1,500 gallons per minute make-up from the facility intake via hard pipe or hose stations. In addition, two onsite FLEX pumper units with a capacity of 1,000 gallons per minute each can provide make-up from the facility intake or from Lake Michigan directly.
9	Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.	The PNP SFP design has no drains in the main pool. Failure of the outlet piping would result in a loss of water only to the level of the outlet piping (located at 647' 6" ft elevation, 6" below skimmer), which would still maintain an adequate amount of water for shielding and cooling requirements. Failure of the inlet piping would result in a loss of water to the level of the inlet opening (located at 644' 5" ft elevation, 3' 7" below skimmer), which would also maintain an adequate water level in the SFP for shielding and cooling. The top of the active fuel is located at the 624’ elevation (24’ below the skimmer). The PNP Technical Specifications, including the Permanently Defueled Tech Specs (PDTS), 3.7.14 requires the SFP level be maintained >647 ft. elevation during movement of irradiated fuel assemblies or a fuel cask in the SFP (Reference 15). The SFP level is allowed to be below the 647 ft elevation to support fuel cask movement if the displaced water level with the cask submerged raises SFP level to the >647 ft. elevation.

TABLE 4 Industry Decommissioning Commitments (IDCs) Comparison		
IDC	**Industry Commitments**	**Response**
		Administrative limitations, such as restrictions on heavy loads movements are controlled by Operating Requirements Manual (ORM) Section 3.21, and PNP procedure FHS-M-23, Movement of Heavy Loads in the SFP Area. Movement of all heavy loads (>1300 lbs.) in the vicinity of the SFP is performed in accordance with FHS-M-23, which provides extensive instructions to ensure that the requirements of NUREG-0612 are met for heavy loads. Dry cask loading operations will have additional management oversight and additional administrative controls in place as a High Integrated Risk activity. An Infrequent Evolution Brief would also be held. Palisades procedure FHS-M-39B, Fuel Loading and DSC Sealing Operations for NUHOMS Dry Fuel Loading Operations, contains numerous requirements to control this operation. These include, but are not limited to: • SFP water level should be adjusted to 13-14 inches below the top of the skimmer plate in preparation for filling the Dry Storage Cask (DSC) with SFP water (when filling DSC with pool water, SFP level will go down between 3" and 4") • Prior to cask loading all required persons will have completed the training program for the Cask System per PNP Administrative Procedure 5.26, "Independent Spent Fuel Storage Installation Training and Certification Program." • The SFP level shall be monitored hourly (via television monitor or locally) whenever the DSC is in the SFP, and fuel is in the DSC to ensure that the SFP is not overflowing and that the water level is not unintentionally rising. • With the DSC slightly suspended in the SFP, inspect lifting rig and load carrying members for any signs of overloading and distortion PNP Procedure FHS-M-34, Unloading the Multi-Assembly Sealed Basket (MSB) contains further requirements: • Care must be taken to ensure the SFP-Impact Limiting Pad (ILP) will not be set on the SFP liner weld seams (setting on weld seams may cause fuel pool leakage) • Use of an ILP in the SFP for setting the cask on • Control Room communications established and verified prior to any lift in the SFP Additional Dry Cask operations in the SFP are controlled under several other PNP FHS-M procedures. Review of these procedures confirms that there are no Dry Cask-related SFP operations which could result in a rapid drain down event.

TABLE 4 Industry Decommissioning Commitments (IDCs) Comparison		
IDC	**Industry Commitments**	**Response**
10	Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.	PNP practices align with this IDC. See the discussion in IDC-8 above for the methods to align makeup sources to the SFP without requiring entry to the SFP floor. If access to the SFP floor is available, additional options exist, and are also described in IDC-8. For the pumps identified in this section, Preventative Maintenance (PM) measures shall be in place to ensure that they will perform as required when placed in service. These PMs shall be implemented and scheduled in accordance with the PM Program. PNP procedures provide guidance for the conduct of operations administrative processes, and specifies the authority and responsibilities of individuals to ensure the requirements of federal regulations, industry good practices, and standards are met, including adherence to operating procedures. Performance of the procedures identified above will be in accordance with these requirements.

TABLE 5

Staff Decommissioning Assumptions (SDAs) Comparison

SDA	Staff Assumptions	Response
1	Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP.	The PNP SFP cooling system (SFPCS) is as capable as that assumed in Section 3.0 of NUREG-1738. The PNP SFPCS is a seismically analyzed system containing two motor-driven cooling pumps and two heat exchangers in series. The heat exchangers transfer decay heat to the Component Cooling Water system, then to the Ultimate Heat Sink. A filtration system is manually operated to maintain SFP cleanup. In addition, PNP has two diesel-driven and one motor-driven pump in the fire-water system and three motor-driven pumps in the service water system capable of delivering inventory to the SFP.
2	Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.	Currently, HDI performs walk-downs of the SFP system each shift as driven by Operator rounds and by surveillance testing procedures. These include local and remote SFP level (both physical level and two redundant remote indicators), and SFP gate inner and outer N2 pressures. There are multiple methods to alert the Control Room of a SFP event, including alarms and redundant SFP water level indicators. Walkdown of the SFP system will remain in place following permanent cessation of power operations. Procedures identified above (or equivalent) will be in place and updated as necessary to reflect the permanently shut down condition. PNP procedures meet the requirements of this SDA by providing guidance on the capability and availability of permanent and portable make up sources. AOP-38, Acts of Nature, requires inspection of plant areas, inherently including the SFP and dry spent fuel storage casks following a seismic event. See the discussion for IDC-8 in Table 4 of this Enclosure, for the procedural direction for methods to diagnose the loss of cooling and/or inventory with description of steps, and sequences to establish make up to the SFP. This discussion also provides direction in a Beyond Design-Basis External Event (BDBEE).
3	Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring a general emergency.	PNP PDTS 3.7.14 requires that the SFP water level be maintained ≥ 647 ft elevation (Reference 15). SFP level is monitored by a local level indicator (ruler), which provides indication in the Control Room via a camera/monitor. A Control Room low-level alarm at 646' is provided via Spent Fuel Pool Level Switch LS-0924. SFP temperature is monitored by temperature indicating alarms TIA-0925, Spent Fuel Pool Temp Alarm Indicator, and TIA-0926, Spent Fuel Pool Temp Alarm Indicator, located in the SFP area. Both TIA-0925 and TIA-0926 provide input to the PPC with workstations in the Control

TABLE 5 Staff Decommissioning Assumptions (SDAs) Comparison		
SDA	**Staff Assumptions**	**Response**
		Room and include urgent PPC alarms at 140°F with audible annunciation as well as PPC warning alarms at 125°F. Additionally, temperature indicating alarm TIA-0917, Discharge From Spent Fuel Pool Heat Exchanger, also located in the SFP area provides an alarm at 115°F on the C-40 Panel in the Auxiliary Building which also results in a Radwaste Panel C-40 Off Normal alarm in the Control Room. Regarding the declaration of a general emergency, PNP will employ permanently defueled EALs using an NRC-approved EAL Scheme, based on Appendix C of NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6. Station conditions will not have the capacity to reach any threshold requiring the declaration of a General Emergency. PNP currently has instrumentation in the SFP that meets the intent of this SDA.
4	Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.	There are no drain paths that could lower the SFP level by more than 15 feet below the normal SFP operating level. Suction and discharge piping for the SFPCS are greater than 20 feet above the top of the active fuel and approximately 3 feet below the normal level, such that a loss of 15 feet cannot occur.
5	Load Drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.	The PNP SFP design is in alignment with this description. Heavy load lifts in and around the area of the SFP are performed by the L-3 crane. The design of the L-3 is single-failure-proof as noted in the response to IDC-1 in Table 4 of this Enclosure. Nonetheless, load drop consequence analyses have been performed, as previously described (in the response to IDC-1 in Table 4 of this Enclosure).
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to	PNP conducted an evaluation to assess seismically induced structural failure and rapid loss of inventory. The assessment demonstrates that the risk of a SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in

TABLE 5

Staff Decommissioning Assumptions (SDAs) Comparison

SDA	Staff Assumptions	Response
	this study [NUREG-1738]. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study (<1 x10^{-5} per year including non-seismic events).	NUREG-1738 (1 x 10^{-5} per year including non-seismic events). This analysis is provided as Attachment 4 to this Enclosure.
7	Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.	With TS Amendment 207 to TS 3.7.16 issued by the NRC on 02/26/2002, Palisades no longer credited Boraflex as a neutron absorber in our Region II racks. With the removal of credit for Boraflex in the Criticality Safety Analysis (CSA), surveillances for Boraflex monitoring were discontinued. As there is no credit taken for Boraflex in the CSA, there is no need to resume surveillance testing during decommissioning. In a similar manner, for the one remaining Region I rack utilizing Carborundum as its neutron absorber, the Carborundum material is no longer credited in the current CSA. Therefore, no surveillance of the neutron absorber is needed for this rack. The remainder of the Region I Carborundum racks were replaced with new racks containing Metamic as their neutron absorber in a re-rack project in 2013. A coupon monitoring surveillance program is utilized to ensure that the Metamic material does not degrade beyond limits. This surveillance program will continue as long as there is fuel in the Region I racks.

7.0 JUSTIFICATION FOR EXEMPTIONS AND SPECIAL CIRCUMSTANCES

10 CFR 50.12 states that the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of 10 CFR Part 50 which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. 10 CFR 50.12 also states that the Commission will not consider granting an exemption unless special circumstances are present. As discussed below, this request for exemptions satisfies the provisions of Section 50.12.

7.1 Exemptions

A. The exemptions are authorized by law

10 CFR 50.12 allows the NRC to grant exemptions from the requirements of 10 CFR Part 50. The proposed exemptions would not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemptions are authorized by law.

B. The exemptions will not present an undue risk to public health and safety

The underlying purpose of 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E is to ensure that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, to establish plume exposure and ingestion pathway emergency planning zones for nuclear power plants, and to ensure that licensees maintain effective offsite and onsite emergency plans.

The requested exemptions, and justification for each presented herein, are based on and consistent with Interim Staff Guidance NSIR/DPR-ISG-02, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," issued May 11, 2015 (Reference 1).

As discussed in this request, analyses were performed indicating that within 90 days of the permanent cessation of power operations of the Palisades Nuclear Plant (PNP), the radiological consequences of the remaining design basis accidents cannot exceed small fractions of the Environmental Protection Agency (EPA) Protective Action Guides (PAGs) at the exclusion area boundary (EAB). In addition, an analysis has been performed, which demonstrates that approximately 12 months after permanent cessation of power operations of the PNP reactor, a minimum of 10 hours is available before fuel cladding temperature of the hottest fuel assembly in the spent fuel pool (SFP) reaches 900°C with a complete loss of SFP water inventory, assuming no heat loss (adiabatic heatup). The analysis is provided in Attachments 1 and 2 to this Enclosure. After the approximately 12-month decay period, there is sufficient time within the 10 hours described in the supporting analysis to mitigate events at PNP that could lead to a zirconium cladding fire.

Additionally, the offsite and Control Room radiological impacts of a postulated complete loss of SFP water were assessed. It was determined that the gamma radiation dose rate at the EAB would be limited to small fractions of the EPA PAG exposure levels and

the limiting dose rate in the PNP Control Room is below 2.5 mrem/hour at one year after shutdown (Reference 21).

Therefore, offsite emergency response plans will no longer be needed for protection of the public beyond the EAB at approximately 12 months after permanent cessation of power operations of the PNP reactor. Based on the reduced consequences of radiological events possible at PNP when it is in the permanently shutdown and defueled condition, the scope of the onsite emergency preparedness organization and corresponding requirements in the emergency plan may be reduced without an undue risk to the public health and safety.

Therefore, the underlying purpose of the regulations will continue to be met. Because the underlying purpose of the rules will continue to be met, the exemptions will not present an undue risk to the public health and safety.

C. The exemptions are consistent with the common defense and security

The reduced consequences of radiological events that will remain possible at PNP when it is in the permanently defueled condition allows for a corresponding reduction in the scope of the onsite emergency preparedness organization and associated reduction of requirements in the emergency plan. These reductions will not adversely affect HDI's ability to physically secure the site or protect special nuclear material. Physical security measures at PNP are not affected by the requested exemptions. Therefore, the proposed exemptions are consistent with the common defense and security.

7.2 Special Circumstances

In accordance with 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to its regulations unless special circumstances are present. HDI has determined that special circumstances are present as discussed below.

Special circumstances will exist at PNP because the facility will be permanently shut down and defueled and the radiological source term at the site will be reduced from that associated with reactor power operation. With the reactor permanently shut down and defueled, the design basis accidents and transients postulated to occur during reactor operation will no longer be possible. In particular, the potential for a release of a large radiological source term to the environment from the high pressures and temperatures associated with reactor operation will no longer exist.

A. Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. (10 CFR 50.12(a)(2)(ii))

The underlying purpose of 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E is to ensure that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency, to establish plume exposure and ingestion pathway emergency planning zones for nuclear power plants, and to ensure that licensees maintain effective offsite and onsite emergency plans.

The standards and requirements in 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E were developed taking into consideration the risks associated with operation of a nuclear power reactor at its licensed full power level. These risks include the potential for a reactor accident with offsite radiological dose consequences.

The radiological consequences of accidents that will remain possible at PNP upon permanent shutdown are substantially lower than those at an operating plant. The upper bound of offsite dose consequences limits the highest attainable emergency class to the Alert level. In addition, because of the reduced consequences of radiological events that will still be possible at the site, the scope of the onsite emergency preparedness organization may be reduced accordingly. Thus, the underlying purpose of the regulations will not be adversely affected by eliminating offsite emergency planning activities or reducing the scope of onsite emergency planning as described in this request.

Analyses were performed indicating that within 90 days of the permanent cessation of power operations of the PNP reactor, the radiological consequences of the remaining design basis accidents cannot exceed small fractions of the EPA PAGs at the EAB. In addition, an analysis has been performed, which demonstrates that at approximately 12 months after permanent cessation of power operations of the PNP reactor, a minimum of 10 hours is available before fuel cladding temperature of the hottest fuel assembly in the spent fuel pool reaches 900°C with a complete loss of spent fuel pool water inventory, assuming no heat loss (adiabatic heatup). The analysis is provided in Attachments 1 and 2 to this Enclosure. After the approximate 12-month decay period, there is sufficient time within the 10 hours described in the supporting analysis to mitigate events at PNP that could lead to a zirconium cladding fire.

Therefore, application of all the standards and requirements in 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR 50, Appendix E are not necessary to achieve the underlying purpose of those regulations. Because the underlying purpose of the regulations would continue to be achieved even with HDI being permitted to reduce the scope of emergency preparedness requirements consistent with placing PNP in the permanently shutdown and defueled condition, the special circumstances are present as defined in 10 CFR 50.12(a)(2)(ii).

B. Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated. (10 CFR 50.12(a)(2)(iii))

Application of all of the standards and requirements in 10 CFR 50.47(b), 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E is not needed for adequate emergency response capability and is excessive for a permanently shut down and defueled facility. Application of all of these standards and requirements would result in undue costs being incurred for the maintenance of an emergency response organization (ERO) in excess of that actually needed to respond to the diminished scope of credible events. Other licensed sites similarly situated, such as NextEra Energy's (NextEra) Duane Arnold Energy Center (DAEC), Exelon Generation’s (Exelon) Three Mile Island Nuclear Station, Unit 1 (TMI-1), and Pilgrim Nuclear Station (Pilgrim), have been granted similar exemptions.

Full compliance with the rule would result in an undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated. Therefore, the special circumstances required by 10 CFR 50.12(a)(2)(iii) exist.

C. The exemptions would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemptions. (10 CFR 50.12(a)(2)(iv))

The plant will be permanently shut down and defueled and the radiological source term at the site will be reduced from that associated with reactor power operation. With the reactor permanently shut down and defueled, the design basis accidents and transients postulated to occur during reactor operation will no longer be possible. In particular, the potential for a release of a large radiological source term to the environment from the high pressures and temperatures associated with reactor operation will no longer exist.

The proposed exemptions would allow revisions to the Post-Shutdown Emergency Plan (PSEP) to correspond to the reduced scope of remaining accidents and events. As such, the emergency plan would no longer need to address response actions for events that would no longer be possible. The Permanently Defueled Emergency Plan (PDEP) would thereby enhance the ability of the ERO to respond to those scenarios that remain credible because emergency preparedness training and drills would focus only on applicable activities. Elimination of requirements for classification of emergency action levels for events that were no longer possible would enhance the ability of the ERO to correctly classify those events that remain credible. As the proposed exemptions will enhance the ability of the organization to respond to credible events, a resultant benefit to the public health and safety is realized.

Therefore, because granting the exemptions would result in benefit to the public health and safety and would not result in a decrease in safety, the special circumstances required by 10 CFR 50.12(a)(2)(iv) exist.

8.0 PRECEDENT

The exemption requests from the 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E requirements proposed by HDI for PNP in this exemption request are consistent with exemptions from the same emergency planning requirements that recently have been issued by the NRC for other nuclear power reactor facilities beginning decommissioning. Specifically, the NRC granted similar exemptions to NextEra for DAEC (Reference 26), Exelon for TMI-1 (Reference 27), and Holtec Decommissioning International, LLC (HDI) for Pilgrim (Reference 28). Similar to the current request, these precedents each resulted in exemptions from certain emergency planning requirements in 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E, related to the elimination of offsite radiological emergency plans and a reduction in the scope of the onsite emergency planning activities. For the same reasons that the NRC recently issued these exemptions, HDI seeks approval of the exemptions proposed in this exemption request.

9.0 ENVIRONMENTAL CONSIDERATIONS

The proposed exemptions meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, *Criterion for categorical exclusion; identification of licensing and regulatory*

actions eligible for categorical exclusion or otherwise not requiring environmental review, paragraph (c)(25), because the proposed exemptions involve: (i) no significant hazards consideration; (ii) no significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (iii) no significant increase in individual or cumulative public or occupational radiation exposure; (iv) no significant construction impact; (v) no significant increase in the potential for or consequences from radiological accidents; and (vi) requirements of an administrative, managerial, or organizational nature. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed exemptions.

(i) No Significant Hazards Consideration Determination

The requested exemptions from portions of Title 10 of the Code of Federal Regulations (10 CFR) 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E would allow Holtec Decommissioning International, LLC (HDI) to revise the scope of the Palisades Nuclear Plant (PNP) emergency plan to reflect the permanently shut down and defueled condition of the station. HDI has evaluated the proposed exemptions to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, *Issuance of amendment*, as discussed below:

1. <u>Does the proposed exemption involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   The proposed exemptions have no effect on structures, systems, and components (SSCs) and no effect on the capability of any facility SSC to perform its design function. The proposed exemptions would not increase the likelihood of the malfunction of any facility SSC.

   When the exemptions become effective, there will be no credible events that would result in doses to the public beyond the Exclusion Area Boundary (EAB) that would exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). The probability of occurrence of previously evaluated accidents is not increased, because most previously analyzed accidents are no longer possible and the probability and consequences of the remaining postulated accidents are unaffected by the proposed exemptions.

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

2. <u>Do the proposed exemptions create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

   The proposed exemptions do not involve a physical alteration of the facility. No new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed exemptions. Similarly, the proposed exemptions will not physically change any SSCs involved in the mitigation of any accidents. Thus, no new initiators or precursors of a new or different kind of accident are created. Furthermore, the proposed exemptions do not create the possibility of a new accident as a result of new failure modes associated with any equipment or personnel failures. No changes are being made to parameters within which the facility is normally operated, or in the

setpoints which initiate protective or mitigative actions, and no new failure modes are being introduced.

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

3. <u>Do the proposed exemptions involve a significant reduction in a margin of safety?</u>

The proposed exemptions do not alter the design basis or any safety limits for the facility. The proposed exemptions do not impact facility operation or any facility SSC that is relied upon for accident mitigation.

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

Based on the above, HDI concludes that the proposed exemptions present no significant hazards consideration, and, accordingly, a finding of "no significant hazards consideration" is justified.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

There are no expected changes in the types, characteristics, or quantities of effluents discharged to the environment associated with the proposed exemptions. There are no materials or chemicals introduced into the facility that could affect the characteristics or types of effluents released offsite. In addition, the method of operation of waste processing systems will not be affected by the exemptions. The proposed exemptions will not result in changes to the design basis requirements of SSCs that function to limit or monitor the release of effluents. The SSCs associated with limiting the release of effluents will continue to be able to perform their functions. Therefore, the proposed exemptions will result in no significant change to the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative public or occupational radiation exposure.

The exemptions will result in no expected increases in individual or cumulative occupational radiation exposure on either the workforce or the public. There are no expected changes in normal occupational doses. Likewise, the dose consequences of the postulated accidents are not impacted by the proposed exemptions.

(iv) There is no significant construction impact.

No construction activities are associated with the proposed exemptions.

(v) There is no significant increase in the potential for or consequences from radiological accidents.

See the no significant hazards considerations discussion in Item (i)(1) above.

(vi) Requirements of an administrative, managerial, or organizational nature.

The proposed exemptions will form the basis for a reduction in the size of the PNP emergency response organization (ERO) commensurate with the reduction in consequences of radiological events that will be possible with the facility in the permanently shutdown and defueled condition. The proposed exemption also modify the requirements for emergency planning and the ERO. Therefore, the exemptions address requirements of an administrative, managerial, or organizational nature.

10.0 REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), NSIR/DPR-ISG-02, Interim Staff Guidance, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," dated May 11, 2015 (ADAMS Accession No. ML14302A490)

2. Entergy Nuclear Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel," dated June 13, 2022 (ADAMS Accession No. ML22164A067)

3. U.S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Nuclear Plant – Issuance of Amendment Re: Changes to the Emergency Plan for Permanently Defueled Condition (CAC No. MG0198; EPID L-2017-LLA-0305)," dated September 24, 2018 (ADAMS Accession No. ML18170A219)

4. U.S. Environmental Protection Agency, "Protective Action Guides and Planning Guidance for Radiological Incidents," EPA-400/R-17-001 (EPA PAG Manual), dated January 2017

5. Federal Register Notice, Vol. 60, No. 120 (60 FR 32430-32442), "Emergency Planning Licensing Requirements for Independent Spent Fuel Storage Facilities (ISFSI) and Monitored Retrievable Storage Facilities (MRS)," dated June 22, 1995

6. NRC, Commission Paper SECY-00-0145, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning," dated June 28, 2000 (ADAMS Accession No. ML003721626)

7. Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML12326A805)

8. NRC, Commission Paper SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated October 9, 2013 (ADAMS Accession No. ML 13256A339)

9. Federal Register Notice, Vol. 76, No. 226 (76 FR 72560 - 72598), "Enhancements to Emergency Preparedness Regulations," dated November 23, 2011 (ADAMS Accession No. ML13091A112)

10. NRC, Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005 (ADAMS Accession No. ML051740058)

11. NRC letter, Mark Thaggard to Susan Perkins-Grew (NEI), "U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01," Revision 6, dated November 2012 (TAC No. D92368), dated March 28, 2013 (ADAMS Accession No. ML12346A463)

12. NRC, NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001 (ADAMS Accession No. ML010430066)

13. Federal Register Notice, Vol. 74, No. 94 (74 FR 23254 - 23286), "Enhancements to Emergency Preparedness Regulations," dated May 18, 2009

14. NRC, NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981 (ADAMS Accession No. ML051390358)

15. U.S. Nuclear Regulatory Commission letter to Entergy Nuclear Operations, Inc., "Palisades Nuclear Plant – Issuance of Amendment No. 272 Re: Permanently Defueled Technical Specifications (EPID L-2021-LLA-0099)," dated May 13, 2022 (ADAMS Accession No. ML22039A198)

16. Entergy Nuclear Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "License Amendment Request to Revise Renewed Facility Operating License and Technical Specifications for Permanently Defueled Condition," dated June 1, 2021 (ADAMS Accession No. ML21152A109)

17. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 31, 2000 (ADAMS Accession No. ML003716792)

18. EC-92748, Clarify Implications of PDTS FHA Analysis EA-EC89582-01 for PDEP

19. NRC, Commission Paper SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants," dated June 30, 1999 (ADAMS Accession No. ML992800087)

20. NRC, NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated October 2002

21. EC-92900, Clarify Applicability of EA-EC72870-01 to May 2022 Shutdown

22. NRC, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (ADAMS Accession No. ML100910006)

23. NRC, NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," dated September 2014 (ADAMS Accession No. ML14255A365)

24. NRC, NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated January 1980 (ADAMS Accession No. ML070205180)

25. NRC, NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” dated May 1979 ADAMS Accession No. ML110450636)

26. NRC letter to Florida Power & Light Company, “Duane Arnold Energy Center – Exemptions from Certain Emergency Planning Requirements and Related Safety Evaluation (EPID L-2020-LLE-0023),” dated April 13, 2021 (ADAMS Accession No. ML21097A141)

27. NRC letter to Exelon Generation Company, LLC, “Three Mile Island Station, Units 1 and 2 – Exemptions from Certain Emergency Planning Requirements and Related Safety Evaluation (EPID L-2019-LLE-0016),” dated December 1, 2020 (ADAMS Accession No. ML20244A292)

28. NRC letter to Holtec Decommissioning International, LLC (HDI), "Pilgrim Nuclear Power Station – Exemptions from Certain Emergency Planning Requirements and Related Safety Evaluation (EPID L-2018-LLE-0011)," dated December 18, 2019 (ADAMS Accession No. ML19142A043)

Attachment 1 to Enclosure

HDI PNP 2022-017

Holtec Spent Fuel Pool Calculations (Holtec Proprietary, Withhold Information from Public Disclosure pursuant to 10 CFR 2.390)

Attachment 2 to Enclosure

HDI PNP 2022-017

Holtec Spent Fuel Pool Calculations (Non-Proprietary)

Nuclear Power Division		3132
Sponsoring Company		**Project No.**
HI-2220400	1	08 Jul 2022
Company Record Number	**Revision No.**	**Issue Date**
Report	~~Proprietary Information~~	
Record Type	**Proprietary Classification**	
Nuclear	No	
Quality Class	**Export Control Applicability**	

Record Title:

Spent Fuel Pool Heat Load Limits for Palisades

Prepared by:	**Reviewed by:**	**Approved by:**
J.Rocheleau, 06 Jul 2022	S.Anton, 06 Jul 2022	P.Gowda, 08 Jul 2022

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Proprietary Classification

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Export Control Status

Not applicable.

Revision Log

Revision	Description of Changes
0	Initial issue.
1	Change references to the RAIs on the Topical Report to the final SE for that report. Texts updated accordingly.

EXECUTIVE SUMMARY

This report documents the calculation of the Spent Fuel Pool (SFP) assembly heat load limits for Palisades Nuclear Plant. Heat load limits[ ]are developed to ensure the fuel cladding remains below the temperature limit that precludes a zirconium fire in the postulated event of an unexpected SFP drain. To the extent possible, the information in the Topical Report (TR), HI-2200750 Rev. 0, [1] and corresponding Safety Evaluation (SE) [6] are referenced and not repeated in this report. **4a,4b**

Assumptions that need Verification

There are no assumptions that need verification.

Glossary

PCT	Peak Cladding Temperature
SFP	Spent Fuel Pool
[	] **4a,4b**
TR	Topical Report
[	] **4a,4b**

Table of Contents

1.0 PURPOSE

The purpose of this report is to document the calculation of the Spent Fuel Pool (SFP) assembly heat load limits for Palisades Nuclear Plant. The Topical Report (TR) HI-2200750 Rev. 0 [1] and corresponding Safety Evaluation (SE) [6] define a methodology to establish[ ] 4a,4b criteria for a SFP that can demonstrate a zirconium fire in the pool under a drain-down event is not credible. To the extent possible, the information in the TR [1] and SE [6] are referenced and not repeated in this report.

2.0 ACCEPTANCE CRITERIA

There are no specific acceptance criteria for the calculations presented here.

However, the heat load criteria determined in the calculations presented here must satisfy the applicable criteria discussed in Section 2 of [1] that preclude a zirc fire in the SFP, namely that the fuel cladding must not exceed 900 °C within 10 hours.

3.0 METHODOLOGY

All calculations performed use the methodology described in Section 3 of [1], also considering the conditions in the SE [6]. Generally, these are[

] 4a,4b

4.0 ASSUMPTIONS

For a list of the main assumptions on the methodology, see Section 4 of [1].

No site-specific assumptions are made in this report.

5.0 INPUT DATA

All generic inputs are documented in Section 5 of [1], specifically Table 5.1 and Table 5.2. The site-specific inputs presented in the table below are taken from [2]. Separate calculations are performed for each respective batch type, rather than evaluating a single conservatively bounding batch type, due to the wide range of some inputs. Parameters marked with an asterisk (*) are calculated from the values in the table, Table 5.2 of [1], and [2].

4a,4b

4a,4b

6.0 COMPUTER PROGRAMS

All calculations were performed using internally developed scripts.

7.0 COMPUTER FILES

All computer files that are used to perform the calculations are located on the Holtec International server under the project 3132 directory in the report HI-2220400 directory.

The md5 checksums of significant calculation files are provided below:

4a,4b

File name	Md5

[ ]

8.0 CALCULATIONS

The methodology presented in [1] was implemented in a straightforward manner into a computer script. [ ] 4,a,4b

9.0 RESULTS

4a,4b The results are summarized in the table below.

[ ]

4,a,4b

10.0 CONCLUSION

This report documents calculations and corresponding results. The SFP Heat Load Limits established in this report can be us [

] 4a,4b

11.0 REFERENCES

[1] "Method for Determining Spent Fuel Assembly Heat Up During a Theoretical Drain Down Event", HI-2200750 Rev. 0.

[2] “Design Input Record – Holtec Information Sharing”, Email form Gretchen Goddard (Entergy) to Joshua Rocheleau (Holtec) dated 04-05-2022.

[3] “Draft Limitations and Conditions for the Holtec Decommissioning International Topical Report HI-2200750, Revision 0”, EPID L-2020-TOP-0056. September 28, 2021.

[4] "NUREG-1738 Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (ML010430066)," 2001.

[5] "NUREG/CR-0479 MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," 1981.

[6] “Final Safety Evalulation by the Office of Nuclear Reactor Regulation for the Topical Report HI-2200750, REVISION 0”, DOCKET: 99902086 EPID: L-2020-TOP-0056.

Nuclear Power Division		3132
Sponsoring Company		**Project No.**
HI-2220348	0	01 Jul 2022
Company Record Number	**Revision No.**	**Issue Date**
Report	~~Proprietary Information~~	
Record Type	**Proprietary Classification**	
None	No	
Quality Class	**Export Control Applicability**	

Record Title:

Assembly Decay Heat Calculations for Palisades Power Plant

Prepared by:	**Reviewed by:**	**Approved by:**
J.Rocheleau, 29 Jun 2022	S.Anton, 29 Jun 2022	P.Gowda, 01 Jul 2022

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Export Control Status

Not applicable.

Revision Log

Revision	Description of Changes
0	Initial issue.

EXECUTIVE SUMMARY

This report documents the calculation of the assembly specific decay heats for Palisades Power Plant. [

] 4a,4b

Assumptions that need Verification

There are no assumptions that need verification.

Table of Contents

1.0 PURPOSE

The purpose of this report is to document the calculation of the assembly specific decay heats for Palisades Power Plant spent fuel inventory.[

] 4a,4b

2.0 ACCEPTANCE CRITERIA

There are no specific acceptance criteria for the calculations presented here.

3.0 METHODOLOGY

The decay heat values are calculated with the[

] 4a,4b

4.0 ASSUMPTIONS

Assumptions for this analysis are listed below.

[ ] 4a,4b

5.0 INPUT DATA

[

] 4a,4b

Table 5-1: Fuel Assembly Parameters for[ ] 4a,4b

[ ] 4a,4b

6.0 COMPUTER PROGRAMS

[ ] 4a,4b

7.0 COMPUTER FILES

4a,4b

8.0 CALCULATIONS

4a,4b

4a,4b

[

] 4a,4b

9.0 RESULTS

This report documents the generation of[

] 4a,4b

10.0 CONCLUSION

This report documents calculations that support subsequent analyses of[ ] 4a,4b
There are no specific conclusions.

11.0 REFERENCES

[1] "B.T. Rearden and M.A. Jessee, Eds., SCALE Code System, ORNL/TM-2005/39, Version 6.2.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2016). Available from Radiation Safety Information Computational Center as CCC-834

[2] Source Terms and Loading Patterns Using SCALE 6.2, HI-2167524, Rev 4.

[3] Design Input Record, EN-DC-141, Document Revision 0, Prepared by Gretchen Goddard (from Palisades), Dated on 04/05/2022.

[4] Email from Dan Clemens (from Palisades) to Joshua Rocheleau (from Holtec), with the Subject of "Holtec SFP Defuel Inputs", Dated on 04/12/2022.

[5] Email from Guy Wiggins (From Palisades) to Joshua Rocheleau (from Holtec), with the Subject of "Palisades SFP LP - 6/01/2022", Dated on 06/14/2022.

Nuclear Power Division	3132
Sponsoring Company	**Project No.**

HI-2220401	0	01 Jul 2022
Company Record Number	**Revision No.**	**Issue Date**

Report	~~Proprietary Information~~
Record Type	**Proprietary Classification**
Nuclear	No
Quality Class	**Export Control Applicability**

Record Title:

Spent Fuel Pool Loading Plan for Palisades

Prepared by:	**Reviewed by:**	**Approved by:**
J.Rocheleau, 29 Jun 2022	S.Anton, 29 Jun 2022	P.Gowda, 01 Jul 2022

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Export Control Status

Not applicable.

Revision Log

Revision	Description of Changes
0	Initial issue.

EXECUTIVE SUMMARY

This report documents the [4a, 4b]Spent Fuel Pool of Palisades Power Plant.[

4a, 4b

]

Assumptions that need Verification

There are no assumptions that need verification.

Glossary

SFP	Spent Fuel Pool
[	] 4a, 4b
HLL	Heat Load Limit
[	] 4a, 4b

Table of Contents

1.0 PURPOSE

The purpose of this report is to document the development of[

] 4a, 4b

2.0 ACCEPTANCE CRITERIA

All assemblies in the SFP are[

] 4a, 4b

3.0 METHODOLOGY

Two sets of calculations are performed to[
] 4a, 4b

First,[

4a, 4b
]to meet applicable zirc fire safety criteria.

Second, a computer code is used with[

] 4a, 4b

[

] 4a, 4b

4.0 ASSUMPTIONS

Assumptions for this analysis are listed below.

4a, 4b

5.0 INPUT DATA

4a, 4b

6.0 COMPUTER PROGRAMS

4a, 4b

7.0 COMPUTER FILES

All computer files that are used to generate the source terms are located on the Holtec International server under the project 3131 directory in the report HI-2220401 directory.

4a, 4b

8.0 CALCULATIONS

Two sets of calculations were performed. First to determine the**[** 4a,4b
]and second**[**
] 4a,4b

8.1 UAL Threshold

4a, 4b

8.2 SFP Loading Configuration

4a, 4b

[

] 4a, 4b

9.0 CONCLUSION

[

] 4a, 4b

10.0 REFERENCES

[1] "Spent Fuel Pool Heat Load Limits for Palisades", HI-2220400, Rev. 0.
[2] "Assembly Decay Heat Calculations for Palisades", HI-2220348, Rev. 0.
[3] "Design Input Record – Holtec Information Sharing", Email form Gretchen Goddard (Entergy) to Joshua Rocheleau (Holtec) dated 04-05-2022.
[4] "1D28 SFP Offload Map-GTW 5.17.22.xls", Email from Guy Wiggins (Entergy) to Joshua Rocheleau (Holtec) data 05-17-2022.
[5] "NUREG-6801 Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," ML031110292, 2003.
[6] "Method for Determining Spent Fuel Assembly Heat Up During a Theoretical Drain Down Event", HI-2200750 Rev. 0.

.

Appendix A: SFP LOADING CONFIGURATION

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

4a, 4b

Attachment 3 to Enclosure

HDI PNP 2022-017

Affidavit Pursuant to 10 CFR 2.390 to Withhold Information from Public Disclosure

U.S. Nuclear Regulatory Commission
ATTN: Document Control Clerk
Non-Proprietary Attachment 3 to HDI PNP 2022-017

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Jean A. Fleming, being duly sworn, depose and state as follows:

1) I have reviewed the information described in paragraph (2) which is sought to be withheld and am authorized to apply for its withholding.

2) The information sought to be withheld is in Attachment 1 to the Enclosure to Holtec Letter HDI PNP 2022-017. These documents information that is proprietary to Holtec International.

3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

4) Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

   b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b above.

5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or

AFFIDAVIT PURSUANT TO 10 CFR 2.390

other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive decommissioning and spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

   The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

   The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF CAMDEN)

Jean A. Fleming, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Camden, New Jersey, this 11th day of July 2022.

Jean Fleming

Jean A. Fleming
VP - Licensing, Regulatory Affairs & PSA
Holtec International

Subscribed and sworn before me this 11th day of July, 2022

Sharon Y. Page

SHARON Y. PAGE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 01/15/2026

Attachment 4 to Enclosure

HDI PNP 2022-017

Palisades Spent Fuel Pool HCLPF Evaluation

☐ ANO-1 ☐ ANO-2 ☐ GGNS ☐ IP-2 ☐ IP-3 ☒ PLP

☐ BRP ☐ RBS ☐ W3 ☐ NP-GGNS-3 ☐ NP-RBS-3

CALCULATION COVER PAGE	(1) EC #: 93028	(2) Page 1 of 27
(3) Design Basis Calc. ☐ YES ☒ NO	(4) ☒ CALCULATION ☐ EC Markup	
(5) Calculation No.: 32-9346162-000		(6) Revision: 0
(7) Title: Palisades Spent Fuel Pool HCLPF Evaluation		(8) Editorial ☐ YES ☒ NO
(9) System(s): SFP	(10) Review Org (Department): Eng	
(11) Safety Class: ☐ Safety / Quality Related ☐ Augmented Quality Program ☒ Non-Safety Related	(12) Component/Equipment/Structure Type/Number	
(13) Document Type: ENG01		
(14) Keywords (Description/Topical Codes) SFP		

REVIEWS

(15) Name/Signature/Date	(16) Name/Signature/Date	(17) Name/Signature/Date
Prepared, reviewed, and approved by Framatome (see page 5)	Ted Diamond Theodore Diamond 5/17/2022	Bob White Robert A White 5-18-22
Responsible Engineer	☐ **Design Verifier** ☐ **Reviewer** ☒ **Owner's Acceptance Review** ☐ Comments Attached	**Supervisor/Approval** ☐ Comments Attached

CALCULATION REFERENCE SHEET	CALCULATION NO: 32-9346162-000 REVISION: 0

I. EC Markups Incorporated (N/A to NP calculations)
1. None – new calc
2.
3.
4.
5.

II. Relationships:	Sht	Rev	Input Doc	Output Doc	Impact Y/N	Tracking No.
1. See page 27 for references			☐	☐		
2.			☐	☐		
3.			☐	☐		
4.			☐	☐		
5.			☐	☐		

III. CROSS REFERENCES:
1.
2.
3.
4.

IV. SOFTWARE USED:
Title: _____ Version/Release: _____ Disk/CD No.: _____

V. DISK/CDS INCLUDED:
Title: _____ Version/Release: _____ Disk/CD No.: _____

VI. OTHER CHANGES:

Revision	Record of Revision
0	Initial issue.

framatome

CALCULATION SUMMARY SHEET (CSS)

Document No. 32 - 9346162 - 000

Safety Related: ☐Yes ☒ No

Title Palisades Spent Fuel Pool HCLPF Evaluation

PURPOSE AND SUMMARY OF RESULTS:

Purpose:
The purpose of this calculation is to evaluate the controlling structural fragility developed for the Auxiliary Building in the previous SPRA (References [4] and [5]). The focus of the evaluation is the seismic risk contribution from the Spent Fuel Pool (SFP) and tilt pits. Thereafter, a site-specific seismic hazard with an AEF of 10^{-5} per year for the Auxiliary Building and SFP structures is assessed. The existing response analysis of the Auxiliary Building considers two smaller tilt pits to the west of the SFP empty (no water). However, in the long-term post-shutdown condition, the north tilt pit will, and the south tilt pit, may be filled with water. Additionally, it is understood that the south tilt pit filled condition is a temporary configuration.

Summary:
It is concluded that the filling with cooling water of the north tilt pit and/or the south tilt pit does not affect the HCLPF capacity of the SFP. The HCLPF capacity of the SFP is determined to be governed by the Auxiliary Building. This calculation also evaluates the impact on the HCLPF capacity if the 10^{-5} AEF seismic hazard level was used to determine the structural fragility as opposed to the GMRS hazard level. The determination is based on the evaluation of a variety of parameters that have significant effects on the Auxiliary Building capacity. It is concluded that there is sufficient margin that the Auxiliary Building HCLPF capacity of 0.71g is also valid if the capacity evaluation was based on a 10^{-5} AEF seismic hazard. Therefore, the SFP HCLPF capacity is larger than the $\mathrm{PGA}_{10^{-5}}$ of 0.569g.

If the computer software used herein is not the latest version per the EASI list, AP 0402-01 requires that justification be provided.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV
N/A	

THE DOCUMENT CONTAINS ASSUMPTIONS THAT SHALL BE VERIFIED PRIOR TO USE

☐Yes
☒ No

framatome

0402-01-F01 (Rev. 021, 03/12/2018)
Document No. 32-9346162-000

Palisades Spent Fuel Pool HCLPF Evaluation

Review Method: ☒ Design Review (Detailed Check)

☐ Alternate Calculation

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title (printed or typed)	Signature	P/R/A/M and LP/LR	Date	Pages/Sections Prepared/Reviewed/Approved
Tobias Richter Principal Engineer	*T RICHTER* *5/4/2022*	LP		All
Mark Stewart Advisory Engineer	*MJ STEWART* *5/4/2022*	R		Section 5.0 and 6.0
Chris McGaughy Advisory Engineer	*CJ MCGAUGHY* *5/4/2022*	LR		All
Tim Schmitt Engineering Supervisor	*TA SCHMITT* *5/4/2022*	A		All

Notes: P/R/A designates Preparer (P), Reviewer (R), Approver (A);
LP/LR designates Lead Preparer (LP), Lead Reviewer (LR);
M designates Mentor (M)

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Name (printed or typed)	Title (printed or typed)	Signature	Date	Comments
Kim Jones	Project Manager	*KM JONES* *5/4/2022*		

Record of Revision

Revision No.	Pages/Sections/Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial issue

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1.0 INTRODUCTION

Palisades Nuclear Plant (PLP) plans to submit a request for exemption from the Emergency Preparedness (EP) requirements for nuclear power reactors that have been permanently shut down and defueled and are planning to transition to a decommissioning state. NSIR/DPR-ISG-02 (Reference [1]) provides interim staff guidance to Nuclear Regulatory Commission (NRC) staff for conducting technical review of these exemption requests. Section 5.0 of NSIR/DPR-ISG-02 defines the expected analyses to be performed by licensees as part of an EP requirement exemption request. Item 4 of this section is an evaluation of the spent fuel pool storage (SFP) structure which demonstrates a High Confidence of Low Probability of Failure (HCLPF) considering a site-specific seismic hazard with an Annual Exceedance Frequency (AEF) less than 10^{-5} per year.

Framatome previously developed structural models, seismic response analysis, and preliminary seismic fragility calculations as part of a Seismic Probabilistic Risk Assessment (SPRA) initiated for Palisades' response to the NRC request for information regarding recommendation 2.1 "Seismic" of the Near-Term Task Force (NTTF) Review of Insights from the Fukushima-Daiichi Accident. The submittal of this SPRA to the NRC was deferred to a date after the planned permanent shutdown date for Palisades.

The purpose of this calculation is to evaluate the controlling structural fragility developed for the Auxiliary Building in the previous SPRA (References [4] and [5]). The focus of the evaluation is the seismic risk contribution from the Spent Fuel Pool (SFP) and tilt pits. Thereafter, a site-specific seismic hazard with an AEF of 10^{-5} per year for the Auxiliary Building and SFP structures is assessed. The existing response analysis of the Auxiliary Building considers two smaller tilt pits to the west of the SFP empty (no water). However, in the long-term post-shutdown condition, the north tilt pit will, and the south tilt pit, may be filled with water. Additionally, it is understood that the south tilt pit filled condition is a temporary configuration.

2.0 ANALYTICAL METHODOLOGY

The calculation methodology is divided into three steps:

1. Evaluate the local effects of the filled tilt pits on the SFP fragility. See Section 4.1.
2. Address the global effects of the filled tilt pits on the SFP fragility. See Section 4.2.
3. Assess the impact on the Auxiliary Building, which controls the SFP fragility, if a higher seismic hazard, AEF of 10^{-5} per year is used. See Section 4.3. Thereby, soil properties, horizontal and vertical spectral shapes, vertical/horizontal ratio, and concrete cracking impacts on the fragility are investigated.

Section 5.0 summarizes the analysis results where the acceptance criterion is a SFP HCLPF larger than 0.569g. Section 6.0 provides the conclusion.

3.0 ASSUMPTIONS

3.1 Unverified Assumptions

None.

3.2 Justified Assumptions

None.

3.3 Modeling Simplifications

None.

Document No. 32-9346162-000

4.0 CALCULATIONS

The structural response analysis model for probabilistic risk assessment in Reference [4] utilizes Foundation Input Response Spectra (FIRS) at elevation 575 ft as input. The FIRS are hazard-consistent with the Ground Motion Response Spectra (GMRS) at elevation 589 ft. A Reference Earthquake (RE) at the GMRS hazard level was selected for the analysis. The Peak Ground Acceleration (PGA) of the RE is PGA_{RE} = 0.278g and resulted in a HCLPF capacity of the Auxiliary Building of $HCLPF_{AB}$ = 0.71g. The RE PGA is in between the Uniform Hazard Response Spectra (UHRS) PGA of $PGA_{10^{-4}}$ = 0.202g at an AEF of 10^{-4} and $PGA_{10^{-5}}$ = 0.569g at an AEF of 10^{-5}.

The SFP is a structure that is filled with cooling water and is important to safety. Despite that, the walls SFP1 through SFP9 (Figure 4-1) were not evaluated in detail against concrete cracking in Reference [4]. This is because the SFP walls are generally rugged and the wall capacity against concrete cracking is large. With the SFP structure remaining uncracked at a PGA of 0.71g the associated loss of coolant is not likely and the SFP can perform its safety function. Therefore, even if SFP concrete cracking and potential loss of inventory would occur before local SFP wall failure, the Auxiliary Building fragility, as determined by a shear failure of the exterior wall, controls the fragility of the SFP.

4.1 Filled Tilt Pit, Local Response Effects

The SFP walls are labeled SFP1 through SFP9 in Reference [4] (Figure 4-1). SFP coolant water and its dynamic effects were modeled in the SFP only where the two tilt pits were left empty. However, all the SFP walls (SFP1 through SFP9) screened out of the concrete cracking assessment because the walls and floor were deemed rugged. If the tilt pits are filled with coolant water walls SFP5, SFP6, and SFP7 would be the walls most susceptible to concrete cracking and failure. A general strength assessment through wall dimensions excludes SFP4 and SFP8 because the 4 ft thick wall SFP4 screened out in [4] and governs the 6.5 ft thick wall SFP8. SFP5 with a thickness of t_5 = 2 ft has the smallest thickness compared to SFP6 and SFP7 with a thickness of 4 ft each while the shortest span of 4 / 2 + 5 + 6.5 / 2 = 10.25 ft is the same for all walls (References [7], [8], and [9]). Therefore, SFP5 at the North tilt pit is representatively evaluated in greater detail.

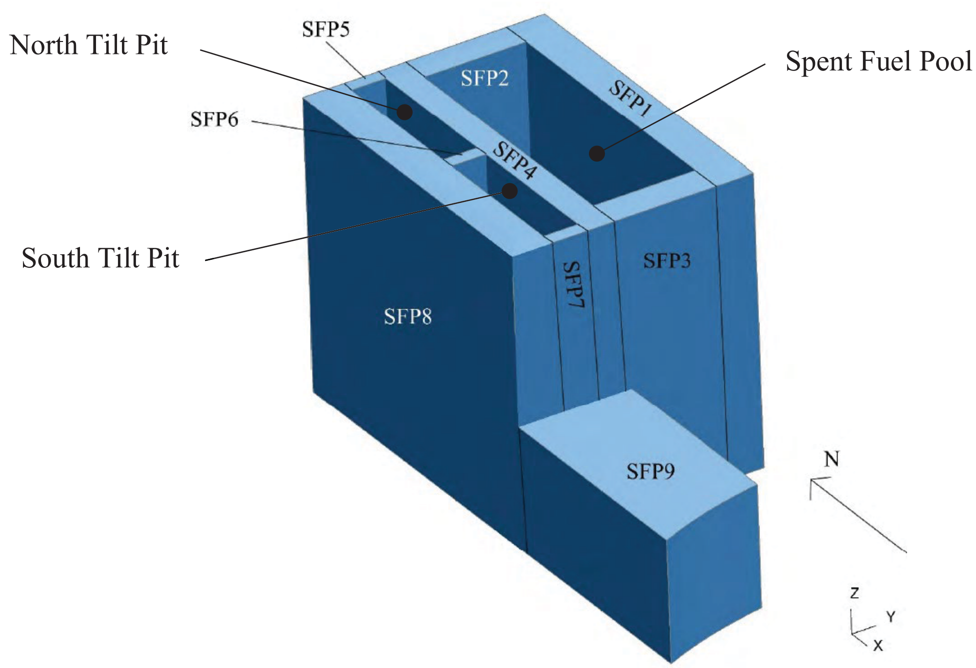

Figure 4-1: Spent Fuel Pool Walls, Isometric View

North tilt pit dimension:

Tilt pit length:	$2L$	$= 21$ ft,	Reference [7]
Tilt pit width:	$2B$	$= 5$ ft,	Reference [7]
SFP liquid depth:	h_{SFP}	$= 41.8$ ft,	Reference [4] (SFP bottom el. 607.5 ft)
Tilt pit liquid depth:	h	$= h_{SFP} - (610 - 607.5) = 39.3$ ft,	Reference [6] (Tilt pit bottom el. 610 ft)

The dynamic fluid pressures on wall SFP5 are evaluated utilizing TID-7024 (Reference [14]). The tilt pit is characterized as a slender rectangular tank because $h / L = 41.8 / (21/2) = 4.0 > 1.5$. Two design cases result for slender tanks:

(1) For the evaluation of impulsive forces, the tilt pit is regarded as having a fictitious bottom at a datum of $h_{top} = 1.5L = 15.75$ ft. Above this datum convective and impulsive forces apply.

(2) Below the datum the water mass responds as a rigid body generating only an impulsive force.

<u>Case (1):</u>

Gravity:	g	$= 32.2 \text{ ft} / \text{s}^2$	
Gravitational constant:	g_c	$= 32.2 \text{ lbm-ft} / \text{lbf-s}^2$	
Convective eigenfrequency:	f_c	$= 1 / (2\pi)\cdot(1.58g / L \tanh(1.58\, h_{top} / L))^{0.5} = 0.35$ Hz	
Coolant water density:	ρ_w	$= 62.4$ lbm/ft,	Reference [4]
Total water mass:	M	$= \rho_w\cdot 2L\cdot 2B\cdot h = 257494$ lbm	
Top water mass:	M_{top}	$= \rho_w\cdot 2L\cdot 2B\cdot h_{top} = 103194$ lbm	
Bottom water mass:	M_{bt}	$= \rho_w\cdot 2L\cdot 2B\cdot (h - h_{top}) = 154300$ lbm	
Top impulsive mass:	$M_{0,top}$	$= M_{top} \tanh(3^{0.5}\, L / h_{top}) / (3^{0.5}\, L / h_{top}) = 103194\cdot 0.71 = 73220$ lbm	
Top convective mass:	$M_{1,top}$	$= M_{top}\cdot 0.527\cdot L / h_{top} \tanh(1.58\, h_{top} / L) = 103194\cdot 0.35 = 35627$ lbm	
Impulsive mode damping:	d_i	$= 3\%$,	Reference [15]
Convective mode damping:	d_c	$= 0.5\%$,	Reference [15]
Concrete damping:	d	$= 7\%$,	cracked concrete properties are assumed for wall evaluation per Reference [13]

The seismic accelerations for design case (1) are conservatively taken from the Auxiliary Building elevation 649 ft. The maximum Zero Period Acceleration (ZPA) from response location AB-649-422 and AB-649-808 near the SFP top in X-direction is used:

$SA_{ZPA,top} = \max(0.350; 0.345) = 0.35$ g, Reference [4]

Spectral acceleration at f_c: $SA_{fc,3\%} = \max([0.038 + 0.055] / 2; [0.038 + 0.056] / 2) = 0.047$ g, Reference [4]

The seismic accelerations for design case (2) are taken from the Auxiliary Building elevation 607 ft. The ZPA for response location AB-607-220 near the SFP bottom in X-direction is used:

$SA_{ZPA,bt} = 0.187$ g, Reference [4]

Reference [4] does not contain response spectra at 0.5% damping. However, the GIP (Reference [16]) provides a method to estimate the response if larger than the ZPA:

$SA_{fc,0.5\%} = \max(SA_{fc,3\%}; SA_{ZPA,top})\cdot(3\% / 0.5\%)^{0.5} = 0.86$ g

The scale factor used to determine the HCLPF capacity of the Auxiliary Building is:

$SF = HCLPF_{AB} / PGA_{RE} = 2.55$

Horizontal design accelerations:

$a_{h,i}$ $= SF \cdot SA_{ZPA,top} = 0.89$ g

$a_{h,c}$ $= SF \cdot SA_{fc,0.5\%} = 2.19$ g

$a_{h,bt}$ $= SF \cdot SA_{ZPA,bt} = 0.48$ g

The reinforced concrete wall is modeled as a b = 12" wide strip of an elastic straight beam with both ends fixed (Figure 4-2).

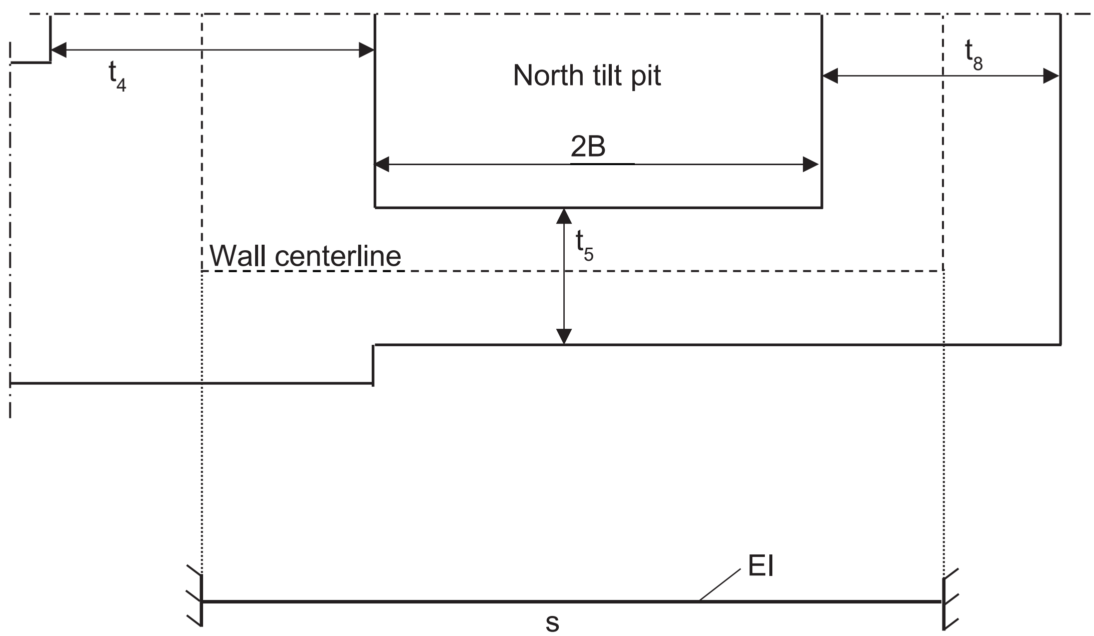

Figure 4-2: Analysis Model

Thickness wall SFP4: t_4 = 4 ft, Reference [7]

Thickness wall SFP8: t_8 = 6.5 ft, Reference [8]

Design beam span: s $= t_4 / 2 + 2B + t_8 / 2 = 10.25$ ft

Design beam moment of inertia:

I $= b \cdot t_5^3 / 12 = 0.667\ ft^4$

Concrete density: ρ_c = 150 lbm/ft, Reference [4]

Concrete modulus of elasticity:

E = 449571 ksf, Reference [4]

Hydrostatic loads are conservatively applied as a unit load with maximal amplitude. Additionally, hydrodynamic loads (impulsive and convective) are applied as a point load. See Figure 4-3.

Hydrostatic load: q_{hs} $= q_{hs,top} = \rho_w \cdot g / g_c \cdot h_{top} \cdot b = 983$ lbf/ft

Hydrodynamic load: Q_{hd} $= Q_{0,top} + Q_{1,top} = (M_{0,top} \cdot a_{h,i} + M_{1,top} \cdot a_{h,c}) / (g_c \cdot h) = 3650$ lbf

Design moment: M_{top} $= q_{hs} \cdot s^2 / 24 + Q_{hd} \cdot s / 8 = 9.0$ kip-ft, Reference [17]

The wall SFP5 is reviewed for structural flexibility.

Concrete unit mass: m_c $= \rho_c \cdot t_5 \cdot b = 300$ lbm/ft

Concrete unit lateral load at the Auxiliary Building HCLPF capacity level:

$w_c \quad = m_c \cdot a_{h,bt} / g_c = 143$ lbf/ft

Water lateral load at the Auxiliary Building HCLPF capacity level:

$W_w \quad = M \cdot a_{h,bt} / g_c \cdot b / h = 3129$ lbf

The natural frequency of the wall SFP5 is estimated using the conservative beam span s (Reference [17], uniform beam, both ends fixed, case 2c). In reality, the effective span for structures with thick members in this arrangement is closer to the clearance between the walls (s = 5 ft):

$f \quad = 13.86 / (2\pi) \cdot (EI \cdot g / (W_w \cdot s^3 + 0.383 w_c \cdot s^4))^{0.5} = 109$ Hz

The calculated frequency for the wall SFP5 is well beyond the frequency of interest for the Auxiliary Building (approx. 30 Hz, Reference [4]). Therefore, the wall SFP5 can be considered rigid, and no additional spectral amplifications need to be considered in the design evaluation.

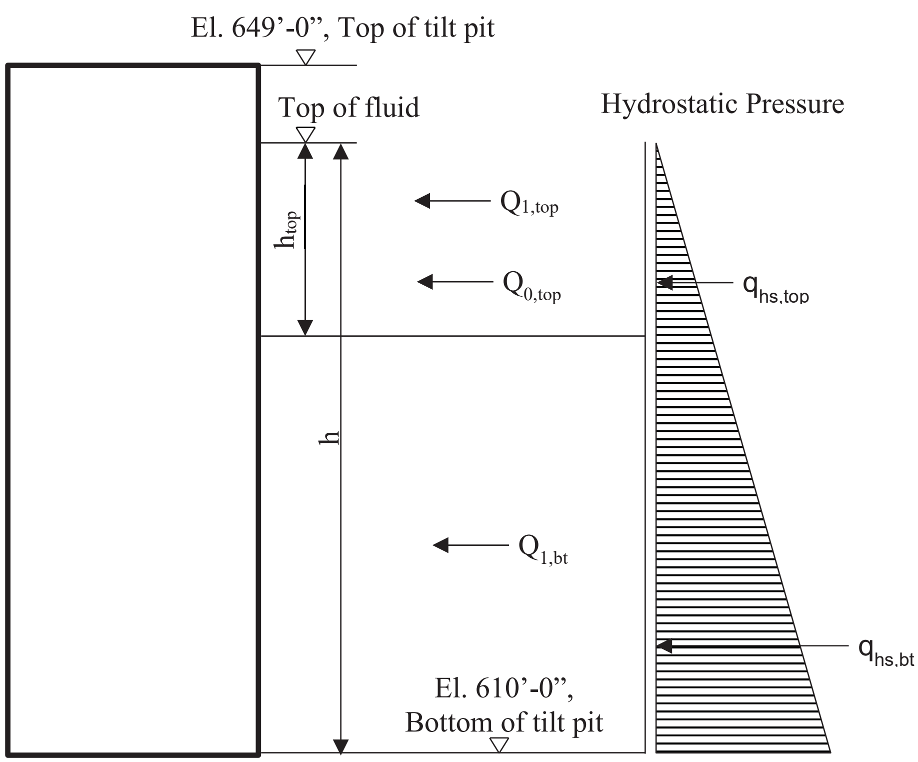

Figure 4-3: Wall SFP5 Vertical Cross Section

Case (2):

Hydrostatic loads are conservatively applied as a unit load with maximal amplitude. Additionally, the hydrodynamic impulsive load is applied as a point load. See Figure 4-3.

Hydrostatic load: $q_{hs} \quad = q_{hs,bt} = \rho_w \cdot g / g_c \cdot h \cdot b = 2452$ lbf/ft

Bottom water cg. location: $h_{bt} \quad = (h - h_{top}) / 2 = 11.78$ ft

Lateral acceleration at cg.: $a_{h,cg} \quad = h_{bt} \cdot (a_{h,i} - a_{h,bt}) / h_{top} + a_{h,bt} = 0.79$ g

Hydrodynamic load: $Q_{hd} \quad = Q_{1,bt} = M_{bt} \cdot a_{h,cg} / (g_c \cdot h) = 3097$ lbf

Design moment: $M_{bt} \quad = q_{hs} \cdot s^2 / 24 + Q_{hd} \cdot s / 8 = 14.7$ kip-ft, Reference [17]

The top and bottom longitudinal (parallel to X-axis) reinforcement evaluation is performed as a singly reinforced concrete section.

Concrete compressive strength including aging factor:
f'_c = 3600 psi, Reference [5]

Maximum concrete strain: ε_c = 0.003, Reference [18], Section 10.2.3

Concrete cover: c_c = 2", Reference [8]

Reinforcement: $d_{\#6}$ = 0.75", Reference [8]
$A_{s,\#6}$ = 0.44 in^2

Reinforcement spacing top and bottom faces: Reference [8]
s_{top} = 12"
s_{bot} = 12"

Tension reinforcement: $A_s = A_{s,\#6} \cdot b / s_{bot} = 0.44 \cdot 12 / 12 = 0.44\ in^2$

Reinforcement yield strength (ASTM A-15 Grade 40, References [5] and [8]):
f_y = 40 ksi

Reinforcement modulus of elasticity (Reference [18], Section 8.5.2):
E_s = 29000 ksi

Tolerance on the distance d, Reference [18], Section 7.5.2.1:
t_d = 0.5"

Distance from extreme compression fiber to centroid of longitudinal tension reinforcement (Figure 4-4):
$d = t_5 - c_c - d_{\#6} / 2 - t_d = 24 - 2 - 0.75 / 2 - 0.5 = 21.1"$

Tension force in the steel: $T_s = f_y \cdot A_s = 40 \cdot 0.44 = 17.6$ kips

Area of concrete compression:
$A_c = T_s / (0.85 f'_c) = 17.6 / (0.85 \cdot 3600) = 5.75\ in^2$

Depth of the equivalent compressed area:
$a = A_c / b = 5.75 / 12 = 0.48$ in

Factor relating depth of equivalent rectangular compressive stress block to neutral axis depth:
β_1 = 0.85, Reference [18], Section 10.2.7.3

Distance from extreme compression fiber to neutral axis:
$c = a / \beta_1 = 0.48 / 0.85 = 0.56$ in

Strain in the tension steel: $\varepsilon_t = \varepsilon_c \cdot d / c - \varepsilon_c = 0.003 \cdot 21.1 / 0.56 - 0.003 = 0.11$
$\varepsilon_t > 0.005$, tension controlled, Reference [18], Section 10.3.4

Distance from centroid of compressed area to extreme compression fiber:
$\lambda = a / 2 = 0.48 / 2 = 0.24$ in

Nominal moment capacity: $M_n = T_s \cdot (d - \lambda) = 17.6 \cdot (21.1 - 0.24) = 30.6$ ft-kip

Strength reduction factor: ϕ = 0.9

Ultimate moment capacity: $M_u = \phi \cdot M_n = 0.9 \cdot 30.6 = 27.6$ ft-kip

The bending failure of the wall is a ductile failure mode. An inelastic energy absorption factor is applicable:
F_μ = 1.25, Reference [15]

Interaction ratio: $IR = \max(M_{top}; M_{bt}) / (F_\mu \cdot M_u) = 14.7 / (1.25 \cdot 27.6) = 0.43$

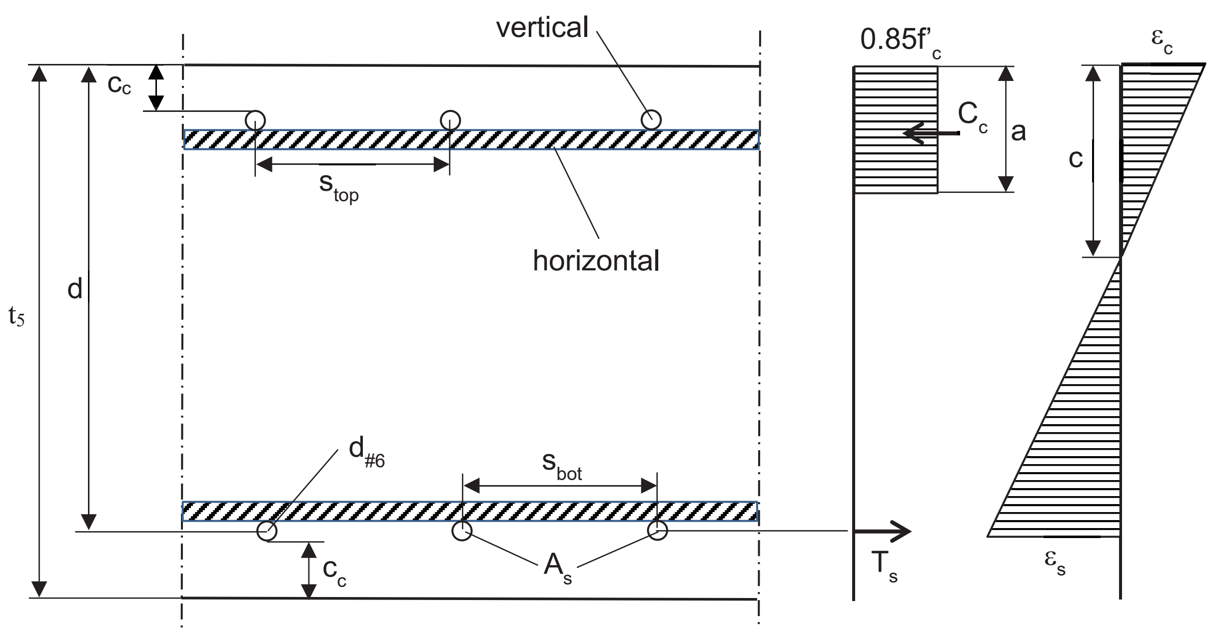

Figure 4-4: Wall SFP5 Horizontal Cross Section

The shear strength is calculated in accordance with ACI 349, Section 11.11 (Reference [18]), Provisions for slabs and footings.

By inspection, case (2) results in the largest shear force.

Hydrostatic shear force: $V_{hs} = q_{hs} \cdot 2B \cdot h/2 = 156$ lbf

Hydrodynamic load: $V_{hd} = Q_{hd} / 2 = 1549$ lbf

Nominal shear strength provided by concrete (Reference [18], Equation (11-31)):

$$v_c = 4 \cdot f'_c{}^{0.5} \cdot b \cdot d = 4 \cdot 3600^{0.5} \cdot 12 \cdot 21.1 = 60.8 \text{ kips/ft}$$

Ultimate shear capacity: $v_u = \phi \cdot v_c = 0.9 \cdot 60.8 = 51.7$ kips/ft

Interaction ratio: $IR = (V_{hs} + V_{hd}) / v_u = (0.156 + 1.549) / 51.7 = 0.03$

Local bending and shear capacity evaluations of wall SFP5 show available margin at the ultimate Auxiliary Building capacity level. Therefore, the filled tilt pits do not cause any local damage or failure.

4.2 Filled Tilt Pit, Global Response Effects

This section evaluates the global change in response of the Auxiliary Building due to the filled tilt pits. The change in frequency is used as a measure to determine the global affect where both North and South tilt pits filled are conservatively considered. A system natural frequency can be calculated utilizing $f = (k / m)^{0.5} / (2\pi)$, where k is the stiffness and m the mass of the system under consideration. The Auxiliary Building has a total mass of $m_{AB} = 2173$ kip·s²/ft (Reference [4]) and the two tilt pits add a mass $m_{tp} = 2 \cdot M / g_c = 16$ kip·s²/ft. An increase in mass of $(m_{AB} + m_{tp}) / m_{AB} = 0.74\%$ results in a reduction in frequency of $(1 / 1.0074)^{0.5} = 0.996$. The first six natural frequencies of the Auxiliary Building (1st and 2nd E-W mode, 1st and 2nd N-S mode, and sloshing modes, Reference [4]) determine the global response of the building where each responds at a frequency below the input motion spectral peak at 10 Hz (Figure 4-8). Hence, a reduction in natural frequency results in a reduction in input motion spectral amplification. However, the change in frequency due to the filled tilt pits is insignificant and the global response is deemed to not change because of the filled tilt pits.

4.3 HCLPF Capacity due to 10^{-5} AEF UHRS

Initially, a RE based on the GMRS seismic hazard level was selected for the PLP SPRA. This selection was informed by preliminary risk insights, prior to continued, iterative, refinement/improvement of the fragilities for top risk contributing structures, systems, and components. A discussion about insights gained from subsequent/later risk quantifications as they relate to the selection of an RE from a higher hazard bin, 10^{-5} AEF, is located in Reference [4], Chapter 10. This Section follows the structure of the SPRA analysis hazard level discussion and offers additional details and quantifications.

4.3.1 Soil Properties

The degraded soil properties at the 10^{-4} AEF, GMRS, and 10^{-5} AEF hazard levels are reviewed (Reference [3]). Shear wave velocity, compression wave velocity, and soil damping are the three main parameters that are different at the hazard levels. In the following a conservative evaluation of these parameters with respect to the impact on the seismic demand is performed:

Shear wave velocity:

Figure 4-5 shows the shear wave velocity soil profile at the PLP site at different seismic hazard levels (References [3] Tables 5-58 and 5-59, and [4] Table H1). The figure is focused on the soil profile from below grade elevation of 589 ft down to an elevation of 445 ft. This is because the soil degradation is primarily impacting the softer soil near the surface which is reasonable. Thereby, it is observed that the 10^{-5} AEF hazard level results in a lower shear wave velocity profile than the RE one. Below an elevation of 445 ft there is a step up in shear wave velocity to a plateau at 5300 ftp (ft/sec). Shear wave velocities below the elevation of 445 ft do not change significantly at the different hazard levels. The primary soil column frequency affecting the site can be calculated with

$$f = V_s / (4 \cdot h) \qquad \text{Equation [4-1]}$$

where V_s is the shear wave velocity and h the soil column height (Reference [11]). For the GMRS hazard level soil column frequency estimate, the mean shear wave velocity $V_{s,RE,m} = 1160$ fps of the softer subsurface soil and the corresponding soil column height of h = 589 - 444.7 = 144.3 ft is used (Figure 4-5), resulting in a frequency of $f_{RE} = 1160 / (4 \cdot 144.3) = 2$ Hz. Compared with Figure 4-9, a horizontal basemat response of the Auxiliary Building, the 2 Hz soil column frequency seems reasonable where additional amplifications due to the 1.8 Hz and 4.93 Hz first Y- and X-directional structural modes (Reference [4]) are visible, respectively. Due to SSI effects the structural modes are shifted towards the lower frequency range. At the 10^{-5} AEF seismic hazard level the soil column frequency drops to $f_{10^{-5}} = V_{s,10^{-5},m} / (4 \cdot h) = 1076 / (4 \cdot 144.3) = 1.86$ Hz (Figure 4-5). The amplitudes of the UHRS at the 10^{-5} AEF at 1.86 Hz $A_{1.86} = 0.663$g and at 2 Hz $A_2 = 0.69$g. A general response amplitude reduction of $A_{1.86} / A_2 = 0.96$ (-4%) in horizontal direction results. This amplitude reduction is primarily applicable to the soil modes below approximately 10 Hz since the horizontal UHRS shape peaks at 10 Hz (Figure 4-8). Also, responses due to structural modes will benefit from the reduction in shear wave velocity as it reduces the input motion to the structure. However, this amplitude reduction shall be considered as a general estimate since it is based on averaged properties and the UHRS ramp up to 10 Hz (Figure 4-8) has different slopes affecting the amplitude reductions differently in different frequency regions.

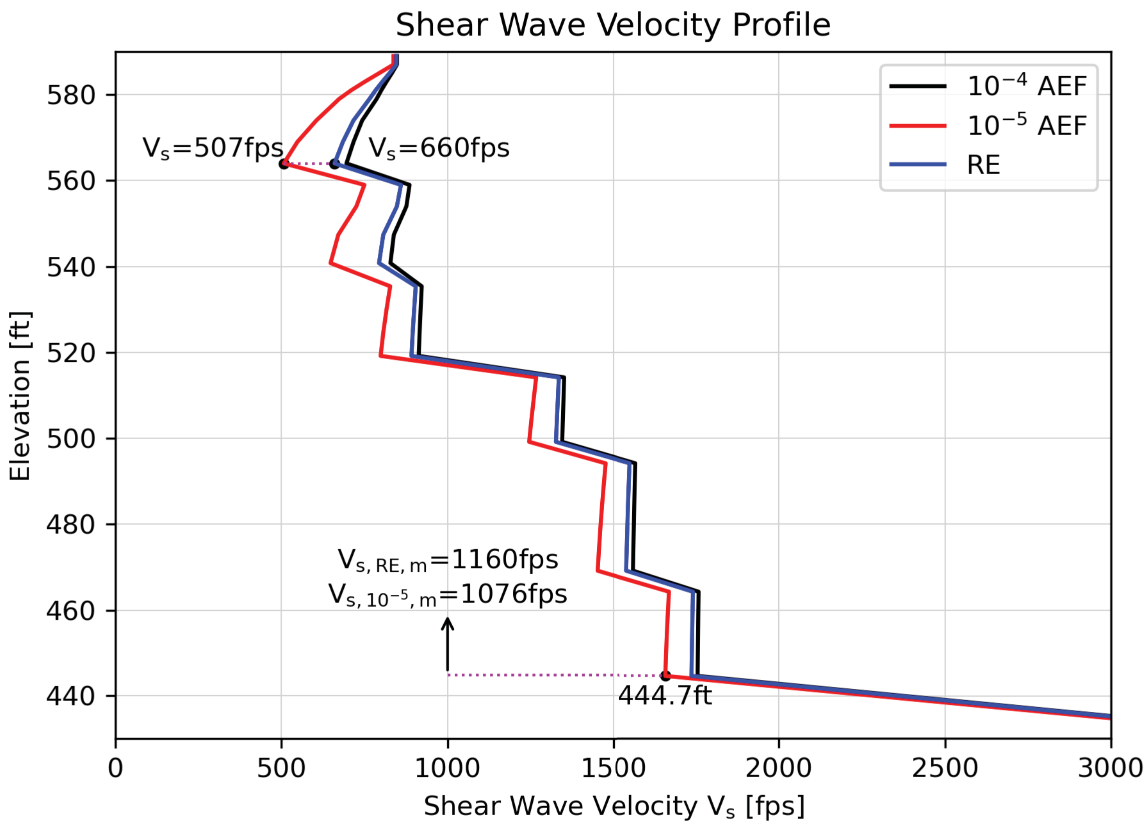

Figure 4-5: Shear Wave Velocity Profile

<u>Compression wave velocity:</u>

Figure 4-6 shows the compression wave velocity soil profile at the PLP site at different seismic hazard levels (References [3] Tables 5-58 and 5-59, and [4] Table H1). Similarly to the shear wave velocity profile, the figure is focused on the soil profile from below grade elevation of 589 ft down to an elevation of 445 ft because of the soil degradation primarily impacting the softer soil near the surface. Thereby, it is observed that the 10^{-5} AEF hazard level results in a lower compression wave velocity profile than the RE one. Below an elevation of 519 ft there is a first step up in compression wave velocity to a plateau at 6800 ftp and a second step up below elevation 445 ft to a plateau at 9700 ftp. Compression wave velocities below the elevation of 445 ft do not change significantly at the different hazard levels. The primary soil column frequencies affecting the site can be calculated utilizing Equation [4-1] with $f = V_p / (4 \cdot h)$, where V_p is the compression wave velocity and h the soil column height. For the GMRS hazard level soil column frequency estimate, the mean compression wave velocities $V_{p,RE,m}$ = [3669; 6804] fps of the softer subsurface soil and the corresponding soil column heights of h = [589 - 519.2; 519.2 - 444.7] = [69.8; 74.5] ft is used (Figure 4-6), resulting in a frequency of f_{RE} = [3669 / (4·69.8); 6804 / (4·74.5)] = [13.1; 22.8] Hz. Compared with Figure 4-11, a vertical basemat response of the Auxiliary Building, the 13.1 Hz and 22.8 Hz soil column frequencies seems reasonable when compared to the amplification peaks at 14.5 Hz and at 25 Hz (Reference [4]). At the 10^{-5} AEF seismic hazard level the soil column frequency drops to $f_{10^{-5}} = V_{p,10^{-5},m}$ / (4·h) = [3222 / (4·69.8); 6460 / (4·74.5)] = [11.5; 21.7] Hz. The amplitudes of the UHRS at the 10^{-5} AEF at 11.5 Hz $A_{11.5}$ = 1.184g, at 13.1 Hz $A_{13.1}$ = 1.198g, at 21.7 Hz $A_{21.7}$ = 1.263g, and at 22.8 Hz $A_{22.8}$ = 1.266g. A general response amplitude reduction of -0.5% in vertical direction results ($A_{11.5} / A_{13.1}$ = 0.99 and $A_{21.7} / A_{22.8}$ = 1.0). This amplitude reduction is applicable to the soil modes but also benefits structural modes. This amplitude reduction shall be considered as a general estimate since it is based

on averaged properties and the UHRS ramp up to 33 Hz (Figure 4-10) has different slopes affecting the amplitude reductions differently in different frequency regions.

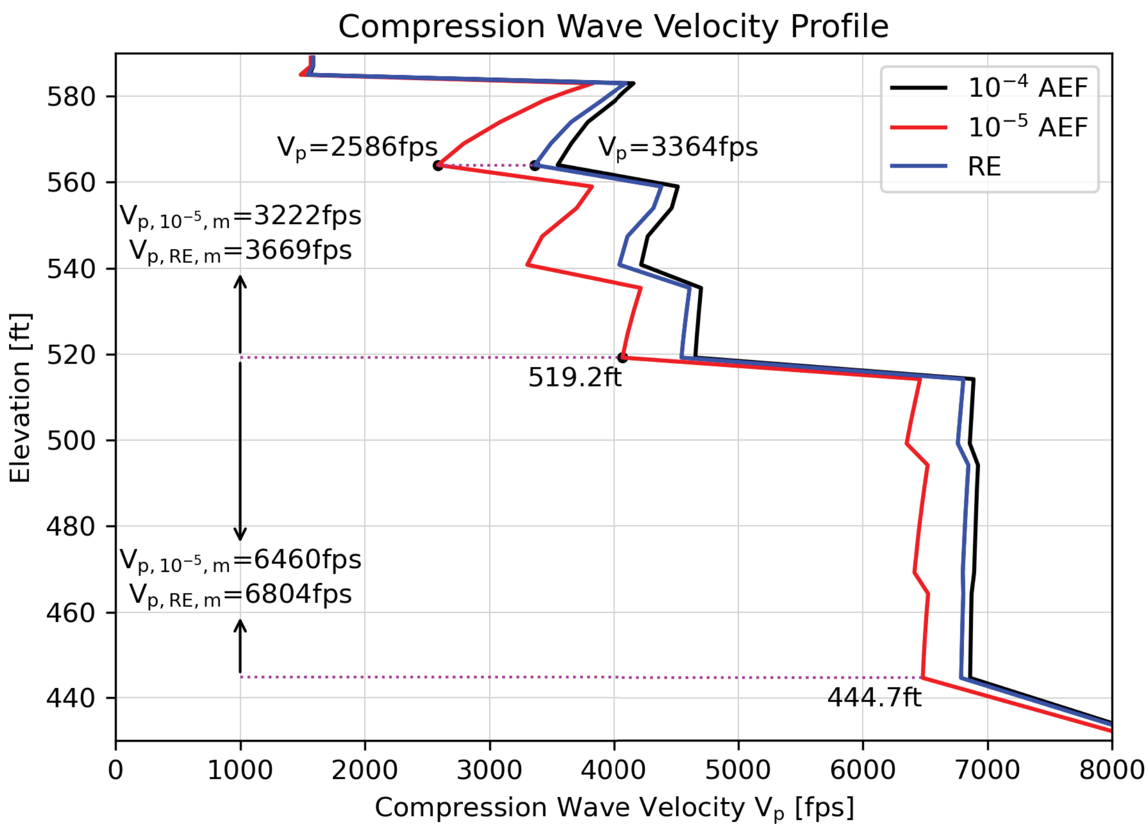

Figure 4-6: Compression Wave Velocity Profile

<u>Soil damping:</u>

Figure 4-7 shows the soil damping profile at the PLP site at different seismic hazard levels (Reference [3]). This damping is applied to both shear and compression wave velocity profiles (Reference [4]). Figure 4-7 is focused on the soil profile from below grade elevation of 589 ft down to an elevation of 445 ft. While there are differences in damping down to about an elevation of 89 ft the damping is most effective in the area with the most oscillatory motion - the near surface area with soil degradation (see shear and compression wave velocity). Thereby, it is observed that the 10^{-5} AEF hazard level results in a larger soil damping profile than the RE one. Damping in general is primarily affecting the magnitude of an oscillatory motion. To quantify the effect of soil damping on the overall response the magnification function M of a harmonic oscillator is utilized (Reference [12]):

$$M = \frac{1}{\sqrt{(1-\eta^2)^2+(2\cdot D\cdot \eta)^2}} \qquad \text{Equation [4-2]}$$

Thereby, D is the damping ratio and the ratio $\eta = \omega / \omega_0$ with ω being the driving force angular frequency and ω_0 the natural system frequency. In the resonant case $\eta = 1.0$, the magnification function reduces to $M = 1 / (2\cdot D)$. For the GMRS hazard level, the mean damping $D_{RE,m} = 2.4\%$ of the softer subsurface soil is used (Figure 4-7), resulting in a damping magnification $M_{RE} = 1 / (2\cdot D_{RE,m}) = 1 / (2\cdot 0.024) = 20.8$. At the 10^{-5} AEF seismic hazard level the magnitude drops to $M_{10^{-5}} = 1 / D_{10^{-5},m} = 1 / (2\cdot 0.039) = 12.8$. A general response amplitude reduction of $M_{10^{-5}} / M_{RE} = 0.62$ (-38%) in horizontal and vertical direction results. This amplitude reduction is applicable to the soil modes but also reduces the amplitudes of input ground motions that the structure experiences.

However, this amplitude reduction shall be considered as a general estimate since it is based on averaged properties.

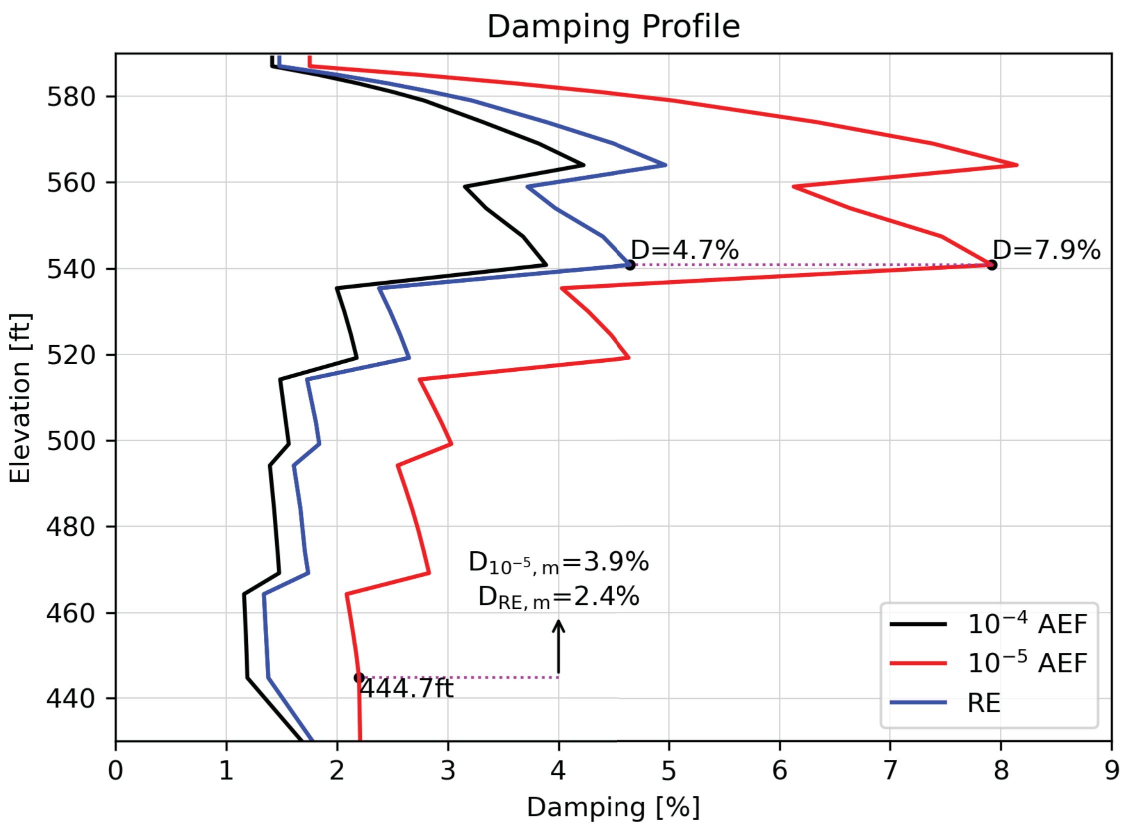

Figure 4-7: Damping Profile

4.3.2 Horizontal Spectral Shape

The horizontal RE (GMRS consistent FIRS), and UHRS at the 10^{-4} and 10^{-5} AEF hazard levels are shown in Figure 4-8 (Reference [3], Table 5-56). The RE is scaled-up to the 10^{-5} AEF hazard level using PGA scaling and is also plotted in Figure 4-8. A scale factor SF = $PGA_{10^{-5}}$ / PGA_{RE} = 0.569 / 0.278 = 2.05 is used. In addition, a frequency dependent ratio S-Ratio = $A_{S\text{-}RE}$ / $A_{10^{-5}}$ is developed using the amplification of the 10^{-5} AEF UHRS, $A_{10^{-5}}$ and the amplification of the scaled RE, $A_{S\text{-}RE}$. Figure 4-8 shows the ratio curve as well as the PGA based ratio (S-Ratio = 1.0) as a measure to determine which frequency regions are amplified or deamplified if the seismic hazard level is scaled from the RE to the 10^{-5} AEF UHRS. Similarly, to the conclusions drawn in Reference [4], the spectral amplifications show more amplification in the low frequency range (< 3 Hz), and more deamplification at high frequencies (> 4 Hz) for 10^{-5} AEF seismic hazard level than for the scaled RE hazard level. The largest differences S_{min} = 0.97 and S_{max} = 1.02 indicate that the spectral amplification differences across the frequency range are small.

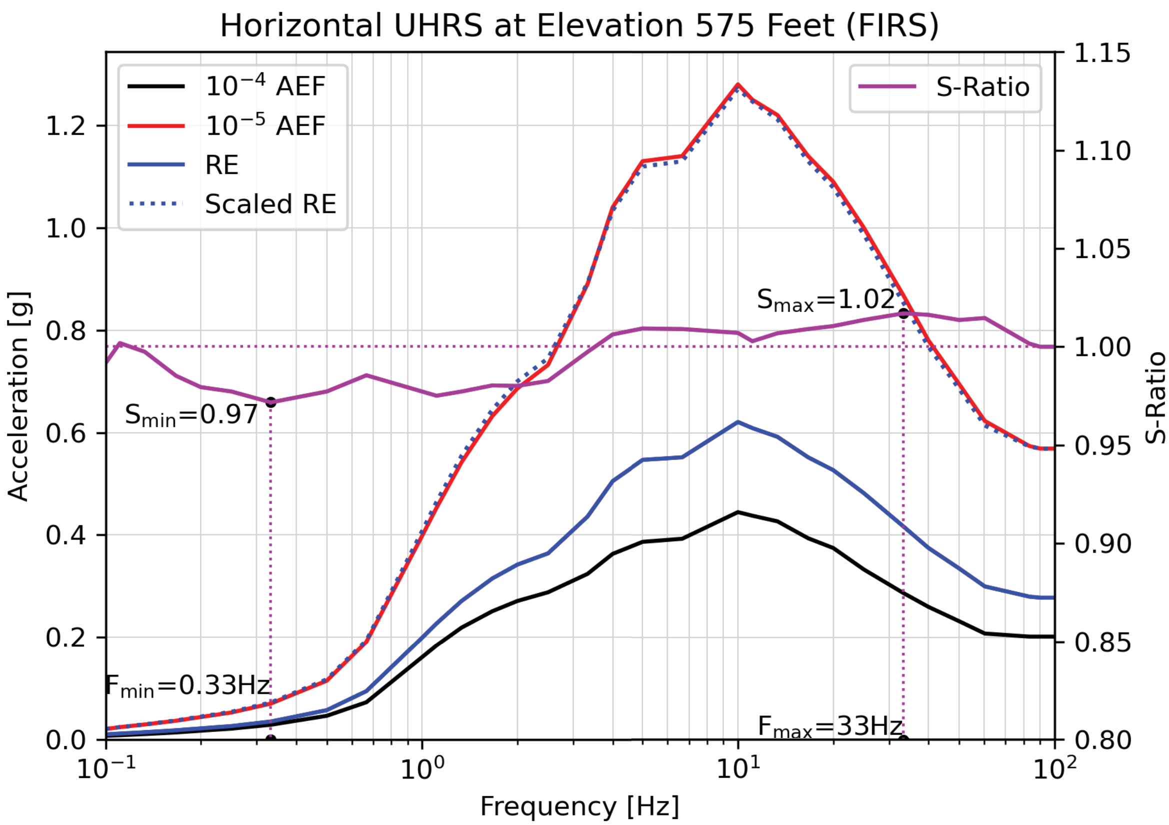

Figure 4-8: Horizontal UHRS at Elevation 575 Feet (FIRS)

To evaluate the slight change in response spectrum shape when the 10^{-5} AEF UHRS is used on the risk insights, the Auxiliary Building In-Structure Response Spectra (ISRS) in horizontal directions are plotted and compared to the RE in Figure 4-9 (Reference [4], page I8-5). The selected location within the Auxiliary Building is AB-570-5 which is a location on the basemat adjacent to the exterior shear wall that governs the Auxiliary Building fragility. This location is representative to provide insights into soil-structure interaction frequencies as well as global building natural frequencies. The comparison plot shows that there are resonant modes beyond approximately 4 Hz causing amplification. In the 1 Hz to 4 Hz frequency range the system enters deamplification. As a result, it is concluded that a -3% (S_{min} = 0.97) decrease in horizontal demand due to the use of a 10^{-5} AEF UHRS is conservative (based on linear scaling of the RE).

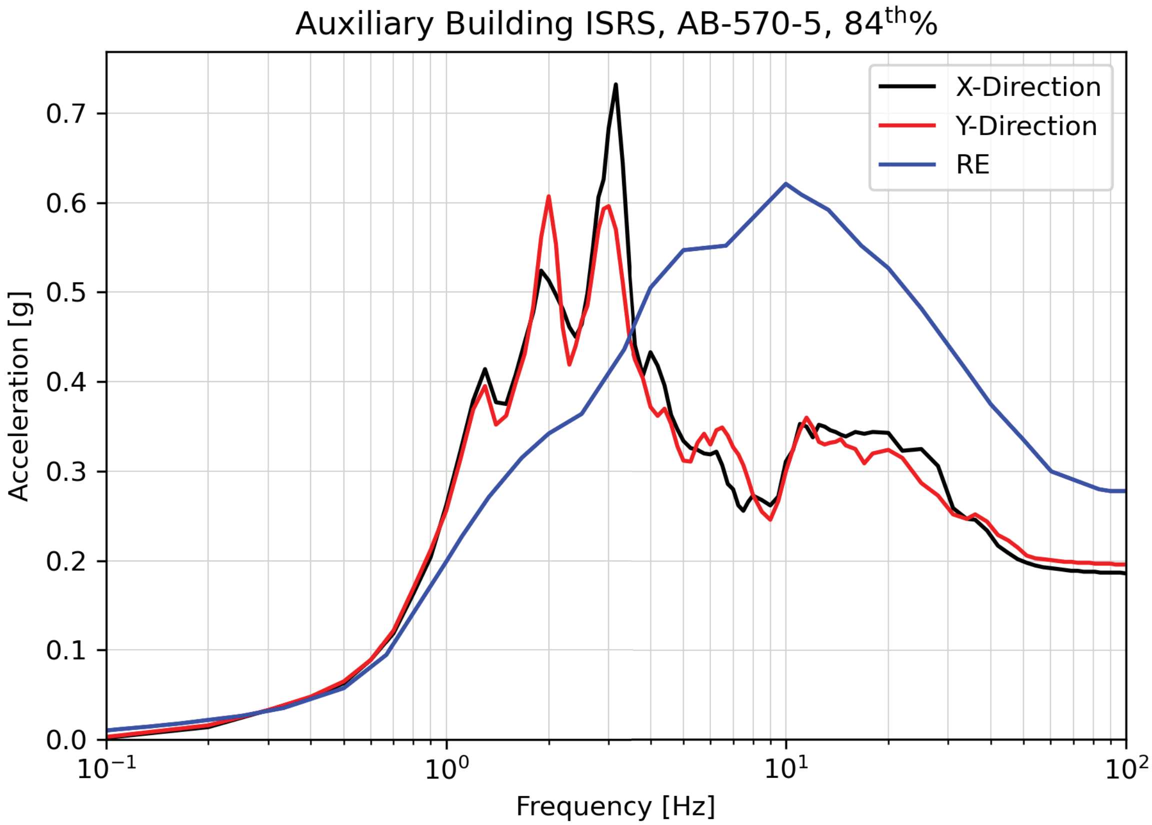

Figure 4-9: Auxiliary Building Horizontal ISRS, AB-570-5, D=5%, 84th%

4.3.3 Vertical Spectral Shape

The vertical RE (GMRS consistent FIRS), and UHRS at the 10^{-4} and 10^{-5} AEF hazard levels are shown in Figure 4-10 (Reference [3], Tables 5-53 and 5-56). Since the HCLPF is expressed in reference to the horizontal PGA the vertical RE is scaled-up to the 10^{-5} AEF hazard level using the horizontal PGA scale factor SF = $PGA_{10^{-5}}$ / PGA_{RE} = 0.569 / 0.278 = 2.05. In addition, a frequency dependent ratio S-Ratio = $A_{S\text{-}RE}$ / $A_{10^{-5}}$ is developed using the amplification of the scaled RE, $A_{S\text{-}RE}$ and the amplification of the 10^{-5} AEF UHRS, $A_{10^{-5}}$. Figure 4-10 shows the ratio curve as well as the PGA based ratio (S-Ratio = 1.0) as a measure to determine the difference between the scaled RE and the 10^{-5} AEF UHRS. As a result, the spectral amplifications of the RE are consistently below the 10^{-5} AEF.

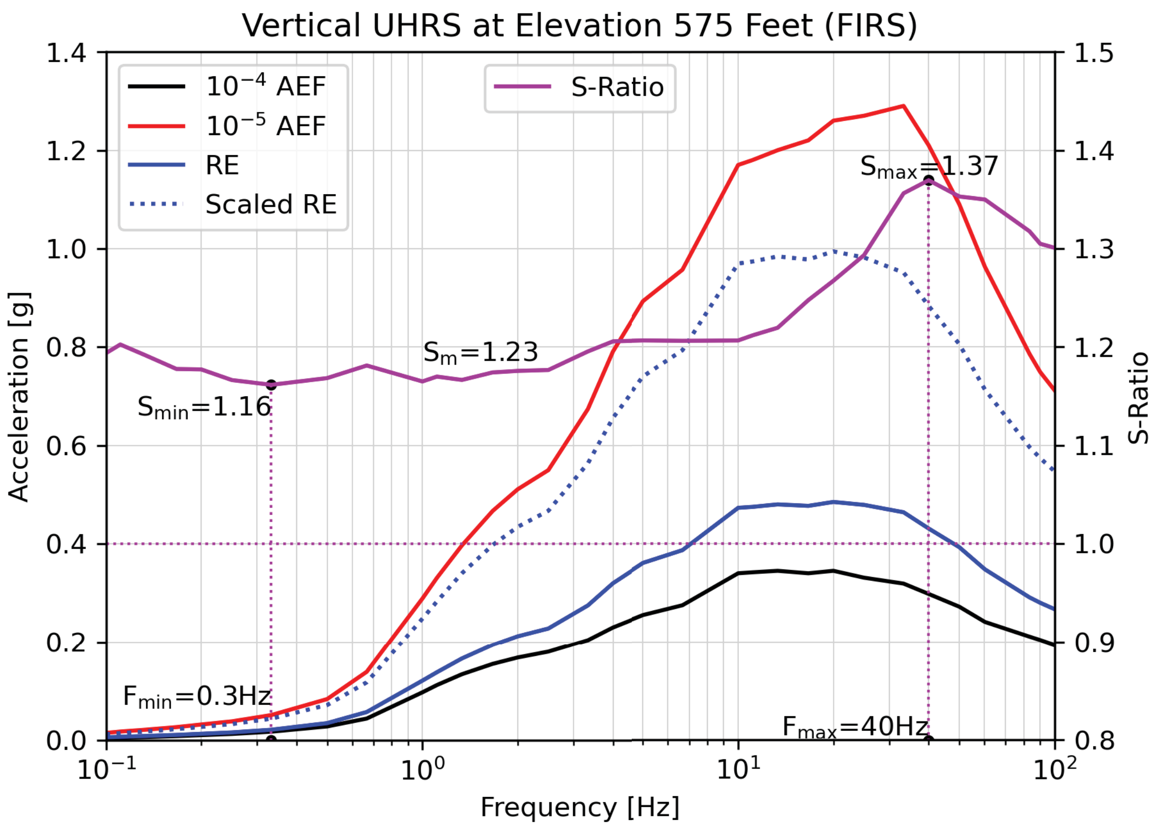

Figure 4-10: Vertical UHRS at Elevation 575 Feet (FIRS)

To evaluate the change in response spectrum shape when 10^{-5} AEF UHRS is used on the risk insights, the Auxiliary Building ISRS in vertical direction is plotted and compared to the RE in Figure 4-11 (Reference [4], page I8-6). The selected location within the Auxiliary Building is AB-570-5 which is a location on the basemat adjacent to the wall that governs the Auxiliary Building fragility. This location is representative to provide insights into soil-structure interaction frequencies as well as global building natural frequencies. The comparison plot shows that there are resonant modes across the entire frequency range causing amplification where major amplifications in the approx. 8 Hz to 50 Hz range are observed. As a result, it is concluded that a 37% (S_{max} =1.37) increase in vertical demand due to the use of a 10^{-5} AEF UHRS is conservative (based on linear scaling of the RE).

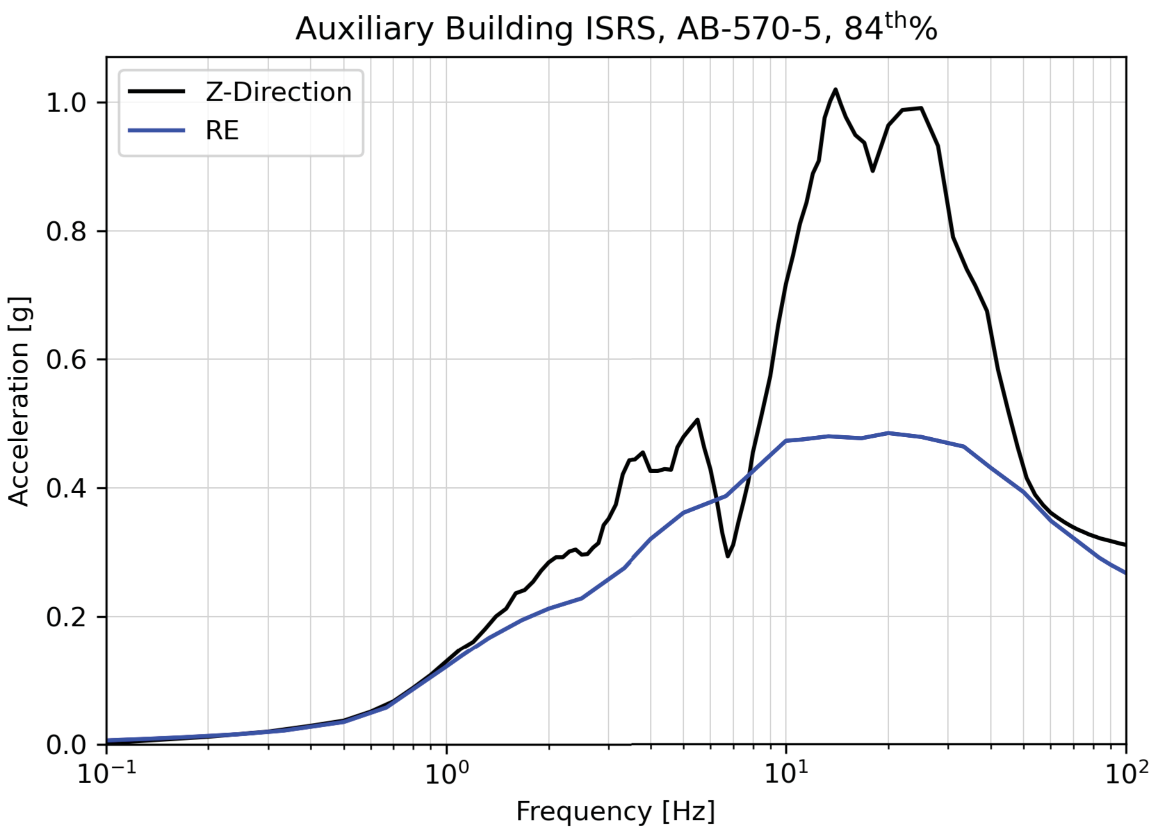

Figure 4-11: Auxiliary Building Vertical ISRS, AB-570-5, D=5%, 84th%

4.3.4 Vertical/Horizontal Ratio

Vertical UHRS at 10^{-4} AEF, GMRS, and 10^{-5} AEF and V/H ratio functions at 10^{-4} AEF and 10^{-5} AEF are reviewed (Reference [3]). The 10^{-5} AEF PGA (0.569g) is close to the 0.5g threshold between NUREG-6728 (Reference [2]) V/H ratio curves, so the 0.2g-0.5g curve would remain appropriate since it is arguably more realistic than the curve for higher acceleration levels. Therefore, the V/H ratio is effectively not different at 10^{-5} AEF than GMRS.

In addition, the V/H ratio functions for the PLP site are developed based on the average of the V/H ratio function from Detroit Edison in 2014 (DTE) and the one developed by the Exelon Generation Company in 2006 (Reference [3]). The comparison plot in Figure 3-8 of Reference [3] shows that the PLP V/H ratio function is conservative in the frequency range of approx. 10 Hz to 60 Hz over the CEUS V/H ratio function (Reference [2]). Since the 10^{-5} AEF PGA just exceeds the threshold while there is some conservatism in the utilized V/H ratio function the GMRS V/H ratio is judged to also apply at the 10^{-5} AEF seismic hazard level.

4.3.5 Auxiliary Building Concrete Cracking

While two below grade shear walls in the Auxiliary Building experienced concrete cracking at the GMRS seismic hazard level the entire building was judged (Reference [4]) to respond consistent with uncracked concrete property behavior. The SPRA also concludes that there is likely to be further cracking at the 10^{-5} hazard level reducing demands slightly. Concrete cracking generally has two effects: the cracking reduces the element stiffness and the material experiences larger damping. The governing fragility failure mode for the Auxiliary Building is a shear failure of an exterior wall that belongs to the primary lateral force resisting system (Reference

[5]). The exterior wall under consideration is oriented in the North-South direction (X-direction) on the West side of the building (Reference [4]).

Reduction in stiffness:

Since the governing failure mode of the primary lateral force resisting system is a global failure mode the reduction in stiffness of the global response is evaluated. The reduction in stiffness would also apply to local modes like out-of-plane responses of slabs and walls. These local modes are typically at higher frequencies and the corresponding amplitude could see an increase due to the concrete cracking induced stiffness reduction. This is because all modes that are located beyond the ground motion peak get a larger ground motion amplitude if their natural frequency shifts lower. However, it takes a relatively large seismic load to induce concrete cracking in local structural members, and the horizontal peak is at approx. 10 Hz (Figure 4-8) applicable to walls and the vertical peak is at approx. 33 Hz (Figure 4-10) applicable to slabs, limiting the number of affected walls and slabs significantly. Therefore, local modes are not risk-significant and are not further evaluated. The contributing global structural mode is the one at 5.16 Hz (Reference [4] Table C47) in the North-South direction. Due to concrete cracking the frequency drops to 4.93 Hz (Reference [4] Table C48). However, the frequencies are based on a fixed base modal analysis. If SSI effects are included the uncracked frequency of 5.16 Hz drops to approx. 3.15 Hz (Figure 4-9). The location of the response is in a frequency region (below the 10 Hz peak) where the response amplitudes would benefit from a frequency shift towards the lower frequency range. While the response reduction potential due to concrete cracking is noted, it cannot be accurately estimated due to effect of SSI.

Increase in damping:

At the GMRS hazard level "Response Level 1" damping ratios D_{RL1} = 4% are used for uncracked concrete. At the 10^{-5} AEF seismic hazard level "Response Level 2" damping ratios D_{RL2} = 7% for cracked concrete can be used. Utilizing Equation (4-2) a general response amplitude reduction of D_{RL1} / D_{RL2} = 0.57 (-43%) results. This amplitude reduction is applicable to the global and local structural modes.

5.0 RESULTS

The local effects of the filled tilt pits are reviewed by evaluation of the representative tilt pit wall that is most susceptible to damage or failure. Bending and shear capacities are computed at the Auxiliary Building HCLPF capacity level resulting in available margin.

Filling the tilt pits with coolant water is reviewed with regards to the effects on the global Auxiliary Building response. The added water mass in the tilt pits is found to have an insignificant effect.

The Auxiliary Building fragility is governed by an exterior shear wall failure of the primary lateral force resisting system. The corresponding HCLPF is 0.71g and is referenced to the RE at the GMRS seismic hazard level. It is established that the SFP capacity against local failure is large, and the capacity of the Auxiliary Building governs the SFP HCLPF which is also true at the 10^{-5} AEF seismic hazard level. This calculation evaluates the potential change in HCLPF capacity if the 10^{-5} AEF hazard level was used to develop the HCLPF. Different factors affecting the HCLPF capacity of the Auxiliary Building at PLP are evaluated (Table 5-1). For a conservative estimate the change in shear wave velocity, soil damping, horizontal spectral shape, and damping due to concrete cracking is judged to directly apply to the existing HCLPF capacity. The contribution of the vertical direction (change in compression wave velocity and vertical spectral shape) to the shear failure of the exterior wall is generally weighted less than the horizontal contribution because the potential shear failure plane is horizontal. Nevertheless, a precise change in HCLPF capacity of the Auxiliary Building would require the knowledge of how the developed factors interact with each other. Some factors only apply to the soil systems, to the structure, or the direction and an accurate change in HCLPF capacity is not possible. However, it is evident that there is margin if the Auxiliary Building HCLPF capacity was developed using the 10^{-5} AEF seismic hazard. Conservatively, the 10^{-5} AEF HCLPF capacity of the Auxiliary Building is set to 0.71g (same as current GMRS base capacity) which is also larger than the $PGA_{10^{-5}}$ of 0.569g.

Table 5-1: Summary of Factors Affecting the 10^{-5} AEF UHRS Auxiliary Building HCLPF

Item	Effect on the 10^{-5} AEF UHRS*
Shear wave velocity	0.96
Compression wave velocity	0.995
Soil damping	0.62
Horizontal spectral shape	0.97
Vertical spectral shape	1.37
Vertical/horizontal ratio	1.00
Concrete cracking stiffness reduction	potential reduction not credited
Concrete cracking increase in damping	0.57

*Factors which are less than 1.0 generally result in an increased HCLPF value

6.0 CONCLUSION

It is concluded that the filling with cooling water of the north tilt pit and/or the south tilt pit does not affect the HCLPF capacity of the SFP. The HCLPF capacity of the SFP is determined to be governed by the Auxiliary Building. This calculation also evaluates the impact on the HCLPF capacity if the 10^{-5} AEF seismic hazard level was used to determine the structural fragility as opposed to the GMRS hazard level. The determination is based on the evaluation of a variety of parameters that have significant effects on the Auxiliary Building capacity. It is concluded that there is sufficient margin that the Auxiliary Building HCLPF capacity of 0.71g is also valid if the capacity evaluation was based on a 10^{-5} AEF seismic hazard. Therefore, the SFP HCLPF capacity is larger than the $PGA_{10^{-5}}$ of 0.569g.

7.0 REFERENCES

References identified with an (*) are maintained within Palisades Records System and are not retrievable from Framatome Records Management. These are acceptable references per Framatome Administrative Procedure 0402-01, Attachment 7. See page 2 for Project Manager Approval of customer references.

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[2] US-NRC, NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines," May 2001.

[3] Amec Foster Wheeler Environment & Infrastructure, Inc., "Final Seismic Hazard Evaluation Report Palisades Nuclear Plant South Haven, Michigan," Revision 0, December 2015.

[4] *PLP Document PSA-PLP-04-01, Revision 0, "Palisades Nuclear Plant Seismic Probabilistic Risk Assessment: Structural Response Analysis."

[5] *PLP Document PSA-PLP-04-04, Revision 0, "Palisades Nuclear Plant Seismic Probabilistic Risk Assessment: Structural Fragility Analysis."

[6] *PLP Drawing C-107, Revision 28, "Auxiliary Building Floor Plans at Elev. 602'-0" 607'-6" & 611'-0"."

[7] *PLP Drawing C-109, Revision 10, "Auxiliary Building Roof Plan at El. 649'-0"."

[8] *PLP Drawing C-110, Revision 7, "Auxiliary Building Spent Fuel Pool - Elevations & Sect. Sheet No. 1."

[9] *PLP Drawing C-111, Revision 6, "Auxiliary Building Spent Fuel Pool - Elevations & Det. Sheet No. 2."

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