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10 CFR 50.12
10 CFR 50.47
10 CFR 50, Appendix E

HDI-IPEC-22-014

February 2, 2022

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

Subject: Revision to Holtec Decommissioning International, LLC (HDI) Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E for Indian Point Unit Nos. 1, 2, and 3

Indian Point Nuclear Generating Unit Nos. 1, 2, and 3
Docket Nos. 50-003, 50-247, and 50-286
Provisional Operating License No. DPR-5
Renewed Facility License No. DPR-26
Renewed Facility License No. DPR-64

References:

1) Letter from Holtec Decommissioning International, LLC (HDI) to U.S. Nuclear Regulatory Commission (NRC), " Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E," (Letter HDI-IPEC-21-015) (ADAMS Accession No. ML21356B693), dated December 22, 2021.

2) Letter from Holtec International, to U.S. NRC, “Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML20280A524), dated September 29, 2020.

3) Letter from Holtec International to US NRC, “Response to Request for Additional Information –Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML21148A289) dated May 28, 2021.

4) Letter from Holtec International to US NRC, “Revised Response to Request for Additional Information—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML21228A262) dated August 16, 2021.

5) Letter from Holtec International to US NRC, “Response to Request for Additional Information 10—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. 211291A161) dated October 18, 2021.

6) Letter from HDI to U.S. NRC, " Supplement to Holtec Decommissioning International, LLC (HDI) Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E for Indian Point Unit Nos. 1, 2, and 3 Including Site-Specific Calculations," (Letter HDI-IPEC-22-013) (ADAMS Accession Nos. ML22032A017 and ML22032A027, dated February 1, 2022.

7) Letter from US NRC to Holtec International, “ Draft Safety Evaluation Report—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” dated December 20, 2021.

Dear Sir or Madam:

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 Indian Point 2, LLC (IP1 & IP2) and Holtec Indian Point 3, LLC (IP3), collectively referred to as Indian Point Energy Center (IPEC), submitted a request for exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E, Section IV on December 22, 2021 (Reference 1). The requested exemptions would allow HDI to reduce IPEC emergency planning requirements consistent with the permanently shutdown and defueled condition of Indian Point 1 (IP1), Indian Point 2 (IP2), and Indian Point 3 (IP3).

The HDI request for exemptions utilizes the methodology developed by Holtec International and submitted for NRC review and approval on September 29, 2020 (Reference 2).

On February 1, 2022, HDI submitted a Supplement to the pending request for exemptions submitted in Reference 1 (Reference 6). The Supplement provided site-specific calculations for Indian Point Units 2 & 3 (IP2 & IP3) utilizing the Holtec methodology provided in the Holtec Topical Report as revised in responses to NRC Requests for Additional Information (RAIs) (References 3, 4 and 5). This letter is provided to update the references to the Holtec Topical Report (Reference 2), the Holtec RAI responses (References 3, 4 and 5), and the site-specific calculations utilized to determine the end of the zirconium fire Period for IP2 and IP3 for the PDEP implementation as requested by HDI in Reference 1.

Enclosure 1 to this letter provides a revised Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47(b), 10 CFR 50.47(c)(20), and 10 CFR Part 50, Appendix E. The revised Enclosure 1 updates the references for the exemption requests and replaces Enclosure 1 in the December 22, 2021, HDI request for exemptions (Reference 1). Note that the calculations provided in Reference 2 are the same as those that were originally provided in the original HDI request for exemptions. This revised request is provided to simplify the documentation references for NRC review.

Should you have any questions or require additional information, please contact Mr. Walter Wittich, IPEC Licensing at 914-254-7212 or myself at 856-797-0900 ext. 3578.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 1, 2022.

Sincerely,

Jean A. Fleming

Digitally signed by Jean A. Fleming
Date: 2022.02.02 12:53:09 -05'00'

Jean A. Fleming
HDI Vice President, Regulatory and Environmental Affairs
Holtec Decommissioning International, LLC

Enclosures:

1. Revised 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 (February 2022)

cc: NRC Senior Project Manager, NRC NRR DORL
NRC Region I Regional Administrator
NRC Senior Regional Inspector, Indian Point Energy Center
New York State Liaison Officer Designee, NYSERDA
New York State (NYS) Public Service Commission

Enclosure 1
HDI-IPEC-22-014

Revised Request for Exemptions from Certain Emergency Planning Requirements of10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E

(98 Pages)

1.0 SPECIFIC EXEMPTION REQUEST

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 Indian Point 2, LLC (IP1 & IP2) and Holtec Indian Point 3, LLC (IP3), collectively referred to as Indian Point Energy Center (IPEC)),requests exemptions from the following regulations:

- Certain standards in 10 CFR 50.47(b) regarding onsite and offsite emergency response plans for nuclear power reactors;
- Certain requirements of 10 CFR 50.47(c)(2) to establish plume exposure and ingestion pathway Emergency Planning Zones (EPZs) for nuclear power plants; and
- Certain requirements of 10 CFR Part 50, Appendix E, which establish the elements that make up the content of Emergency Plans.

The IPEC Emergency Plan encompasses IP1, IP2, and IP3.

The underlying purpose of the 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR 50 Appendix E, Section IV 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 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 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 shutdown from those for an operating power reactor. However, once a plant is permanently shutdown and defueled, and a sufficient decay of the spent fuel has occurred, some of these requirements exceed what is necessary to protect the health and safety of the public. The requested exemptions would allow HDI to reduce emergency planning requirements and subsequently revise the IPEC Emergency Plan to reflect the permanently shutdown and defueled condition of the station.

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

IPEC is located on the east bank of the Hudson River at Indian Point, in the Village of Buchanan, in upper Westchester County, New York. The site is operated by Holtec Decommissioning International, LLC (HDI) and contains facilities located on approximately 239 acres, bounded on the north, south, and east by privately owned land and on the west by the Hudson River. IP2 and IP3 are located north and south, respectively, of IP1, which is in safe storage (SAFSTOR) until subsequent decommissioning. The site is located about 24 miles north of the New York City boundary line. The nearest urban area within 6 miles of the site is the City of Peekskill, New York, which is located approximately 2.5 miles northeast of the IPEC site.

IP1 was permanently shutdown on October 31, 1974, and all spent fuel was removed from the IP1 reactor vessel in 1975. All spent fuel has since been removed from the IP1 Spent Fuel Pool (SFP) and transported offsite or placed in the existing Independent Spent Fuel Storage Installation (ISFSI) as reported in the Entergy letter dated December 11, 2008 (Reference 2). The IP1 Provisional Operating License prohibits taking the reactor to criticality or operation of the facility at any power level, and the IP1 Technical Specifications do not allow fuel to be loaded into the reactor core or moved into the reactor containment building without prior review and authorization by the U.S. Nuclear Regulatory Commission (NRC). The IP1 Technical Specifications also preclude fuel from being stored in the IP1 fuel storage area. IP1 is being maintained in SAFSTOR status. There are ongoing activities in the IP1 space that are support services for IP2 and, to a lesser extent, IP3. Only those areas that either store or process radioactive materials (the Fuel Handling Building (FHB) and waste storage/process areas in the Chemical Systems Building and the Integrated Liquid Radwaste Systems Building) are considered in evaluating the radiological hazards for the IPEC Emergency Plan. Based on its current configuration and licensing basis, with no spent fuel stored in the IP1 SFP, there are no postulated Design Basis Accidents (DBAs) that remain applicable to IP1 (Reference 3). The IP1 SFP is no longer in use because all spent fuel and other material has been removed, and the IP1 SFP has been drained.

By letter dated February 8, 2017 (Reference 4), in accordance with 10 CFR 50.82(a)(1)(i), Entergy submitted certification to the NRC indicating its intention to permanently cease power operations at IP2 and IP3 by April 30, 2020, and April 30, 2021, respectively.

By letter dated April 15, 2020 (Reference 5), the NRC issued Amendment Nos. 62, 293, and 268 for IP1, IP2, and IP3, respectively, approving the IPEC Post-Shutdown Emergency Plan (PSEP). The changes to the IPEC Emergency Plan that were approved for the PSEP support the planned permanent cessation of operations of IP2 and IP3 and permanent defueling of the reactor vessels. The approved changes revised the IPEC emergency response organization (ERO) and augmented ERO staffing commensurate with the reduced spectrum of credible accidents for a permanently shutdown and defueled nuclear power reactor facility as each reactor (IP2 and IP3) is shutdown and permanently defueled. The PSEP currently maintains the effectiveness of the IPEC Emergency Plan in accordance with 10 CFR 50.47 and 10 CFR 50, Appendix E.

By letters dated May 12, 2020, and May 11, 2021, (References 6 and 7, respectively) Entergy certified to the NRC, in accordance with 10CFR 50.82(a)(1)(i), that power operations ceased at IP2 on April 30, 2020, and at IP3 on April 30, 2021. In addition, Entergy certified in accordance with 10 CFR 50.82(a)(1)(ii), that the fuel was permanently removed from the IP2 reactor vessel and placed in the IP2 SFP on May 12, 2020, and that the fuel was permanently removed from the IP3 reactor vessel and placed in the IP3 SFP on May 11, 2021. HDI understands and acknowledges that upon docketing of these certifications, the 10 CFR Part 50 license(s) no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel(s).

With the IP1, IP2, and IP3 reactors in the conditions described above, the reactors will never again enter any operational mode and reactor-related accidents, abnormal operational transients, and special events are no longer a possibility. The spectrum of credible accidents is much smaller than for an operational plant and the number and severity of potential radiological accidents are significantly less than when the IPEC reactors were operating. The majority of the accident scenarios postulated with the reactors in operation will no longer be applicable. Section 6, “ of the IP2 and IP3 Defueled Safety Analysis Reports (DSARs) describe the design basis accident (DBA) scenarios that are applicable to IP2 and IP3, respectively. The analyzed DBAs that remain applicable to IP2 and IP3 in the permanently shutdown and defueled condition are the fuel handling accident (FHA) in the FHBs (i.e., Fuel Storage Buildings (FSBs), accidental release of waste gas, and an accidental release-recycle of waste liquid. As previously discussed, there are no DBAs that remain applicable to IP1 with all remaining spent fuel stored at the ISFSI.

The offsite radiological consequences of accidents possible at IPEC will be substantially lower than during plant operation. 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 8) beyond the site boundary.

3.0 BASIS FOR EXEMPTION REQUEST

To allow for a reduction in emergency planning requirements commensurate with the hazards associated with IPEC's permanently shutdown and 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.

Based on its current configuration and licensing basis, with no spent fuel stored in the IP1 SFP, there are no postulated DBAs that remain applicable to IP1. The IP1 SFP is no longer in use because all spent fuel has been transferred offsite or to the ISFSI and other material removed, and the IP1 SFP has been drained. HDI operates the IPEC onsite ISFSI in accordance with 10 CFR Part 72, Subpart K, under the general license authorized by 10 CFR 72.210. The regulatory requirements for an ISFSI emergency plan are specified in 10 CFR 72.32. In accordance with 10 CFR 72.32(c)(1), the emergency plan required by 10 CFR 50.47 satisfies the requirements for an emergency plan for an ISFSI which is located onsite, and a separate ISFSI emergency plan is not required.

Operation of the ISFSI is adequately addressed for IP1, IP2, and IP3, and will continue to be addressed in the IPEC Emergency Plan. Those portions of 10 CFR 50.47(b) and (c)(2) and Appendix E to 10 CFR Part 50 from which exemptions are being requested, are no longer applicable to IP1. Therefore, additional technical justification for IP1 is not necessary considering the existing inherent non-applicability of the requested exemptions. The remainder of this exemption request provides the justification and underlying technical bases, specific to IP2 and IP3, to request exemptions from portions of 10 CFR 50.47(b); 10 CFR 50.47(c)(2); and 10 CFR Part 50, Appendix E.

HDI contracted Holtec to perform site specific analyses for the IP2 and IP3 SFPs for beyond design basis events. The site specific analyses demonstrate that 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. These analyses have been performed using the Holtec Spent Fuel Pool Heat Up Calculation Methodology submitted to NRC in Reference 9 as amended by References 10, 11, and 12, and have been submitted to NRC as a supplement to the HDI Request for Exemptions (Reference 13). NRC has issued a Draft Safety Evaluation Report documenting their review to date of the methodology (Reference 14). NRC has conducted an ACRS subcommittee meeting and a full committee meeting for review of the Topical Report is scheduled in February 2022. Holtec anticipates NRC issuance of the Final Safety Evaluation Report in March 2022.

Based on the results of the site specific analyses, in the unlikely event of a beyond design basis event, a minimum of 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.

The length of time it takes to raise the fuel temperature to 900 degrees °C (10 hours or greater) is ample time for IPEC to respond to any draindown event by restoring cooling or makeup or providing spray to the IP2 or IP3 SFPs. 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, HDI has 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, will not be necessary to protect the health and safety of the public and continued applicability of the regulations would be unduly burdensome when the IPEC units are all in the permanently shutdown and defueled condition and there is at least 10 hours to respond to a beyond design basis event resulting in the drain down of the SFP to the point that the cooling is not effective. Approval of the exemptions requested in Tables 1 and 2 would not present an undue risk to the public or prevent an appropriate response in the event of an emergency at IPEC.

HDI plans to submit a Permanently Defueled Emergency Plan (PDEP), including a Permanently Defueled Emergency Action Level (EAL) scheme, for NRC review and approval pursuant to 10 CFR 50.54(q)(4) and 10 CFR 50, Appendix E, Section IV.B.2. The proposed Emergency Plan will be based on the exemptions requested herein.

4.0 REQUESTED EXEMPTIONS FROM EMERGENCY PLANNING REQUIREMENTS DEFINED BY 10 CFR 50.47 AND 10 CFR PART 50, APPENDIX E

HDI requests exemptions from portions of 10 CFR 50.47(b) and (c)(2) and Appendix E to 10 CFR Part 50 to the extent that these regulations apply to specific provisions of onsite and offsite emergency planning that will no longer be applicable once the certifications required by 10 CFR 50.82(a)(1)(i) and (ii) have been submitted to the NRC and sufficient decay of the IP2 andIP3 spent fuel has 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 exemption is not being requested), will remain applicable to IPEC. Details related to specific exemption requests are provided in the Basis for Exemption column in each table.

The requested exemptions and 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 50.47(c)(2)

Item #	10 CFR 50.47, Emergency Plans	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 15), 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 similar to an ISFSI or MRS than an operating nuclear power plant. The EP program would be similar to 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. The draft proposed rulemaking in SECY-00-0145 (Reference 16) 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(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. Because of the slow rate of the event scenarios in the postulated accident and postulated beyond-design-basis events analyses, time is available to complete actions necessary to mitigate an emergency

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
		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. Exemptions from offsite emergency planning requirements have previously been approved when the site-specific analyses show that at least 10 hours is available from a partial draindown event where cooling of the spent fuel is not effective until the hottest fuel assembly reaches 900°C. The technical basis that underlies the approval of the exemption request is based partly on the analysis of a time period that spent fuel stored in the SFP is unlikely to reach the zirconium ignition temperature in less than 10 hours. This time 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 an 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. HDI has performed analyses demonstrating that 30 days after permanent cessation of power operations of each unit, the radiological consequences of the analyzed DBAs that remain applicable to IP2 andIP3 will not exceed the limits of the U.S. Environmental Protection Agency's (EPA) Protective Action Guides (PAGs) at the Exclusion Area

Table 1

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

Item #	10 CFR 50.47, Emergency Plans	Basis for Exemption
		Boundary (EAB). The radiological consequences of the remaining applicable DBAs are discussed in Section 5.2 of this Enclosure. HDI has performed site specific analyses for the IP2 and IP3 SFPs for beyond design basis events, which demonstrate that 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. These analyses are provided in Enclosure 1 to Reference 13. IPEC maintains procedures and strategies to mitigate events involving a loss of SFP cooling and/or water inventory required under the provisions of 10 CFR 50.155(b)(2) (formerly 10 CFR 50.54(hh)(2). These strategies are maintained to satisfy applicable portions of License Condition 2.N of the IP2 Renewed Facility License (FL) and License Condition 2.AC of the IP3 Renewed Facility Operating License (FOL) and provide defense-indepth and ample time to provide makeup water or spray to the SFPs prior to the onset of zirconium cladding ignition when considering very low probability of beyond design basis events affecting the SFPs. Two (2) trained on-shift individuals at IP2 and IP3 can implement necessary actions to supply makeup water to either SFP within to (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 task that would inhibit timely performance. Direction and selection of the tasks related to adding makeup water to the IP2 and IP3 SFPs will continue to be the responsibility of the Emergency Director.

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
		Training of the IP2 and IP3 on-shift staff will be maintained, and they will implement such strategies and plans to mitigate the consequences of an event involving a catastrophic loss-of-water inventory from the SFP.
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 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 IP2 and IP3 are in the permanently shutdown and defueled condition. The IPEC Emergency Plan will 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 protective measures, and significant time is available to take mitigative or, if

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
		needed, offsite protective actions using a CEMP between the initiating event and before the onset of a postulated zirconium fire. Therefore, an Emergency Operations Facility (EOF) is not required. The IP2 and IP3 Control Rooms, or another onsite 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 medical (ambulance and hospital) services, as appropriate. Also, refer to the basis for 10 CFR 50.47(b) (Item 1 in Table 1).
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~~.	IPEC will adopt a Permanently Defueled EAL scheme consistent with the guidance provided in Appendix C of Nuclear Energy Institute (NEI) 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6 (Reference 17), endorsed by the NRC in a letter dated March 28, 2013 (Reference 18). HDI has performed site specific analyses for the IP2 and IP3 SFPs for beyond design basis events, which demonstrate that 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. These analyses are provided in Reference 13. Based on the results of the site-specific analyses, in the unlikely event of a beyond design basis event, a minimum of 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 before the hottest fuel assembly reaches the zirconium fire

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
		temperature. 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) (Item 1 in Table 1).
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~~.	As described in SECY-00-0145 (Reference 16), after approximately one (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 Water Reactor," (Reference 19) 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 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 10 CFR 50.47(b) (Item 1 in Table 1).

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
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 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 andmaintained.	No exemption is requested.
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 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~~	Site specific 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 8) beyond the site boundary. 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.

Table 1 Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
	~~actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~	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 Statements of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and MRS facilities (60 FR 32435) (Reference 15), 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 Statements of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 15), 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: "The 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 exemption from offsite emergency planning requirements as discussed in the exemption from 10 CFR 50.47(b) (Item 1 of Table 1), evacuation time estimates are also not needed. Also, refer to the basis 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 Appendix E, Section IV.1 (Item 2 in Table 2), for a 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	No exemption is requested.

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
	include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.	
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.
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~~	Current 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 isbased on the maximum distance at which a PAG might be exceeded (Reference 8).

Table 1 **Exemptions Requested from 10 CFR 50.47(b) and 50.47(c)(2)**		
Item #	**10 CFR 50.47, Emergency Plans**	**Basis for Exemption**
	~~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.~~	Also, refer to the basis for 10 CFR 50.47(b) (Item 1 in Table 1).

<u>Table 2</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	10 CFR Part 50, Appendix E	Basis for Exemption
1	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.
2	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,	Following docketing of the certifications of permanent cessation of operations and permanent removal of fuel from the reactor vessels for IP2 and IP3, in accordance with 10 CFR 50.82(a)(1)(i) and (ii), IPEC has become a permanently shutdown facility with spent fuel stored in the IP2 and IP3 SFPs and the ISFSI. In the EP Final Rule (76 FR 72560, Nov. 23, 2011) (Reference 20), the NRC defined "hostile action" as, in part, an

<u>**Table 2**</u>

Exemptions Requested from 10 CFR 50, Appendix E

Item #	10 CFR Part 50, Appendix E	Basis for Exemption
	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.	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, "Emergency Preparedness and Response Actions for Security- Based Events," dated July 18, 2005 (Reference 21). 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 the definition. Similarly, a decommissioning power reactor or ISFSI is not a "nuclear reactor" as defined in the NRC's regulations. A decommissioning power reactor also has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For all of these reasons, the NRC staff has concluded that a decommissioning power reactor is not a facility that falls within the definition of "hostile action." Although, the analysis described above and in the basis for 10 CFR Part 50, Appendix E, Section IV.1, provides a justification for exempting IPEC 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. The following similarities between IPEC and NPRs show that the IPEC facility should be treated in a similar fashion as an NPR. Similar to NPRs, IPEC poses lower radiological risks to the public from accidents than do power reactors because: (1) IP1, IP2 and IP3 are permanently shutdown and IPEC is a permanently

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	10 CFR Part 50, Appendix E	Basis for Exemption
		shutdown facility (with fuel stored in the IP2 and IP3 SFPs and on the IPEC ISFSI) and the site no longer generates fission products; 2) upon transition to the post-Zirconium fire window, fuel stored in the IP2 and IP3 SFPs will have lower decay heat resulting in lower risk of fission product release in the event of a beyond design basis boil-off or draindown event; and 3) no credible accident at IPEC will result in radiological releases requiring offsite protective actions.
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 licenseesshall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submitthis ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State~~	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 #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	~~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 availablefor 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 laterthan 365 days after the licensee's determination that the criteria forupdating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it 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).

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
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 laterthan 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 IPEC 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 notification of such individuals in the event of an emergency. Specifically, the following shall be included:	No exemption is requested.
10	A.1. A description of the normal plant ~~operating~~ organization.	Following the docketing of the certifications for IP2 and IP3 required by 10 CFR 50.82(a)(1)(i) and (ii), IPEC is no longer a facility that can be operated to generate electrical power. Therefore, IPEC does not have a

Table 2

Exemptions Requested from 10 CFR 50, Appendix E

Item #	10 CFR Part 50, Appendix E	Basis for Exemption
		"plant operating organization." Rather, the site is maintained by an on-shift staff responsible for safely managing and storing spent fuel.
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 IPEC during the decommissioning process is 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 at IPEC 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).
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	HDI has developed an analysis indicating that 15 months after permanent cessation of power at IP3, no credible or beyond design basis accident at IPEC will result in radiological releases requiring offsite protective actions. HDI will maintain the capability at IPEC 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.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	local authorities, NRC, and other appropriate governmental entities.	Also, refer to the basis 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 site specific spent fuel pool heatup analyses, the time available to initiate compensatory actions in the event of a loss of SFP inventory precludes the need to identify and describe the special qualifications of these individuals in the emergency plan. The number of staff at IPEC with IP1, IP2, and IP3 permanently shutdown and defueled is small but is 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 tasks within 2 hours.
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 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 forceto 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.	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 exemption from 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

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
		concern that a sabotage attack could cause offsite radiological consequences is removed. Although, the analysis described 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 IPEC 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 organizations (i.e., local law enforcement, firefighting, and medical assistance) under a CEMP concept will still be required. HDI will maintain appropriate actions for the protection of IPEC onsite personnel in a security-based event. The scope of protective actions will be appropriate for the defueled plant status of the IPEC facility 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 Indian Point Physical Security, Training and Qualification, Safeguard Contingency Plan (IPEC 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, notification of offsite authorities, and coordination with offsite response organizations (i.e., local law enforcement, firefighting, medical assistance) under a CEMP concept.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
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 medical (ambulance and hospital) services, as appropriate. Due to the low probability of DBAs or other credible events to exceed the EPA PAGs, protective actions such as evacuation should not be required, but could be implemented at the discretion of offsite authorities using a CEMP. 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 of the on-shift and emergency response personnel will be detailed in the PDEP and implementing procedures and will be tested through drills and exercises. 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 (76 FR 72560, Nov. 23, 2011) (Reference 20, 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 SFPs or in a dry cask storage system in a safe condition requires minimal personnel and is governed by Technical Specifications. Because of the slow rate of the event scenarios in the postulated DBAs and postulated beyond design basis events analyses, 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 nuclear power plant

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
		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 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.	Because of the geographic location of IPEC, annual review of EALs have historically involved the State of New York and Westchester, Rockland, Putnam, and Orange Counties. 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 New York and Westchester and Rockland Counties. However, based upon the reduced scope of EALs for the permanently defueled plants at the IPEC facility, the scope of the annual review of EALs is expected to be limited (i.e., informal mailings, etc.). IPEC will develop EALs consistent with the guidance on Permanently Defueled EALs detailed in Appendix C of NEI 99-01, Revision 6 (Reference 17). Also, refer to the basis for 10 CFR Part 50, Appendix E, Section IV.1 (Item 2 of Table 2), for the justification from the regarding "hostile action."
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.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
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 emergency~~. These classes are further discussed in NUREG–0654/FEMA–REP–1.	The Permanently Defueled EALs, developed consistent with the guidance provided in Appendix C of NEI 99-01, Revision 6 (Reference 17), 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 is no longer appropriate because these parameters do not provide indication of the conditions at a defueled plant, and emergency core cooling systems are no longer required. Other indications, such as SFP level or temperature, can be used at sites where there is spent fuel in the SFPs. The guidance presented in NEI 99-01, Revision 6, was endorsed by the NRC in a letter dated March 28, 2013 (Reference 18). 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 permanently shutdown and defueled plant/facility that meets the conditions for relaxation of emergency planning requirements, there will be available time for event mitigation, and if necessary, implementation of offsite protective actions using a comprehensive approach to emergency planning. Also, refer to the basis 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. In the Statement of Considerations for the Final Rule for Emergency Planning requirements for ISFSIs and for MRS facilities (60 FR 32430) (Reference 15), 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 offsite for more than the immediate site area." The

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
		probability of a condition reaching the level above an emergency classification of Alert is very low. In the event of an accident at a permanently shutdown and defueled plant/facility that meets the conditions for relaxation of emergency planning requirements, there will be time to initiate mitigative actions consistent with plant conditions, and if necessary, for offsite authorities to employ their CEMP to take protective actions. As stated in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (February 2001) (Reference 22), 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 permanently shutdown and 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 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	In the Statement of Consideration for the Emergency Preparedness Rule (74 FR 23254) (Reference 23) to amend certain emergency planning requirements for 10 CFR Part 50, the NRC reviewed the need for additional requirements for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. The NRC recognized a decommissioned power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures and determined non-power reactor licensees should not be required to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition.

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.	HDI proposes to maintain the capability to assess, classify, and declare an emergency condition within 30 minutes. Emergency declaration is required to be made as soon as conditions warranting classification are present and recognizable, but within 30 minutes after the availability of indications to operations staff that an EAL threshold has been reached. With the IPEC units in a permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor power operation are no longer credible. The consequences resulting from the 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 emergency response organizations, and no objections to the proposed changes were received. Also, refer to the basis 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 10 CFR Part 50, Appendix E, Section IV.1, discussion (Item 2 in Table 2) on the similarities between a permanently shutdown and defueled reactor facility and an NPR.
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~~	Refer to the basis 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 #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	~~agency,~~ of the State and local government agencies ~~within theEPZs~~.	
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 ifan 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 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 make a public alerting and notification decision promptly on being informedby the licensee of an emergency condition. Prior to initial operationgreater than 5 percent of rated thermal power of the first reactor ata 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~~	HDI proposes to complete IPEC emergency notifications to appropriate State and local government agencies within 60 minutes 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 and is appropriate because with the IPEC units in a permanently shutdown and 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 PDEP will address the primary and backup means for conducting the required notifications to the appropriate State and local government agencies. Because of the geographic location of IPEC, emergency planning responsibilities have historically involved coordination with the State of New York; Westchester, Rockland, Putnam, and Orange Counties; and

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	~~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 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.~~	the City of Peekskill. The decommissioning-related emergency plan submittals for IPEC 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 pertaining to those agencies that are provided emergency notifications, the 30-minute 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 emergency 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).
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~~	Refer to the basis for Section IV.D.3 (Item 25 in Table 2) regarding the Alert and Notification System requirements.

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	~~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.~~	
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.
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.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
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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;	Due to the low probability of DBAs or other credible events to exceed the EPA PAGs, the significantly reduced staff and the minimal expected offsite response required, offsite agency response will not be required at an Emergency Operations Facility (EOF) and onsite actions may be directed from the Control Room or another 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 IPEC Emergency Plan will continue to maintain arrangements for requesting assistance and using resources from appropriate offsite support organizations. Also, refer to the basis 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 24) 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 facility with permanently shutdown and defueled plants, an OSC is no

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
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		longer required to meet its original purpose of an assembly area for plant logistical support during an emergency. Onsite capabilities will continue to be maintained at IP2 and IP3, from which Control Room support, emergency mitigation, radiation monitoring, and effective control may be exercised during an emergency.
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 25miles 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 NRCand offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:~~	In accordance with paragraph E.8.e, the requirements of paragraph E.8.b.(1) – (5) do not apply to the IPEC EOF because it was an approved facility prior to December 23, 2011. However, the exemption is requested to clearly reflect that the requirement no longer applies to the IPEC facility with the units in a permanently shut down and defueled condition. Also, refer to the basis 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~~	

Table 2 Exemptions Requested from 10 CFR 50, Appendix E		
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38	E.8.b.(2) ~~Additional space for conducting briefings with emergency response personnel;~~	
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 10 CFR 50.47(b)(3) (Item 4 in Table 1) and Appendix E to 10 CFR Part 50, 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**		
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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 forcommunication 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 Appendix E to 10 CFR Part 50, 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 10 CFR 50.47(b)(3) (Item 4 in Table 1) and Appendix E to 10 CFR Part 50, Section IV.E.8.b (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	Refer to the basis for 10 CFR 50.47(b) (Item 1 in Table 1) and (b)(10)(Item 11 in Table 1). HDI will maintain IPEC primary and backup communications with the contiguous State and local governments; specifically, the State of New York and Westchester and Rockland Counties. Because EPZs would be eliminated, the IPEC PDEP would no longer describe provisions to

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	communication. Where 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.	communicate with Putnam and Orange Counties, and the City of Peekskill. The onsite response facilities will be combined into a single facility, as described in the basis for Appendix E to 10 CFR Part 50, Section IV.E.8.a(i) (Item 34 in Table 2). A description of the communications systems and the testing frequencies will be included in the PDEP.
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.~~	Because of the low probability of DBAs or other credible events that would be expected 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, licensees that meet the criteria for exemptions from offsite emergency planning requirements do not need the TSC, EOF, or offsite field assessment teams. Therefore, there is no need for IPEC to maintain the TSC, EOF, or field assessment teams. Additionally, there is no need to maintain and test committed provisions for communications with State and local emergency operations centers (EOCs) with these facilities. Also, refer to the basis 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 IP2 and IP3, from which effective command and control can be maintained during an emergency. Communication with State and local EOCs is maintained to coordinate onsite assistance, if required. The provisions remaining in

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Item #	10 CFR Part 50, Appendix E	Basis for Exemption
		10 CFR Part 50, Appendix E, Section IV.E.9.a, b, and d include the necessary requirements for communication 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 will be combined into one or more locations due to the smaller facility staff and the greatly reduced interaction required with State and local emergency response facilities. Onsite facilities will continue to be maintained for IP2and IP3, from which effective command and control will be maintained and direction can be given during an emergency. HDI will maintain the capability for IPEC to communicate with the NRC. Also, refer to the basis 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;	
53	F.1.ii. Personnel responsible for accident assessment, including control room shift personnel;	

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Item #	10 CFR Part 50, Appendix E	Basis for Exemption
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 IPEC facility staff during the decommissioning process is 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 at IPEC 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. Therefore, exempting licensee's headquarters personnel from training requirements is reasonable. Also, refer to the basis for 10 CFR 50.47(b) (Item 1 in Table 1).
60	F.1.ix. Security personnel.	No exemption is requested.
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~~.	Due to 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

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		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 andnotification system,~~ and ensure that emergency organization personnel are familiar with their duties.[3]	There is low probability of DBAs or other credible events that would be expected to exceed the limits of 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. As such, 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 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 New York and Westchester and Rockland Counties to participate in the IPEC periodic drills and exercises conducted to assess their ability to perform responsibilities related to an emergency at IPEC, to the extent defined by the IPEC Emergency Plan. Because the need for offsite emergency planning is relaxed due to the low probability of the DBAs or other credible events that would be expected to result in an offsite radioactive release that would exceed the EPA PAGs and the available time for event mitigation, no offsite emergency plans will be in place to test. 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 defueled reactor sites, there are limited events that could occur and the previously routine
64	F.2.a.(i) ~~For an operating license issued under this part, this exercise must be conducted within two years before the 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~~	

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	~~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.~~	progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning facility. HDI considers IPEC to be exempt from 10 CFR Part 50, Appendix E, Section F.2.a.(i) - (iii), because IPEC will be exempt from the umbrella provision of 10 CFR Part 50, Appendix E, Section IV.F.2.a (Item 63 in 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 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~~	

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	~~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, 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	Refer to the basis for Appendix E to 10 CFR Part 50, Section IV.F.2.a(Item 63 in Table 2). The low probability of DBAs or other credible events that would result inan offsite radioactive release that would exceed the EPA PAGs and the available time for event mitigation at IPEC 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 IPEC will be combined, such that there will be a single unit-specific facility for IP2 and IP3. HDI will continue to conduct biennial exercises at IPEC and will invite the State of New York, Westchester County, Rockland 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, to the extent defined by the IPEC Emergency Plan.

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	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 basis for Appendix E to 10 CFR Part 50, 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;~~	
71	F.2.c.(3) ~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite fullor 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;~~	

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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 10 CFR Part 50.47(b)(10) (Item 11 in Table 1).
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 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	FEMA is currently responsible for the evaluation of an offsite response exercise. No action is expected from State or local government organizations in response to an event at a decommissioning facility 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

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	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.~~	protect the health and safety of the public using a CEMP approach as they would at any other industrial facility. 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 the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.	No exemption is requested.
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.	At IPEC, 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, HDI does not expect to demonstrate IPEC response to a wide spectrum of events. Also, refer to the basis 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

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
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		Part 50, Appendix E, Section IV.1 (Item 2 in Table 2), regarding "hostile action."
80	F.2.j. ~~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. 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. Additionally, in each eight calendar year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for theERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements: hostile action directed at the plant site, no radiological release or an unplanned minimal radiological release that does not require public protective actions, an initial classification of or rapid escalation to a Site Area Emergency or General Emergency, implementation of strategies, procedures, and guidance developed under § 50.54(hh)(2), and integration of offsite resources with onsite response. The licensee shall maintain a record of exercises conducted during each eight year exercise cycle that documents the content of scenarios used to comply with the requirements of this paragraph. Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015. The first eight-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 Part 52, the first~~	Refer to the basis for 10 CFR Part 50, Appendix E, Section IV.F.2. Also, refer to the basis 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 10CFR Part 50, Appendix E, Section IV.1 (Item 2 in Table 2), regarding 'hostile action."

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	~~eight-year exercise cycle begins in the calendar year of the initial exercise required by Section IV.F.2.a.~~	
81	G. Maintaining Emergency Preparedness 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.	No exemption is requested.
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 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	No exemption is requested.

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Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	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.	
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	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 IPEC units all permanently shutdown and defueled, this system is no longer be necessary to transmit safety system parameter data. No exemption is requested because this change in the ERDS data requirements is identified in 10 CFR Part 50, Appendix E, Section IV.2.

Table 2 **Exemptions Requested from 10 CFR 50, Appendix E**		
Item #	**10 CFR Part 50, Appendix E**	**Basis for Exemption**
	NRC, will assemble the data to be transmitted and transmit data from each unit via an output port on the appropriate data system.	
86	10 CFR 50 Appendix E Footnotes 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 sufficientto 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 IPEC to be exempt from Footnotes 4, 5, and 6 because IPEC 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 #	**10 CFR 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

As specified in 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses no longer authorize operation of the IPEC reactors or emplacement or retention of fuel in the reactor vessels 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) (References 6 and 7). With the termination of power operations at IP1, IP2, and IP3, and the permanent removal of the fuel from the reactor vessels, the postulated accidents involving failure or malfunction of the reactor and supporting structures, systems, and components (SSCs) are no longer applicable. A summary of the radiological accidents analyzed for the permanently shutdown and defueled condition is presented below.

Based on its current configuration and licensing basis, with no spent fuel stored in the IP1 SFP, there are no postulated DBAs that remain applicable to IP1. The IP1 SFP is no longer in use because all spent fuel has been removed from the site or transferred to the ISFSI and other material removed, and the IP1 SFP has been drained. HDI operates the IPEC onsite ISFSI in accordance with 10 CFR Part 72, Subpart K, under the general license authorized by 10 CFR 72.210. Therefore, operation of the ISFSI is adequately addressed for IP1, IP2, and IP3, and will continue to be addressed in the IPEC Emergency Plan. Accordingly, the analyses discussed within this section only address the risks associated with the storage of spent fuel in the IP2 and IP3 SFPs.

Section 5.0 of ISG-02 indicates that site-specific analyses should demonstrate that: (1) the radiological consequences of the 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 from a loss of SFP cooling, there is sufficient time to implement pre-planned mitigation measures to provide makeup or spray to the SFP before the onset of zirconium cladding ignition.

Table 3 contains a listing of seven analyses described in ISG-02, that are expected to be evaluated by a decommissioning power reactor licensee requesting exemption of emergency planning requirements. The Table also contains a description of how HDI addresses each of these analyses.

Table 3 **Interim Staff Guidance-02 Comparison**		
ISG-02 Analysis	**ISG-02 Description**	**HDI 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 will remain applicable to IP2 and IP3 and could contribute to dose upon implementation of the requested exemptions are FHAs in the IP2 and IP3 FSBs, accidental release of waste gas, and an accidental release-recycle of waste liquid. The results of these analyses indicate that the dose at the EAB would not exceed the EPA PAG criterion of 1 rem Total Effective Dose Equivalent (TEDE) aftera 30-day fuel decay period following permanent cessation of power operations, and the results apply to both IP2 and IP3 since both units have been permanently shutdown for more than 30 days (Reference 25). 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);	HDI has performed site specific analyses for the IP2 and IP3 SFPs for beyond design basis events, which demonstrate that 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. Based on the results of the bounding analysis summarized in Reference 13, in the unlikely event of a beyond design basis event, a minimum of 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 thehealth 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 Enclosure 1 of the HDI Supplement (Reference 13.) NRC has issued a DSER based on their review of the Holtec Topical Report (Reference 14)
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	HDI performed a bounding analysis of the IP2 and IP3 SFPs to determine the radiological impacts of a complete loss of SFP water (Reference 26). It was determined that at one year after permanent cessation of power, the gamma radiation dose rates at the EAB and the IP2 and IP3 Control Rooms for each unit would be less than the regulatory defined limits. This analysis is described in Section 5.3.2 of this Enclosure.
4	Considering the site-specific seismic hazard, either an evaluation demonstrating a 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	HDI developed an analysis (Reference 27) demonstrating successful completion of the Enhanced Seismic Checklist provided in Attachment 1 to Appendix 2B of NUREG-1738 (Reference 22) for the IP2 and IP3 SFPs demonstrating a high confidence of a low-probability (less than 1 x 10^{-5} per year) of seismic failure of the SFP structures. This analysis is described in Section 5.4 and is summarized in Table 6 of this Enclosure.

	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).	
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 for IP2 and IP3. The IDCs and SDAs are addressed in Section 5.4 and Tables 4 and 5 of this Enclosure.
6	Verify that the licensee presents a determination that there are 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 IP2 and IP3 SFPs are incorporated into IPEC procedures and utilize adequately trained on-shift resources for implementation. There are multiple ways to initiate mitigative actions and add makeup water to the IP2 and IP3 SFPs within the 10-hour minimum time period with or without entry to the SFP floors. Additionally, although the number of facility staff at IPEC is small since IP2 and IP3 are permanently shutdown, the staffing level is commensurate with the need to operate the facility in a manner that is protective of public health and safety. 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 IPEC 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 2.N of the IP2 Facility License and License Condition 2.AC of the IP3 Facility License. These diverse strategies provide defense-in-depth and ample time to provide makeup water or spray to the IP2 and IP3 SFPs prior to the onset of zirconium cladding ignition when considering very low probability beyond design basis events affecting the SFPs. Refer to SDA-4 in Table 5 of this Enclosure.

5.2 Consequences of Design Basis Events

The NRC approved the IP2 Permanently Defueled Technical Specifications (PDTS) on April 28, 2020, with the issuance of IP2 License Amendment No. 294 (Reference 28). The license amendment included the statement that the applicable DBAs for IP2 in the permanently defueled condition are: (1) an FHA in the FHB, (2) an accidental release of waste gas, and (3) an accidental release-recycle of waste liquid. Similarly, NRC approved the IP3 Permanently Defueled Technical Specifications on April 22, 2021, with the issuance of IP3 License Amendment No. 270 (Reference 29) reflecting the permanently shutdown and defueled condition. The IP3 amendment includes the statement that the applicable DBAs for IP3 in the permanently defueled condition are: (1) the FHA in the FHB, (2) an accidental release of waste gas, and (3) an accidental release-recycle of waste liquid.

The DBAs that remain applicable to IP2 and IP3 are discussed in the following paragraphs in this Section.

<u>FHA Analysis</u>

An FHA may occur in the FSB during movement of a fuel assembly. The fuel assembly is moved under water and the accident is assumed to occur when one fuel assembly is damaged. The IP2 and IP3 post-permanent shutdown FHA (Reference 25) was evaluated utilizing the Alternate Source Term (AST) methodology described in Regulatory Guide 1.183 (Reference 30). This analysis did not credit the function of FSB filtration, high-rad alarm, dispersion from the FSB ventilation system, Control Room isolation, or emergency filtration. The analysis credits the decontamination of the 23 feet of water over the fuel assemblies in the SFP with an overall effective decontamination factor of 200, consistent with Regulatory Guide 1.183 (Reference 30).

The analysis indicates that after a decay time of at least 720 hours (30 days) following permanent cessation of power operations of each unit, the FHA results in an EAB TEDE dose of 0.47 rem (Reference 25), which is below the EPA early phase PAG criteria of 1 rem TEDE. In addition, the NRC has previously noted that the doses from an FHA are dominated by the isotope Iodine-131.

<u>Accidental Release of Waste Gas</u>

This calculation includes the determination of the dose consequences for a waste gas decay tank rupture accident using a 50,000 curie (Ci) dose-equivalent Xe-133 waste gas tank activity limit without any credit for mitigating systems. The waste gas decay tanks receive the radioactive gases from the radioactive liquids from the various laboratories and drains processed by the waste disposal system. The 50,000 Ci dose-equivalent Xe-133 waste gas tank activity assumed in this calculation bounds the current Xe-133 dose-equivalent limit of 29,761 Ci, as well as the administrative Xe-133 dose-equivalent limit of 6,000 Ci (Reference 25).

Other tanks that contain waste gas during operations (the volume control tank and liquid holdup tank) were not considered in this analysis, since gaseous products from these liquid tanks are collected and compressed in the waste gas decay tanks for decay prior to release. Potential liquid waste releases are considered from these tanks; however, any liquid releases are retained in the building or sumps and only volatilized components would be released to the environment. These volatilized components are evaluated as part of the waste gas decay tank accident.

With the IP2 and IP3 reactors permanently shutdown and defueled, there is no mechanism to raise the primary coolant activity. Therefore, the source term initially contained within the waste gas tanks represents the worst-case source term, which is less than the assumed waste gas tank rupture analysis of record and is thus bounded. Subsequent additions to the waste gas tanks resulting from water management issues were less than the final shutdown and cooldown values.

The analysis concludes that without crediting any mitigating systems or the Plant Auxiliary Building(PAB) ventilation system, the calculated TEDE to the Control Room is less than the limit set forth in10 CFR 50.67 and the whole-body dose value of 500 millirem (mrem) at the EAB. The dose consequences from a waste gas tank decay tank rupture are less than the dose consequences following an FHA and meet the applicable radiological dose criteria at the Control Room, EAB, and Low Population Zone (LPZ) (Reference 25).

Accidental Release-Recycle of Waste Liquid

Section 6.2 of the IP2 Defueled Safety Analysis Report (DSAR) and Section 6.4 of the IP3 DSAR address the accidental release of waste liquid. In both documents, the referenced sections state that the hazard from these releases is derived only from the volatized components. Thus, the release of liquid waste is evaluated in the accidental release of waste gas, which is addressed above.

5.3 Consequences of Beyond Design Basis Events

5.3.1 Spent Fuel Assembly Heat Up During a Theoretical Drain Down Event

The analyses, provided in Enclosure1 to Reference 13, compare the heat load limits for the hottest fuel assembly and for a 2X2 group of assemblies stored in each SFP (IP2 and IP3) to a criterion proposed in Commission Paper SECY-99-168, “Improving Decommissioning Regulations for Nuclear Power Plants," (Reference 31) 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 heatup time of 10 hours from the time the spent fuel is uncovered, was determined to be sufficient to take mitigating actions and, if necessary, offsite protective measures without offsite emergency preplanning addressing the facility.

The bounding analyses for the IP2 SFPs for beyond design basis events demonstrate that 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. The bounding analyses for the IP3 SFPs for beyond design basis events demonstrate that 15 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,"Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," (February 2001) (Reference 22) 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 bounding 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 length of time it would take for the fuel to heatup, there is ample time to respond to any draindown event that might cause such an occurrence by restoring cooling or makeup or providing spray to the IP2 or IP3 SFPs. As a result, the likelihood that such a scenario would progress to a zirconium fire is deemed not credible.

5.3.2 Spent Fuel Pool Draindown Event

NRC NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (Reference 32) Supplement 1, Section 4.3.9, identifies that a SFP draindown event is a beyond design basis event. The analyses discussed in Section 5.3.1 demonstrate that a significant release of radioactive material from the SFP is not possible within 10 hours from the time the spent fuel in either the IP2 or IP3 SFP is uncovered. However, the potential exists for radiation exposure if shielding of the fuel in the IP2 or IP3 SFP is lost.

HDI analyzed the bounding radiological consequences of a postulated complete loss of SFP water from the IP2 and IP3 SFPs as a function of time after shutdown of IP2 and IP3. The analysis considered limiting distances from both SFPs to both Control Rooms and the EAB and a combination of IP3 fuel in the IP2 SFP, to bound both units (Reference 26).

The SFP water and the concrete SFP structures 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 SFP and being scattered back to a receptor at the site boundary. The analysis determined that the gamma radiation dose rates at the EAB from a loss of water shielding at the IP2 or IP3 SFP would be less than the EPA PAGs (Reference 26).

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 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 preplanning if efforts to reestablish shielding over the fuel are delayed.

Based on the analysis, the dose rate to a receptor at the EAB and the limiting dose rate in the IP2 and IP3 Control Rooms at one year after shutdown are less than 11.55 mrem/hr and 0.0259 mrem/hr, respectively (Reference 26).

5.4 Comparison to NUREG-1738 Industry Decommissioning Commitments and StaffDecommissioning Assumptions

Although the absence of DBAs applicable to IP1 and the limited scope of DBAs and beyond design basis accidents that remain applicable to IP2 and IP3 justify a reduction in the necessary scope of emergency response capabilities, HDI also evaluated the Industry Decommissioning Commitments (IDCs) and Staff Decommissioning Assumptions (SDAs) contained in NUREG-1738 (Reference 22).

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. 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 33). 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 (Reference22). Tables 4 and 5 of this Enclosure identify how the IP2 and IP3 SFPs meet or compare with each of these IDCs (Table 4) and SDAs (Table 5).

SDA #6 allows for the decommissioning plant to complete the seismic checklist provided in Appendix 2B to NUREG-1738 (Reference 22). The Enhanced Seismic Checklist includes elements to assure there are no weaknesses in the design or construction or any service-induced degradation of the SFPs that would make them vulnerable to failure during earthquake ground motions that exceed their design-basis ground motion but are less than the 1.2 g peak spectral acceleration.

An analysis for IP2 and IP3 SFPs has been developed (Reference 27) and demonstrates successful completion of the Enhanced Seismic Checklist provided in Attachment 1 to Appendix 2B of NUREG-1738 (Reference 22) . Successful completion of the checklist requires record reviews and walkdown inspections of the IP2 and IP3 SFPs. Maintenance Rule Walkdowns of the accessible areas of the IP2 and IP3 SFP structures and FSBs are performed every 5 years and those inspections are documented in engineering reports. Also, supplemental inspections of the condition of the SFP walls and reinforcement have been performed and documented as discussed in HDI's response to Checklist Item 1 in Table 6. In addition, walkdowns performed to meet NRC requirements in response to the events associated with the earthquake and subsequent tsunami that affected the Fukushima Daichi nuclear power plant verified that there were no vulnerabilities for a rapid draindown of the IP2 or IP3 SFPs as documented in Engineering Reports IP-RPT-12-00037, Rev. 1 (Reference 34) and IP-RPT-12-00039, Rev. 1 (Reference 35). These inspections meet the intent of the walkdown requirement specified in the Design Feature section of the Enhanced Seismic Checklist Items listed above and verify that there are no structural concerns with the IP2 or IP3 SFPs and no issues that could cause a rapid draindown of the SFPs.

A summary of the analysis demonstrating successful completion of the Enhanced Seismic Checklist for the IP2 and IP3 SFPs is provided in Table 6 to this Enclosure.

Based on the analysis summarized in Table 6, there is a high confidence in a low probability of failure (HCLPF) for seismic ground motions up to 1.2 g peak spectral acceleration (or with peak ground acceleration (PGA) of approximately 0.5 g), which in turn assures that the frequency of fuel uncovery from seismic events for IP2 and IP3 is less than or equal to $1x10^{-5}$ per year (Reference 27).

The IP1 SFP is not addressed in Tables 4, 5, and 6 because the IP1 SFP is no longer in use, all spent fuel has been removed from the site or transferred to the ISFSI, other material has been removed from the IP1 SFP, and the IP1 SFP has been drained. Therefore, there is no risk associated with the IP1 SFP.

5.5 Consequences of a Beyond-Design Basis Earthquake

NUREG-1738 (Reference 22) 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 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 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 8) at the EAB.

However, the risk associated with zirconium cladding fire events decreases as the spent fuel ages because, as the decay time increases, the generation of decay heat decreases. 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 and mitigative 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," dated September 2014 (Reference 36), suggests that, while other radiological consequences can be extensive, a postulated accident scenario leading to a zirconium fire, where the fuel has had significant decay time, will have little potential to cause offsite early fatalities due to dose, even if formal offsite radiological emergency preparedness was relaxed.

The purpose of NUREG-2161 (Reference 36) was to determine if accelerated transfer of older, colder spent fuel from the SFP at a reference plant to dry cask storage would significantly reduce 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 the spent fuel. The study shows the likelihood of a radiological release from the spent fuel 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 the individual 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 ($2x10^{-6}$ /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 shows 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 hig-hand 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 Section 5.4 of this Enclosure, an IPEC analysis was developed (Reference 27) and demonstrates successful completion of the Enhanced Seismic Checklist provided in Attachment 1 to Appendix 2B of NUREG-1738 (Reference 22) for the IP2 and IP3 SFPs. Based on the analysis summarized in Table 6, there is a HCLPF for seismic ground motions up to 1.2 g peak spectral acceleration (or with peak ground acceleration (PGA) of approximately 0.5 g), which in turn assures that the frequency of fuel uncovery from seismic events for IP2 and IP3 is less than or equal to $1x10^{-5}$ per year (Reference 27).

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 IP1, IP2, and IP3 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 IPEC.

Based on the above, HDI has demonstrated that no credible DBA or beyond design basis accident will result in radiological releases requiring offsite protective actions for any of the three IPEC units in decommissioning status. Additionally, there is sufficient time, resources, and personnel available to initiate mitigative actions that will prevent a radiological release that exceeds EPA PAG doses offsite.

Table 4

Industry Decommissioning Commitments (IDCs)

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 IP2 and IP3 crane designs meet the intent of this IDC. IP2 and IP3 both have single-failure-proof cranes designed to meet the requirements of ASME NOG-1-2004,Appendix C of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," (Reference 37) and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," (Reference 38). These single-failure-proof cranes are used to support spent fuel cask handling activities at IP2 and IP3. Because the cranes are single-failure-proof, an accidental load drop is not considered 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) is not required.
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.	HDI maintains IPEC procedures to ensure onsite and offsite resources can be brought to bear during an event, including: • IPEC-EP – IPEC Emergency Plan • 0-AOP-SEC-1 – Response to Security Compromise • 0-AOP-SEC-2 – Aircraft Threat • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Actions • 0-AOP-SEISMIC-1 – Seismic Event • 2-AOP-SF-1 / 3-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 2-AOP-FLOOD / 3-AOP-FLOOD - Flooding • 2-AOP-CCW-1 / 3-AOP-CCW-1 – Loss of Component Cooling Water • 0–SOP-WEATHER-002 – Severe Weather Preparations These procedures (or equivalent) are required by NRC regulations and will be implemented as necessary depending on the type of event. The procedures and associated training will be updated as necessary to reflect the permanently shutdown and defueled condition. Following permanent shutdown and permanent removal of fuel from the IP2 and IP3 reactor vessels, the on-shift plant operators, including Shift Managers (qualified as Certified Fuel Handlers (CFHs)) and Non-Certified Operators, will continue to be

Table 4

Industry Decommissioning Commitments (IDCs)

IDC	Industry Commitments	Response
		appropriately trained on the relevant procedures and on the various actions needed to provide makeup to the SFP based on a systematic approach to training. Following permanent cessation of power operations, maintaining SFP inventory would be the highest priority activity. Therefore, the personnel needed to perform these actions will be available at all times. The IPEC CFH training program was approved by the NRC by letter dated December 18, 2019 (Reference 39). Periodic Emergency Plan drills and exercises 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.	HDI maintains IPEC procedures that provide guidance to establish and maintain communications between onsite and offsite organizations during severe weather and seismic events, including the following: • IPEC-EP – IPEC Emergency Plan • 0-AOP-SEISMIC-1 – Seismic Event • EN-LI-108 – Event Notification and Reporting • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Actions These procedures (or equivalent) provide direction for additional actions and communications with onsite and offsite stakeholders if the event does not reach the threshold for entry into the PDEP. If the severity of the event requires entry into the PDEP, communications with onsite and offsite organizations will be directed by the PDEP and associated procedures. The procedures (or equivalent) will be updated as necessary to reflect the permanently shutdown and defueled condition. These procedures are required by NRC Regulations and will be implemented as necessary depending on the type of event. Communications are described in the procedures for onsite and offsite communications, they are not specifically referenced in the existing IPEC Emergency Plan and will not be included in the PDEP (to be submitted for NRC approval). Therefore, it is not necessary for the communications procedures to be specifically

Table 4

Industry Decommissioning Commitments (IDCs)

IDC	Industry Commitments	Response
		referenced in the Emergency Plan. Equipment requirements are specified in the pertinent procedures.
4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.	IPEC has multiple portable pumps and portable emergency generators that meet the Extensive Damage Mitigation Guidelines (EDMGs). In addition, the IPEC Emergency Plan provides guidance for communicating with and obtaining offsite resources. In addition, the following procedures may be used to support mitigation strategies for SFP damage and water supply: • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Action The procedures (or equivalent) will be updated as necessary to reflect the permanently shutdown and defueled condition of the IPEC units.
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.	The IP2 and IP3 designs meet the intent of this IDC. IP2 SFP water level is monitored via two independent level channels that were added to meet the post-Fukushima requirements. LS-6500A and LS-6500B indicate remotely in the IP2 Fan House. High and low SFP water level is indicated by LC-650 and alarmed in the IP2 Control Room. The high-low alarm is a float switch assembly set for plus or minus 6” of the normal level, which is 93 feet-9 inches. The IP2 SFP temperature is locally indicated by TIC-651 and high temperature alarmed by TIC-651 in the IP2 Control Room at 125 degrees Fahrenheit (°F). IP3 SFP water level is monitored via two independent level channels that were added to meet the post-Fukushima requirements. LS-6500A and LS-6500B indicate remotely in the IP3 PAB 67 foot elevation. LC-650 actuates the SFP level alarm in the IP3 Control Room. Determination of the alarm condition (i.e., high or low SFP level) is accomplished locally. The high-low alarm is a float switch assembly set for 10 inches from the top of the SFP for the high alarm and 22 inches from the top of the SFP for the low-level alarm. The IP3 SFP temperature is locally indicated by TIC-651. TIC- 651 also actuates the SFP high temperature alarm at 135ºF in the IP3 Control Room.

Table 4

Industry Decommissioning Commitments (IDCs)

IDC	Industry Commitments	Response
		Area radiation monitors are located in each of the IP2 and IP3 FSBs and the IP2 PAB. Audible alarms are provided in each respective Control Room.
6	SFP seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self-limiting to leakage or otherwise engineered so that drainage cannot occur.	Neither SFP (IP2 or IP3) have gates with seals that could lead to fuel uncovery. However, a gate isolates the SFP from the fuel transfer canal at each unit. The canal is connected to the Fuel Transfer Tube to the Vapor Containment (VC). The Fuel Transfer Tube to the VC for IP2 has been filled with concrete and the Fuel Transfer Tube for IP3 is sealed welded with a blank flange on the VC side and a locked gate valve on the SFP side. Therefore, if the SFP gates were to leak by, there is no path for SFP leakage into the VC. Although the top of the fuel racks at both units are higher than the bottom of the fuel transfer canal slot, if the transfer gate seals were to fail, the volume of the transfer canal is significantly smaller when compared to the SFP such that following the loss of the gate seal, the SFP would only lose enough water volume to lower the pool level by less than 4 feet. Therefore, failure or leakage of a SFP gate seal in either unit would not lead to fuel uncovery.
7	Procedures or administrative controls to reduce the likelihood of rapid draindown events will include: (1) prohibitions on theuse 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.	Design features and administrative controls which reduce the likelihood of rapid draindown events are in place for the IP2 and IP3 SFPs. The Technical Specification minimum SFP level is greater than or equal to 23 feet above the top of the fuel assemblies seated in the storage racks, which is at 92 feet-2 inches for IP2 and 91 feet-8 inches for IP3, and there are two alarms that would annunciate in the event of SFP draindown at either unit. The top of the fuel storage rack is at 69 feet-8¼ inches at IP2 and 69 feet-7½ inches at IP3. The lowest drain point with available alignment to installed pumps is the SFP cooling return line in both units, which is equipped with an anti-siphon hole, although it is not functionally tested. If unmitigated draining were to occur through this line, the lowest pool level that could be reached would still be above the Technical Specification minimum level for each unit and well above the top of the fuel assemblies in the fuel storage racks. If draining were to occur in the SFP, it would be signaled by two level alarms that annunciate in each respective Control Room.
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	Repairs to equipment designated for the SFPs will be performed using the normal online work management system (or equivalent).

Table 4

Industry Decommissioning Commitments (IDCs)

IDC	Industry Commitments	Response
	alignment of the makeup source to the SFP without requiring entry to the refuel floor.	Onsite procedures will remain in place to provide guidance for filling the SFPs in both normal and emergency conditions. Sources of makeup to the IP2 and IP3 SFPs include the Primary Water Storage Tank (PWST) water, fire water inside the SFPbuildings, and fire water using a temporary diesel pump from outside of the SFP buildings. The following procedures (or equivalent) will remain in place to perform filling and loss of cooling recovery of the SFPs during an abnormal loss of cooling or level: • 3-SOP-SFP-003 – Operation of the Backup SFP Cooling System • 2-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 3-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 2-AOP-CCW-1 – Loss of Component Cooling Water • 3-AOP-CCW-1 – Loss of Component Cooling Water The following procedures (or equivalent) will remain in place to perform filling the SFPs in the event that access to the SFP Floor is inaccessible: • 0-AOP-SEC-1 – Response to Security Compromise • 0-AOP-SEC-2 – Aircraft Threat • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Action • 0-SOP-ESP-2 – Emergency Contingency Plan There are multiple ways to add makeup water to the IP2 and IP3 SFPs with or without entry to the refuel floors.
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	IP2 procedure 2-DCS-009-GEN "MPC Transfer & HI-STORM Movement," requires the 110 Ton Gantry Crane to pass a pre-use inspection per procedures 2-DCS-026-GEN, "FSB 110 Ton X-SAM Gantry Crane" and 2-DCS-027-GEN, "FSB 110 Ton X-SAM Gantry Crane Preventative Maintenance," prior to moving any load. A qualified CFH is required to approve any heavy load moved in the FSB.

Table 4

Industry Decommissioning Commitments (IDCs)

IDC	Industry Commitments	Response
	designated operations or administrative limitations such as restrictions on heavy load movements.	IP2 procedure 2-DCS-009-GEN, "MPC Transfer & HI-STORM Movement," limits lifts and movement of multi-purpose canisters (MPCs) to a section of the SFP with no fuel assemblies in place. The transfer path is limited to one section of the SFP wall and is designed to limit interaction with SFP cooling piping. IP3 procedure 3-SOP-CM-002, "Fuel Storage Building Crane Operation," delineates the specific path the Shielded Transfer Canister (STC) must follow. The procedure specifies that the spent fuel transfer cask and the STC shall not be moved over the spent fuel storage racks in any region of the SFP containing irradiated fuel. If the SFP contains irradiated fuel, then movement across the SFP involving loads greater than 2000 pounds and movement across the SFP with FSB ventilation inoperable are also limited by this procedure. These procedures (or equivalent) will remain in effect for IP2 and IP3 SFP operations. The following procedures (or equivalent) will remain in place to perform filling and loss of cooling recovery of the SFP in the event of an abnormal loss of cooling or level: • 3-SOP-SFP-003 – Operation of the Backup SFP Cooling System • 2-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 3-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 2-AOP-CCW-1 – Loss of Component Cooling Water • 3-AOP-CCW-1 – Loss of Component Cooling Water The following procedures (or equivalent) will remain in place to perform filling the IP2 and IP3 SFPs if access to the SFP Floor is inaccessible: • 0-AOP-SEC-1 – Response to Security Compromise • 0-AOP-SEC-2 – Aircraft Threat • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Action • 0-SOP-ESP-2 – Emergency Contingency Plan

<u>Table 4</u>

Industry Decommissioning Commitments (IDCs)

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.	Both IP2 and IP3 have motor and diesel-driven fire pumps, as well as two diesel driven B.5.b pumps (shared between the Units) that can be used to provide makeup water to either SFP. Repairs to equipment designated for the SFPs will be performed using the normal work management system (or equivalent). Current Preventative Maintenance (PM) and Work Orders (or equivalent) will remain in place for all SFP equipment. Testing remains in place for SFP equipment and includes level indication, pumps, and installed backup pumps. B.5.b equipment PMs will remain in effect until all fuel is transferred out of each SFP.

Table 5

Staff Decommissioning Assumptions (SDAs)

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 IP2 and IP3 designs meet with the intent of this SDA. Both units' SFP Cooling System designs have two independent trains of SFP cooling. Each train rejects its heat to the Component Cooling Water (CCW) System at each unit, and its heat, in turn, is rejected to the Service Water (SW) System at each unit, with its heat being rejected to the Hudson River. Any changes to the SFP cooling configuration as a result of permanent cessation of power operations will be evaluated to confirm that the resulting configuration is at least as capable as the design assumed in Section 3.0 of NUREG-1738 (Reference 22). Both units have motor and diesel-driven fire pumps, as well as two diesel driven B.5.b pumps (shared between the units).
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 a walkdown of the SFP systems at IPEC is performed each shift (twice per day) and SFP normal instrumentation readings are recorded during operator rounds. The backup level instrumentation readings are recorded on a weekly basis during operator rounds. The capability to monitor SFP temperature and level (via alarms) is in place in the IP2 and IP3 Control Rooms. These rounds (or equivalent) will remain in place following the permanent shutdown of IP2 and IP3 and defuel of the reactor vessels. Both units will maintain the procedures (or equivalent) used for operation and filling of the SFP and its systems. The procedures listed below (or equivalent) will remain in place and provide the details regarding the use of multiple sources of replacement inventory. The following procedures (or equivalent) will remain in place to perform filling and loss of cooling recovery of the SFP in the during an abnormal loss of cooling or level: • 3-SOP-SFP-003 – Operation of the Backup SFP Cooling System • 2-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 3-AOP-SF-1 – Loss of Spent Fuel Pit Cooling • 2-AOP-CCW-1 – Loss of Component Cooling Water • 3-AOP-CCW-1 – Loss of Component Cooling Water

Table 5

Staff Decommissioning Assumptions (SDAs)

SDA	Staff Assumptions	Response
		The following procedures (or equivalent) will remain in place to perform filling the SFP in the event that access to the SFP Floor is inaccessible: • 0-AOP-SEC-1 – Response to Security Compromise • 0-AOP-SEC-2 – Aircraft Threat • 0-AOP-SEC-3 – Event Contingency Actions • 0-AOP-SEC-4 – Wide Area Event Contingency Action • 0-SOP-ESP-2 – Emergency Contingency Plan Walkdown of the IP2 and IP3 SFP systems will remain in place following permanent cessation of power operations. The procedures listed above (or equivalent) will be in place and updated as necessary to reflect the permanently shutdown and defueled condition of IP2 and IP3.
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.	The IP2 and IP3 designs meet the intent of this IDC. IP2 SFP water level is monitored via two independent level channels that were added to meet the post-Fukushima requirements. LS-6500A and LS-6500B indicate remotely in the IP2 Fan House. High and low SFP water level is indicated by LC-650 and alarmed in the IP2 Control Room. The high-low alarm is a float switch assembly set for plus or minus 6 inches of normal level, which is 93 feet-9 inches. The IP2 SFP temperature is locally indicated by TIC-651 and high temperature alarmed by TIC-651 in the IP2 Control Room, at 125°F. IP3 SFP water level is monitored via two independent level channels that were added to meet the post-Fukushima requirements. LS-6500A and LS-6500B indicate remotely in the IP3 PAB 67-foot elevation. LC-650 actuates the SFP level alarm in the IP3 Control Room. Determination of the alarm condition (i.e., high or low SFP level) is accomplished locally. The high-low alarm is a float switch assembly set for 10 inches from the top of the SFP for the high alarm and 22 inches from the top of the SFP for the low-level alarm. The IP3 SFP temperature is locally indicated by TIC-651. TIC- 651 also actuates the SFP high temperature alarm at 135ºF in the IP3 Control Room. Regarding the declaration of a General Emergency, IPEC will employ permanently defueled EALs using an NRC-approved EAL Scheme, based on the guidance provided

Table 5

Staff Decommissioning Assumptions (SDAs)

SDA	Staff Assumptions	Response
		in Appendix C of NEI 99-01, Revision 6. Station conditions will not reach any threshold requiring the declaration of a General Emergency.
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 resources.	The IP2 and IP3 SFP designs meet the intent of this SDA. The lowest point of the suction line in the SFPs is just a few feet below the Technical Specifications minimum levels at each unit. The lowest drain point with available alignment to installed pumps is the SFP cooling return line for both units, both of which are equipped with an anti- siphon hole located at an elevation slightly above the Technical Specifications minimum level. If unmitigated draining were to occur through this line, the lowest SFP level that could be reached would still be above the Technical Specification minimum level for each unit and well above the top of the fuel assemblies in the fuel storage racks. If draining were to occur in either SFP, it would be signaled by two level alarms that annunciate in the respective Control Room. Therefore, neither drain path is considered a credible failure mode for inventory loss given that inventory loss is not the direct result of catastrophic failures.
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 IP2 and IP3 designs meet the intent of this SDA. Heavy load lifts in and around the area of the SFPs are performed by single-failure-proof cranes that handle casks in the FSBs. Therefore, performance of load drop consequence analyses is not required.
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to 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	HDI developed an analysis (Reference 27) demonstrating successful completion of the Enhanced Seismic Checklist provided in Attachment 1 to Appendix 2B of NUREG-1738 (Reference 22) for the IP2 and IP3 SFPs. This analysis is described in Section 5.4 and is summarized in Table 6 of this Enclosure.

Table 5

Staff Decommissioning Assumptions (SDAs)

SDA	Staff Assumptions	Response
	estimates provided in this study [NUREG-1738] (<1 x10^{-5} per year including non-seismic events).	
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.	This SDA does not apply to the spent fuel racks for IP2, as the IP2 SFP criticality analysis does not credit the Boraflex panels in its spent fuel racks. The IP2 Technical Specification controls on SFP boron concentration and spent fuel rack storage provide assurance that the required 5 percent sub-criticality margin is maintained without crediting neutron absorber inserts in the spent fuel racks. The IP3 spent fuel racks utilize Boral (boron carbide/aluminum powder clad in aluminum) rather than Boraflex as a neutron absorber material. All of the storage cells in the two regions of spent fuel racks are bounded on four sides by Boral sheets, except on the periphery of the rack array. As described in Appendix A of the IP3 DSAR, the Boral Surveillance Program is an existing aging management program which provides assurance that the Boral neutron absorbers in the spent fuel racks maintain validity of the criticality analysis in support of the spent fuel rack design. The program relies on representative coupon samples mounted in surveillance assemblies located in the SFP. Surveillance assemblies are removed from the SFP on a prescribed schedule and the physical and chemical properties are measured. From this data, the performance, stability, and integrity of the Boral in the storage cells are monitored and assessed without disrupting the integrity of the storage system.

Table 6

Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning

Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
1	Requirement: Identify Preexisting Concrete and Liner Plate Degradation Basis: A detailed review of plant records concerning spent fuel pool concrete and liner plate degradation should be performed and supplemented by a detailed walkdown of the accessible portions of the spent fuel pool concrete and liner plate. The purpose of the records review and visual inspection activities is to accurately assess the material condition of the SFP concrete and liner in order to assure that these existing material conditions are properly factored into the remaining seismic screening assessments. Design Feature: The material condition of the SFP concrete and liner, based upon the records review and the walkdown inspection, will be documented and used as an engineering input to the following seismic screening assessments.	<u>IP2 Evaluation</u> A search of Condition Reports (CRs) was conducted for IP2 SFP issues related to the concrete or liner plate. Several Corrective Actions (CAs) associated with the CRs were identified, and a significant amount of verbiage associated with the various CAs for the CRs was reviewed. However, none of the CR responses and associated CAs indicated that the SFP structural integrity was affected. CA-8 under CR-IP2-2005-3557 entailed the development of a calculation to address the SFP wall rebar condition and structural integrity of the SFP. The conclusion section of the calculation (IP-CALC-05-00952) noted: *"… any potential degradation of the Indian Point Unit 2 Spent Fuel Pit Structure due to presently identified levels of pool water leakage will not adversely have an affect [sic] on its Class I safety function at the present time, or within the foreseeable future."* In addition, two 4-inch diameter core bores were drilled into the wall at the location of the observed indication exposing the rebar. Exposed rebar was observed to be in excellent condition with no indication of wall loss or corrosion products present. During the 2005 timeframe, a study of the permeability of the IP2 SFP was conducted to investigate the limit of potential steady state transfer of SFP water through the concrete wall of the SFP. The report did not address SFP integrity. In 2016, a report was developed to address the use of 308 weld wire to attach the pool liner to the structural steel. The report noted: *"There is no evidence that SFP leakage is affecting the structural integrity of the SFP structure. Therefore, if elimination of the current leakage cannot be achieved, MPR [Associates, Inc.] considers that continued monitoring and*

<u>**Table 6**</u>

Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning

Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		management of the leakage would be appropriate. This approach is currently employed by other nuclear facilities in the United States with SFP leakage." Fuel Storage Building (FSB) Maintenance Rule Inspection reports were reviewed. These reports identify minor cracking on the north wall of the truck bay (south wall of the SFP). No effect on structural integrity was noted and no leaching was noted. Based on these reviews, it can be concluded that there has been past leakage through the IP2 SFP liner plate. However, the structural integrity of the IP2 SFP structure has not been impacted. As a result, the existing material condition of the SFP concrete and liner do not have to be factored into the remaining items on the seismic checklist. <u>IP3 Evaluation</u> CRs were searched for SFP-related issues and several relevant CRs were found. There has never been any identified SFP water at the tell-tale drain during the 15 years of testing by Chemistry Technicians. This test is currently performed monthly, per 3-CY-2325, including the measurement of boron and isotopic activity. This record, along with the fact that the boron in the SFP has been stable, is conclusive evidence that the IP3 SFP does not have anyappreciable liquid leaks and the loss rate from the IP3 SFP is entirely due to evaporation. The IP3 SFP liner (leak detection piping) was inspected by inserting a video probe into the pipe. No obstructions were found during this inspection. Since the drainpipe is approximately 37 to 38 feet long, a metal snake was also inserted into the drainpipe approximately 37.5 feet and no obstructions were encountered. Minor amounts of water were found in the pipe which

Table 6

Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning

Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		would further confirm the absence of blocking of the piping. Based on this information, it is concluded that no obstructions were found during this inspection. While performing the two-year Preventative Maintenance (PM) to verify that the SFP liner drainpipe was not clogged, water drained from the pipe when the snake was removed from the pipe. Initial testing indicated that the water was not contaminated which would indicate that the source was most likely condensation. FSB Maintenance Rule Inspection Reports were reviewed which identified minor cracking and a minor cold joint on the north wall of the truck bay (south wall of the SFP). There are also several minor cracks noted on the exterior east wall of the structure (east wall of the SFP) that had no apparent increase in size. No effect on structural integrity was noted and no leaching was noted. Based on the above discussion it can be concluded that there has not been any known leakage through the IP3 SFP liner plate. There are no issues with the structural integrity of the IP3 SFP structure. As a result, the existing material condition of the SFP concrete and liner do not have to be factored into the remaining items on the seismic checklist.
2	Requirement: Assure Adequate Ductility of Shear Wall Structures Basis: The expert panel involved with the development of Reference 1 ["A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," (EPRI NP-6041-SL), August 1991] concluded that, "For the Category 1 structures which comply with the requirements of either ACI [American Concrete Institute] 318-71 or ACI 349-76 or later building codes and are designed for an SSE [safe shutdown earthquake] of at least 0.1g PGA [peak ground acceleration], as long as they do not have	<u>IP2 Evaluation</u> IP2 Calculation UEC-00035-00 (Fuel Storage Building Concrete Components– Pit, Floor Slab 95 foot, 80 foot, Column Footing & Pedestal) designed and evaluated the shear walls in accordance with ACI 318-63 using an SSE PGA of 0.15 g. Drawing No. 9321-1196 shows that the thickness of the shear walls is robust with the lower 16 feet-2 inches of wall having a thickness of 48 inches. The wall transitions from a thickness of 48 inches to 75 inches over a vertical distance of 2 feet-3 inches. The remaining 22 feet height of the walls has a thickness of 75 inches. Drawing No. 9321-1196 shows that 16 feet of the 75-inch-thick shear walls are above

Table 6 Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning		
Item	**Enhanced Seismic Checklist Requirement, Basis, and Design Feature**	**Response**
	any special problems as discussed below, the HCLPF [high confidence with low probability of failure] capacity is at least 0.5g PGA." This conclusion was based upon the assumption that the shear wall structure will respond in a ductile manner. The "special problems" cited deal with individual plant details which could prevent a particular plant from responding in the required ductile fashion. Examples cited in Reference 1 included an embedded structural steel frame in a common shear wall at the Zion plant (which was assumed to fail in brittle manner due to a potential shear failure of the attached shear studs) and large openings in a "crib house" roof (also at the Zion plant) which could interrupt the continuity of the structural slab. Other examples which could impact the ductility of the spent fuel pool structure include large openings which are not adequately reinforced or reinforcing bars that are not sufficiently embedded to prevent a bond failure before the yield capacity of the steel is reached. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	grade (ground level), which means 24 feet-5 inches of the shear walls (the majority of the height of each wall) is embedded within the soil/backfill. The Checklist Requirement notes the use of ACI 318-71 or later editions. ACI 318-71 specifies the use of the Ultimate Strength Method which uses factored design loads for dead and live load. The method places the rebar in full yield so the strain relationship between reinforcement and concrete is ignored and a rectangular concrete compression block stressed at design strength is formed. The method ends up providing a reduction in steel reinforcement (up to about 20%) and gives smaller dimensions of cross sections of the concrete members compared with the ACI 318-63 Working Stress Method. The design of the shear walls for IP2 and IP3 utilized the Working Stress Method and therefore have a greater amount of rebar and cross section dimensions than if it had been designed utilizing the ACI 318- 71 Ultimate Strength Method and in turn the shear walls are more robust. The term ductility when used in earthquake engineering is to designate how well a building (structure) will endure large lateral displacements imposed by ground shaking. As noted above the shear walls are quite thick and robust, including above grade where amplification of the structure will occur in a seismic event. The IP2 SFP walls were designed for an SSE PGA of 0.15 g which is larger than the minimum 0.1 g PGA specified in the Item 2 requirement and are of sufficient strength that they will resist the lateral forces from a seismic event. A review of the IP2 calculations and drawings do not have any of the special problems discussed within Item 2 of the Seismic Checklist, and it can be concluded that the HCLPF capacity of the IP2 shear wall structures is at least 0.5 g PGA. Therefore, it can be concluded that this requirement is met for IP2.

Table 6 Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning		
Item	**Enhanced Seismic Checklist Requirement, Basis, and Design Feature**	**Response**
		IP3 Evaluation IP3 Calculation IP3-CALC-STR-00634 (Fuel Storage Building Spent Fuel Pit Concrete Design, Seismic Check and Horizontal Missiles) designed and evaluated the shear walls in accordance with ACI 318-63 using an SSE PGA of 0.15 g. Drawing No. 9321-F-11973 shows that the thickness of the shear walls is robust with a thickness of 75 inches. Calculation IP3-CALC-STR-00634 shows that about 5-1/2 feet of the shear walls are embedded in soil/backfill. The Checklist Requirement notes the use of ACI 318-71 or later editions. ACI 318-71 specifies the use of the Ultimate Strength Method which uses factored design loads for dead and live load. The method places the rebar in full yield so the strain relationship between reinforcement and concrete is ignored and a rectangular concrete compression block stressed at design strength is formed. The method ends up providing a reduction in steel reinforcement (up to about 20%) and gives smaller dimensions of cross sections of the concrete members compared with the ACI 318-63 Working Stress Method. The design of the shear walls for IP2 and IP3 utilized the Working Stress Method and therefore have a greater amount of rebar and cross section dimensions than if it had been designed utilizing the ACI 318- 71 Ultimate Strength Method and in turn the shear walls are more robust. The term ductility when used in earthquake engineering is to designate how well a building (structure) will endure large lateral displacements imposed by ground shaking. As noted above the shear walls are quite thick and robust, including above grade where amplification of the structure will occur in a seismic event. The IP3 SFP walls were designed for an SSE PGA of 0.15 g which is larger than the minimum 0.1 g PGA specified in the Item 2 requirement and are of sufficient strength that they will resist the lateral forces from a seismic event. A review of the IP3 calculations and drawings do not have any of the special problems discussed within Item 2 of the Seismic Checklist, and it can be

Table 6

Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning

Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		concluded that the HCLPF capacity of the IP3 shear wall structures is at least 0.5 g PGA. Therefore, it can be concluded that this requirement is met for IP3.
3	Requirement: Assure Design Adequacy of Diaphragms (including Roofs) Basis: In the design of many nuclear power plants, the seismic design of roof and floor diaphragms has often not received the same level of attention as have the shear walls of the structures. Major cutouts for hatches or for pipe and electrical chases may pose special problems for diaphragms. Since more equipment tends to be anchored to the diaphragm compared to shear walls, moderate amounts of damage may be more critical for the diaphragm to the same amount of damage in a wall. Based upon the guidance provided in Reference 1, diaphragms for Category I structures designed for an SSE of 0.1 g or greater do not require an explicit evaluation provided that: (1) the diaphragm loads were developed using dynamic analysis methods; (2) they comply with the ductility detailing requirements of ACI 318-71 or ACI 349-76 or later editions. Diaphragms which do not comply with the above ductility detailing or which did not have loads explicitly calculated using dynamic analysis should be evaluated for a beyond-design-basis seismic event in the 0.45 - 0.5g PGA range. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	<u>IP2 and IP3 Evaluation</u> IP2 Calculation UEC-00036-00 (Fuel Storage Building Structural Steel Components – Crane Column, Crane Girder, Roof Truss, Platform, and Liner Plates) designed and evaluated the structure of the FSB above the SFP. The calculation was performed by using standard textbook static analysis methods. A review of IP2 FSB Drawing Nos. 9321-1306, 9321-1307, and 9321-1308 were compared to IP3 FSB Drawing Nos. 9321-F-13063, 9321-F-13073, and 9321-F-13083 and it was concluded that the two FSB structures are identical. In the late 1990's a dynamic structural analysis of the IP3 FSB was developed and is contained in Technical Report No. 2123-39-01 (Dynamic and Structural Analysis of the FSB [Heat Exchanger Area]), dated March 1999. The dynamic analysis generated new north-south and east-west in-structure response spectra using an SSE ground acceleration of 0.15 g. Synthetic time history was developed, in accordance with American Society of Civil Engineers (ASCE) 4-86, as the seismic excitation. The peaks of the response spectra were broadened ±15 percent. The dynamic analysis utilized the American Institute of Steel Construction (AISC) Manual of Steel Construction, 9th Edition, and ACI 318-95. The dynamic analysis concluded that the steel columns and vertical braces, concrete beams and columns, and block walls were all acceptable. From the analysis it can be concluded that the design adequacy of the roof and floor diaphragms are acceptable as the diaphragm loads were developed using dynamic analysis methods and they comply with the ductility detailing requirements of ACI 318-95. The dynamic analysis was developed for IP3. However, it can be concluded that it is also applicable to IP2 since the buildings are identical, as noted above. Therefore, it can be concluded that this requirement is met for IP2 and IP3.

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
4	Requirement: Verify the Adequacy of the SFP Walls and Floor Slab to Resist Out-of-Plane Shear and Flexural Loads Basis: For PWR pools that are fully or partially embedded, an earthquake motion that could cause a catastrophic out-of-plane shear or flexural failure is very high and is not a credible event. For BWR pools (and PWR pools that are not at least partially embedded), the seismic capacity is likely to be somewhat less and the potential for out-of-plane shear and/or flexural wall or base slab failure, at beyond-design-basis seismic loadings, is possible. A structural assessment of the pool walls and floor slab out-of-plane shear and flexural capabilities should be performed and compared to the realistic loads expected to be generated by a seismic event equal to approximately three times the site SSE. This assessment should include dead loads resulting from the masses of the pool water and racks, seismic inertial forces, sloshing effects and any significant impact forces. Credit for out-of-plane shear or flexural ductility should not be taken unless the reinforcement associated with each failure mode can be shown to meet the ACI 318-71 or ACI 349-49 requirements. Design Feature: Compliance with this design feature will be documented based upon a review of drawings (in the case of embedded or partially embedded PWR pools) or based upon a review of drawings coupled with the specified beyond-design-basis shear and flexural calculations outlined above.	The IP2 and IP3 SFPs are partially embedded. Therefore, the amount of earthquake motion to cause an issue is very high and therefore this is not a credible event.

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
5	Requirement: Verify the Adequacy of Structural Steel (and Concrete) Frame Construction Basis: At a number of older nuclear power plants, the walls and roof above the top of the spent fuel pool are constructed of structural steel. These steel frames were generally designed to resist hurricane and tornado wind loads which exceeded the anticipated design basis seismic loads. A review of these steel (or possibly concrete) framed structures should be performed to assure that they can resist the seismic forces resulting from a beyond-design-basis seismic event in the 0.45 - 0.5g PGA range. Such a review of steel structures should concentrate on structural detailing at connections. Similarly, concrete frame reviews should concentrate on the adequacy of the reinforcement detailing and embedment. Failure of the structural steel superstructure should be evaluated for its potential impact on the ability of the spent fuel pool to continue to successfully maintain its water inventory for cooling and shielding of the spent fuel. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	<u>IP2 and IP3 Evaluation</u> Section 1.7.6.2 of the IP2 Defueled Safety Analysis Report (DSAR) (Rev. 0, 2020) provides a discussion of the seismic evaluation of the FSB Structure above the SFP. It notes that the superstructure above the SFP was designed as a Class III structure. The seismic loads used in the original plant construction analysis of the steel superstructure were: 1) Zero period ground acceleration: 0.15 g horizontal, 0.10 g vertical 2) 7 percent damping 3) Response spectrum curve as defined in DSAR Figures 1.11-1 and 1.11-2. 4) Inertial forces for each mass point were determined on the basis of the square root of the sum of the squares (SRSS). A dynamic multi-degree of freedom, modal analysis of the structure was constructed. The stiffness properties of the elements were determined by the combined stiffness of the frame bents in the north-south and east-west directions taken separately. The stiffness of each bent was determined by the computer program STRUDL. The total inertial forces determined by the dynamic analysis were distributed to each individual bent and resultant member stresses were determined. The crane was assumed to be fully loaded. Evaluation of these seismic stresses show maximum stresses occurring in the diagonal bracing. The maximum stress thus determined in the cross bracing was 18.5 kilo pounds per square inch (ksi). It also states that the maximum dead and seismic column load stress determined by the analysis was 12.8 pounds per square inch (psi). From a review of the column loads shown on Drawing No. 9321-1308 the column load stress was actually 12.8 ksi. IP2 Drawing Nos. 9321-1306, 9321-1307, and 9321-1308 were reviewed including available bolting and welding connection details. Based on the stresses noted above, along with a review of the IP2 structural drawings, it can be concluded from engineering judgement that the structure can withstand the additional ground accelerations from a beyond-design-basis

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		seismic event although some of the members may go beyond yield strength. Strain hardening will ensure that the structure remains in-place over the SFP. IP3 Drawing Nos. 9321-F-13063, 9321-F-13073, and 9321-F-13083 were reviewed and show that the IP3 superstructure is identical to IP2. Therefore, the conclusion based on the IP2 DSAR discussion of the seismic evaluation of the IP2 FSB Structure above the SFP is also applicable to IP3 and it can be concluded that the IP3 structure will remain in-place over the SFP during a beyond-design-basis seismic event of 0.45 - 0.5 g PGA.
6	Requirement: Verify the Adequacy of Spent Fuel Pool Penetrations Basis: The seismic and structural adequacy of any spent fuel pool (SFP) penetrations whose failure could result in the draining or syphoning of the SFP must be evaluated for the forces and displacements resulting from a beyond-design-basis seismic event in the 0.45 - 0.5g PGA range. Specific examples include SFP gates and gate seals and low elevation SFP penetrations, such as, the fuel transfer chute/tube and possibly piping associated with the SFP cooling system. Failures of any penetrations which could lead to draining or syphoning of the SFP should be considered. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	<u>IP2 Evaluation</u> IP2 DSAR Rev. 0, Section 3.3.3.2 (Spent Fuel Pit Cooling Loop) states: *"The most serious failure of this loop is complete loss-of-water in the SFP. To protect against this possibility, the SFP cooling connections enter near the water level so that the SFP cannot be either gravity drained or inadvertently drained."* In response to the events associated with the earthquake and subsequent tsunami that affected the Fukushima Daichi nuclear power plant, the NRC issued a 10 CFR 50.54(f) Letter requesting information to assure that Near Term Task Force recommendations were addressed by all U.S. nuclear power plants. Electric Power Research Institute (EPRI) Report 1025286 (Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic) provides guidance for conducting seismic walkdowns as required in the 50.54(f) Letter, Enclosure 3, Recommendation 2.3: Seismic. Page 3-7 of the EPRI Report provides a listing of SFP-related items. Screen #4 is for the Rapid Drain-Down SFP items. Identification of any of these items are to be added to Seismic Walkdown Equipment List (SWEL) 2. The IP2 Fukushima Seismic Walkdown effort is contained in Engineering Report IP-RPT-12-00037, Rev. 1. Table 4 on Page B-40 of the report

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		provides a listing of SWEL 2 Rapid Draw-Down (Drain-Down) items. They are the Fuel Transfer Tube Blind Flange, Fuel Transfer Canal Weir Gate, and the abandoned 4-inch pipe penetration. These are the only SFP penetrations whose failure could result in the draining or syphoning of the SFP. The Fukushima report does not evaluate or walkdown these items and notes that they are excluded. However, these items are addressed in this evaluation to ensure they will not result in the draining of the SFP during a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range. The Fuel Transfer Tube Blind Flange for IP2 was changed to the design that is used at IP3. The current blind flange is shown on Drawing No. EDSK-319535-F, Rev. 6. Calculation IP-CALC-11-00065 addresses the changed design of the IP2 blind flange. Page 2 of Attachment A of the calculation calculates the maximum bolt stress for the bolted blind flange and the calculation also evaluates the structural integrity of the new blind flange. The horizontal seismic acceleration (1.85 g) used for the blind flange location at Elevation 67-foot appears to be for the FSB 122-foot east-west direction 1 percent SSE (Ref: Eng. Report IP-RPT-04-00481). IP2's SSE PGA is 0.15 g. The seismic checklist states to use a beyond design-basis PGA of 0.45 - 0.5 g. Therefore, the 1.85 g acceleration in the calculation will be increased by 3 (0.45 ÷ 0.15) resulting in a horizontal seismic acceleration of 5.55 g. Utilizing this seismic acceleration results in the following acceptable values for maximum blind flange bolt stress: P_e = horizontal seismic force from blind flange = 2,581 pounds B_t = tensile force per bolt = 6,571 pounds f_t = tensile stress per bolt = 8,092 psi B_s shear force per bolt = 821 pounds f_v shear stress per bolt = 1,011 psi F_{tb} = 13,200 psi F_{vb} = 9,300 psi Stress interaction = 0.39 < 1.0

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		A review of the calculation (IP-CALC-11-00065) for the structural integrity of the blind flange shows no seismic accelerations were used. The thickness of the flange is 1.687 inches with a groove depth of 0.22 inches or a net thickness of 1.467 inches provided. This is considered quite robust, and the calculation determined a minimum thickness of 0.82 inches. The structural integrity of the blind flange will not be affected by a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range. Based on the acceptable structural integrity of the blind flange, it can be concluded that a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range will not result in a failure of the Fuel Transfer Tube Blind Flange and therefore will not result in the draining or syphoning of the SFP. The IP2 Fuel Transfer Canal Weir Gate (also called FSB Tank Gate) is shown on Drawing No. 9321-1303, Rev. 13. The IP3 Fuel Transfer Canal Weir Gate is shown on Drawing No. 9321-F-13033, Rev. 9. A review of both drawings shows that the gate for IP2 and IP3 are identical. IP3 calculation IP3-TS-049 shows that the total weight of the door (i.e., weir gate) is 1,200 pounds. The door/gate has a height of 26 feet-6 inches, width of 1 foot-11 inches, and a thickness of ½ inch. A review of Drawing No. 9321-1303 (Plan View at Gate) shows that the door/gate is retrained in place on one end by 9 door hinge pin support plates with a ¾-inch 304 stainless steel hinge pin in each of the door hinge pin support plates. The other end of the door is restrained by nine 2 inch x 3/8 inch bronze throw bars which bear against a 1 inch x 1-1/2 inch flat bar. Engineering Report IP-RPT-04-00481 contains the seismic response spectra for IP2. From a review of that report the maximum acceleration on the seismic response spectra for the FSB at Elevation 95-foot for the 1 percent damping SSE is 0.66 g for both the north-south and east-west directions. Using SRSS result in a horizontal seismic acceleration of 0.93 g. The

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		seismic checklist states to use a beyond-design-basis PGA of 0.45 - 0.5 g. Therefore, the 0.93 g acceleration in the calculation will be increased by 3 (0.45 ÷ 0.15) resulting in a horizontal seismic acceleration of 2.79 g. This results in a horizontal load of 1,200 pounds (weight of the door) x 2.79 g or 3,348 pounds. The load seen by an individual hinge pin is equal to (3,348 ÷ 2) ÷ 9 = 186 pounds. The shear area of a 3/4 inch pin is equal to 0.44 $inch^2$. Therefore, the shear stress on the pin is equal to 423 psi, which is much less than the shear allowable of 12,000 psi (30,000 x 0.4) for a stainless-steel weld. The door hinge pin support plates are judged acceptable by inspection. As noted above the other end of the door is restrained by nine 2 inch x 3/8 inch bronze throw bars. Section 2-2 of Drawing No. 9321-1303 shows the length of the bar is 3-1/2 inch. Therefore, the bending moment on each individual bar is equal to 186 pounds (calculated above for the hinge pin) x 3-1/2 inch = 651 pound-inch. The section modulus of the bar is equal to 1/6 x 2 x $3/8^2$ = 0.0469 $inch^3$. This results in a stress due to bending of 13,880 psi. Table A-1 of Mechanics of Materials, 3rd Edition by Higdon, Ohlsen, Stiles, Weese, and Riley shows a tensile strength of 20,000 psi for annealed bronze. The calculated bending stress of 13,880 psi is less than 20,000 psi (yield strength) and it can therefore be concluded that the 9 throw bars will keep the door in place during a beyond-design-basis PGA of 0.45 - 0.5 g. Based on the above evaluation, it can therefore be concluded that a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range will not result in a failure of the Fuel Transfer Canal Weir Gate and therefore will not result in the draining or syphoning of the IP2 SFP. The last IP2 evaluation for this checklist item is the abandoned 4-inch pipe penetration. The abandoned 4-inch pipe penetration was originally part of the SFP Alternate Cooling System which was retired-in-place under ER-04-2-012. The 4-inch piping line (4"-AC-151R) and pipe supports at Elevation77-foot were removed from inside the SFP. The pipe was cut to about 3/8

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		inch from the existing fillet weld between the outside diameter of the pipe and the SFP liner plate. A ¾-inch thick 304 stainless steel plate (plug) was placed inside the diameter of the pipe and seal welded with a 3/16-inch fillet weld (Ref: Detail 3 & Detail 5 on Dwg. 9321-2577). The modification stated that the 3/16-inch fillet weld was larger than the 1/8-inch liner-to-pipe seal weld. Therefore, the seismic integrity of the SFP was maintained. The weld on the plug is quite robust and most of its loading is from the hydrostatic load from the water in the SFP. The additional loading from a seismic event is quite small when compared to the hydrostatic load, and the increased load from a beyond-design-basis PGA of 0.45 - 0.5 g would still be small. The weld has an area of 1.6 inch^2 (circumference of 12.57 inches x 0.707 x 3/16 inches). Using a shear allowable of 12,000 psi (0.4 x 30,000) for a stainless-steel weld would mean that the weld could withstand a load of nearly 20,000 lbs. It is concluded that the pipe penetration and pipe plug will remain in place during a beyond-design-basis PGA of 0.45 - 0.5 g. Based on the above pipe plug evaluation, it can be concluded that a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range will not result in a failure of the abandoned 4 inch pipe penetration and therefore will not result in the draining or syphoning of the IP2 SFP. <u>IP3 Evaluation</u> IP3 DSAR Rev. 0, Section 3.3.3 (Spent Fuel Pit Cooling Loop) states: *"The most serious failure of this loop is complete loss-of-water in the SFP. To protect against this possibility, the SFP cooling connections enter near the water level so that the SFP cannot be either gravity drained or inadvertently drained."* Section 3.5.2 of the IP3 DSAR states:

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		"Loss of water in the spent fuel pit and the resultant uncovering of the spent fuel by way of drains and permanently connected system cannot take place for the following reasons: *1) The suction of the spent fuel pit pump is taken from a point approximately six (6) feet below the top of the pool wall; therefore, this pump cannot be used to uncover the fuel, even accidently.* *2) The spent fuel pit pump discharge pipe terminates in the pool at elevation 74 feet-4¾ inches. This elevation is approximately five (5) feet above the top of the spent fuel assemblies; therefore, this pipe could not accidently become a siphon to uncover the fuel.* *3) The skimmer pump takes suction from, and discharges to the surface of the pool; therefore, it could not accidently or otherwise uncover the spent fuel.* *4) There are no drains on the bottom or side walls of the spent fuel pit. Draining would have to be done deliberately by a temporary pump.* *5) The spent fuel pit cooling loop was designed to seismic Class II and the cleanup loop was designed to seismic Class III criteria; however, their failure could not result in the uncovering of the spent fuel, as explained above."* The IP3 evaluation is similar to the above IP2 evaluation for this checklist item. The IP3 Fukushima Seismic Walkdown effort is contained in Engineering Report IP-RPT-12-00039, Rev. 1. Table 4 on Page B-34 of the report provides a listing of SWEL 2 Rapid Draw-Down (Drain-Down) items. They are the Fuel Transfer Tube Blind Flange and the Fuel Transfer Canal Weir Gate. These are the only SFP penetrations whose failure could result in the draining or syphoning of the SFP. The Fukushima report does not evaluate

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Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning

Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
		or walkdown these items and notes that they are excluded. However, these items are addressed in this evaluation to ensure they will not result in the draining of the SFP during a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range. The Fuel Transfer Tube Blind Flange for IP3 is shown on Drawing No. 9321-F-27153, Rev. 5, Drawing No. IP3V-0386-0010, Rev. 2, and Drawing No. IP3V-0439-1636, Rev. 1. A review of these drawings confirm that it is of the same design as that for IP2. The seismic response spectra were reviewed for the Containment and the SFP for Elevation 56 feet-7 inches (centerline of the Fuel Transfer Tube as shown on Dwg. 9321-F-27153) for the next elevation above 56 feet-7 inches and it was determined that the seismic accelerations were smaller than those used by the IP2 calculation IP-CALC- 11-00065 (for the changed design of the IP2 blind flange). Therefore, the evaluation shown above that was performed for the IP2 Fuel Transfer Tube Blind Flange is applicable to the IP3 Fuel Transfer Tube Blind Flange and the structural integrity is acceptable. The maximum acceleration on the seismic response spectra for the SFP at Elevation 95-foot for the 1 percent damping SSE is 0.55 g for the north-south and east-west directions. This is a lower seismic acceleration (0.55 g versus 0.66) than that which was used for the IP2 Evaluation of the Fuel Transfer Weir Gate discussed above. Therefore, the evaluation done for the IP2 Fuel Transfer Canal Weir Gate bounds the loading seen by the IP3 gate and can serve as the evaluation for the IP3 gate. Based on the above IP3 evaluation for this checklist item, it can be concluded that a beyond-design-basis seismic event in the 0.45 - 0.5 g PGArange will not result in a failure of the IP3 Fuel Transfer Tube Blind Flange orFuel Transfer Canal Weir Gate and therefore will not result in the draining or syphoning of the SFP.

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
7	Requirement: Evaluate the Potential for Impacts with Adjacent Structures Basis: Structure-to-structure impact may become important for earthquakes significantly above the SSE, particularly for soil sites. Structures are usually conservatively designed with rattle space sufficient to preclude impact at the SSE level but there are no set standards for margins above the SSE. In most cases, impact is not a serious problem but, given the potential for impact, the consequences should be addressed. For impacts at earthquake levels below 0.5g PGA, the most probable damage includes the potential for electrical equipment malfunction and for local structural damage. As cited previously, these levels of damage may be found to be acceptable or to result in the loss of SFP support equipment. The major focus of this impact review is to assure that the structure-to-structure impact does not result in the inability of the SFP to maintain its water inventory. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	<u>IP2 and IP3 Evaluation</u> In the Plan at Elevation 54 feet-7 inches on Drawing No. 9321-1197, Rev. 8, for IP2 and Drawing No. 9321-F-11973, Rev. 8, for IP3, the seismic gap between the FSB and the Containment (the nearest structure to the FSB) is shown to be equal to 17-5/8 inches. IP2 and IP3 are a rock site, and each FSB is founded on bedrock. Therefore, a soil structure interaction analysis isnot required, and additional movement or deflection of the buildings will not occur due to being founded on bedrock. The 17-5/8 inches provided is considered significantly large regarding potential deflection of the FSB. Therefore, an impact with the nearby Containment is not plausible. Based on the above, it can be concluded that there is no potential for impacts with adjacent structures for IP2 and IP3.
8	Requirement: Evaluate the Potential for Dropped Loads Basis: A beyond-design-basis seismic event in the 0.45 - 0.5g PGA range has the potential to cause the structural collapse of masonry walls and/or equipment supports systems. If these secondary structural failures could result in the accidental dropping of heavy loads which are always present (i.e. not loads associated with cask movements) into the SFP, then the consequences of these drops must be considered. As in previous evaluations, the focus of the drop consequence analyses should consider the possibility of draining the SFP. Additionally, the evaluation should evaluate the consequences of	<u>IP2 and IP3 Evaluation</u> The IP2 and IP3 drawings for the IP2 and IP3 FSBs were reviewed for such items as masonry walls and equipment support systems. The drawings which were reviewed for IP2 include: Drawing Nos. 9321-1196 Rev. 8, 9321-1197 Rev. 8, 9321-1198 Rev. 8, 9321-1199 Rev. 7, 9321-1200 Rev. 5, 9321-1201 Rev. 2, 9321-1202 Rev. 2, 9321-1203 Rev. 1, 9321-1204 Rev. 1, 9321-1205 Rev. 2, 9321-1206 Rev. 3, 9321-1207 Rev. 1, 9321-1306 Rev. 8, 9321-1307 Rev. 7, 9321-1308 Rev. 3, and 9321-1389 Rev. 12. The drawings which were reviewed for IP3 include:

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
	any resulting damage to the spent fuel or to the spent fuel storage racks. Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	Drawing Nos. 9321-F-11963 Rev. 6, 9321-F-11973 Rev. 8, 9321-F-11983 Rev. 6, 9321-F-11993 Rev. 9, 9321-F-12003 Rev. 7, 9321-F-12013 Rev. 3, 9321-F-12023 Rev. 3, 9321-F-12033 Rev. 3, 9321-F-12043 Rev. 2, 9321-F-12053 Rev. 4, 9321-F-12063 Rev. 4, 9321-F-12073 Rev. 2, 9321-F-13063 Rev. 5, 9321-F-13073 Rev. 4, 9321-F-13083 Rev. 1, and 9321-F-13893 Rev. 7. The review of the IP2 and IP3 drawings revealed that there were no masonry walls near the SFPs. The review of the drawings also revealed that there were no equipment support systems that would collapse or fall into the SFPs during a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range. Engineering Report No. IP-RPT-18-00046, Rev. 0, (Maintenance Rule Structural Monitoring Inspection Report [Fifth Cycle] for the IP2 FSB) and IP3 Engineering Report No. IP-RPT-15-00061, Rev. 0, (Maintenance Rule Structural Monitoring Inspection Report [Fifth Cycle]) were reviewed. There were no items for either unit in the vicinity of the SFP that were identified as deficient or unacceptable that could drop into the SFP. Based on the above evaluation for this checklist item, it can be concluded that there is no potential for accidental dropping of heavy loads into the SFP during a beyond-design-basis seismic event in the 0.45 - 0.5 g PGA range for IP2 or IP3.
9	Requirement: Evaluation of Other Failure Modes Basis: Experienced seismic engineers should review the geotechnical and structural design details for the specific site and assure that there are not any design vulnerabilities which will not be adequately addressed by the review areas listed above. Soil-related failure modes including liquefaction and slope instability should be screened by the approaches outlined in Reference 1 (Section 7 & Appendix C).	<u>IP2 and IP3 Evaluation</u> The Responsible Engineer responsible for developing this seismic checklist and the reviewer of this seismic checklist are very experienced seismic engineers, and both are Seismic Qualification Utility Group (SQUG) Seismic Capability Engineers. IP2 and IP3 are both founded on bedrock. As a result, there are no soil- related failure modes such as liquefaction or slope instability. The review areas above (Items 1 through 8), adequately address any potential design

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Item	Enhanced Seismic Checklist Requirement, Basis, and Design Feature	Response
	Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.	vulnerabilities and there are no other failure modes which need to be considered.
10	Potential Mitigation Measures Although beyond the scope of this seismic screening checklist, the following potential mitigation measures may be considered in the event that the requirements of the seismic screening checklist are not met at a particular plant. a. Delay requesting the licensing waivers (E-Plan, insurance, etc.) until the plant specific danger of a zirconium fire is no longer a credible concern. b. Design and install structural plant modifications to correct/address the identified areas of non-compliance with the checklist. (It must be acknowledged that this option may not be practical for significant seismic failure concerns.) c. Perform plant-specific seismic hazard analyses to demonstrate that the seismic risk associated with a catastrophic failure of the pool is at an acceptable level. (The exact "acceptable" risk level has not been precisely quantified but is believed to be in the range of $1x10^{-5}$ per year.)	<u>IP2 and IP3 Evaluation</u> No additional mitigation measures will be required for IP2 or IP3 because all requirements of the seismic checklist are met as noted within this table.

7.0 JUSTIFICATION FOR EXEMPTIONS AND SPECIAL CIRCUMSTANCES

In accordance with 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements 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. The Commission will not consider granting an exemption unless special circumstances are present. As discussed below, this exemption request satisfies the provisions of 10 CFR 50.12.

7.1 Exemptions

A. The exemptions are authorized by law

The provisions of 10 CFR 50.12 allow 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 provide 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, HDI has developed analyses (Reference 25) indicating that within 30 days after permanent cessation of power operations of each unit, the radiological consequences of the remaining design basis accidents (DBAs) at IP2 and IP3 will not exceed the limits of the Environmental Protection Agency (EPA) Protective Action Guides (PAG) at the exclusion area boundary (EAB). In addition, HDI has performed site specific heat up analyses for the IP2 and IP3 SFPs for beyond design basis events, which demonstrate that 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) assuming a complete loss of SFP water inventory. Based on the results of these analyses, 10 hours is available to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel in the unlikely event of a beyond design basis event (Reference 13).

Additionally, the offsite and Control Room radiological impacts of a postulated complete loss of SFP water were assessed (Reference 26). It was determined that the gamma radiation dose rate at the EAB is limited to small fractions of the EPA PAG exposure levels and the limiting dose rate in the IP2 and IP3 Control Rooms is 0.0259 mrem/hr at one year after permanent shutdown.

Therefore, offsite emergency response plans will no longer be needed for protection of the public beyond the EAB. Based on the reduced consequences of radiological events possible at IPEC with IP1, IP2, and IP3 in the permanently shutdown and defueled condition, the scope of the onsite emergency preparedness organization and offsite requirements in the IPEC Emergency Plan may be reduced without an undue risk to the public health and safety.

The underlying purpose of the regulations will continue to be met. Because the underlying purpose of the regulations 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 IPEC with IP1, IP2, and IP3 in the permanently shutdown and defueled condition allows for a corresponding reduction in the scope of the onsite emergency preparedness organization and associated reduction of requirements in the IPEC Emergency Plan. These reductions will not adversely affect IPEC's ability to physically secure the site or protect special nuclear material. Physical security measures at IPEC 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 exist at IPEC because IP1, IP2, and IP3 are permanently shutdown and defueled and the radiological source term at the site has been reduced from that associated with reactor power operation. With the reactors permanently shutdown and defueled, the DBAs and transients postulated to occur during reactor operation are no longer possible. Specifically, the potential for a release of a large radiological source term to the environment from the high pressures and temperatures associated with reactor operation no longer exists.

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 the postulated accidents that will remain possible at IPEC with IP1, IP2, and IP3 permanently shutdown and defueled are substantially lower than those at an operating plant. The upper bounds of the analyzed dose consequences limit 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 exemption request.

As discussed in this request, HDI performed analyses (Reference 25) indicating that within 30 days after permanent cessation of power operations of each unit, the radiological consequences of the remaining design basis accidents (DBAs) at IP2 and IP3 will not exceed the limits of the Environmental Protection Agency (EPA) Protective Action Guides (PAG) at the exclusion area boundary (EAB). HDI has performed site specific heat up analyses for the IP2 and IP3 SFPs for beyond design basis events, which demonstrate that at 15 months after permanent shutdown 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) assuming a complete loss of SFP water inventory. Based on the results of these analyses, 15 months after permanent shutdown a minimum of 10 hours is available to initiate appropriate mitigating actions to restore a means of heat removal to the spent fuel in the unlikely event of a beyond design basis event.

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, Section IV, 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 IPEC being permitted to reduce the scope of emergency preparedness requirements consistent with placing the units at IPEC 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, Section IV is not necessary for adequate emergency response capability and is excessive for a permanently shutdown and defueled condition. 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 Exelon Generation's Three Mile Island Nuclear Station, Unit 1 (TMI-1), Omaha Public Power District's Fort Calhoun Station (FCS), HDI's Pilgrim Nuclear Power Station (Pilgrim) and Oyster Creek Generating Station (Oyster Creek), Vermont Yankee Nuclear Power Station (VY), Southern California Edison Company's San Onofre Nuclear Generating Station (SONGS), Duke Energy Florida, Inc.'s Crystal River Unit 3 Nuclear Generating Station (CR3), and Dominion Energy Kewaunee, Inc.'s Kewaunee Power Station (KPS), have been granted similar exemptions.

Full compliance with the above listed regulations 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 defined in 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 granting of the exemptions. (10 CFR 50.12(a)(2)(iv))

All three IPEC plants are permanently shutdown and defueled and the radiological source term at the site has been reduced from that associated with reactor power operation. With the reactors permanently shutdown and defueled, the DBAs and transients postulated to occur during reactor operation are no longer 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 no longer exist.

The proposed exemptions would allow HDI to revise the IPEC Emergency Plan 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 revised IPEC Emergency Plan 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 (EALs) for events that are no longer possible would enhance the ability of the ERO to correctly classify those events that remain credible. As the proposed exemption 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 defined in 10 CFR 50.12(a)(2)(iv) exist.

8.0 PRECEDENT

The exemptions from the 10 CFR 50.47(b), 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E, Section IV requirements proposed by HDI for IPEC in this exemption request are consistent with exemptions from the same emergency planning requirements that have been issued by the NRC for other nuclear power reactor facilities beginning decommissioning. Specifically, the NRC granted similar exemptions to Exelon for TMI-1 (Reference 40), HDI for Pilgrim (Reference 41), Exelon for Oyster Creek (Reference 42), OPPD for FCS (Reference 43), Entergy for VY (Reference 44); Southern California Edison Company for SONGS, Units 1, 2, and 3 (Reference 45); Duke Energy Florida, Inc. for CR3 (Reference 46); and Dominion Energy Kewaunee, Inc. for KPS (Reference 47). 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, Section IV related to the elimination of offsite radiological emergency plans and 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(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) the requirements from which the exemptions are sought involve 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, Section IV would allow HDI to revise the scope of the IPEC Emergency Plan to reflect the permanently shutdown and defueled condition of Indian Point Nuclear Generating Units 1, 2, and 3 (collectively referred to as "facility" or "the facility"). 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 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. The proposed changes do not affect accident initiators or precursors, nor does it alter design assumptions that could increase the probability or consequences of a previously evaluated accident.

When the exemptions become effective, there will be no credible events which would result in doses to the public beyond the Exclusion Area Boundary (EAB) that exceed the Environmental Protection Agency (EPA) Protective Action Guidelines (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 that could create the possibility of a new or different kind of accident. 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. All the SSCs associated with limiting the release of effluents will continue to be capable of performing their design 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 change 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 IPEC Emergency Response Organization (ERO) commensurate with the reduction in consequences of radiological events that will be possible at IPEC with the facility in the permanently shutdown and defueled condition. The proposed exemptions will 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

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2. Entergy letter to NRC, "Notification of Unit 1 Transfer of 160 Spent Fuel Assemblies from the Spent Fuel Pool to the Indian Point Independent Spent Fuel Storage Installation," (ADAMS Accession No. ML083510667), dated December 11, 2008

3. Entergy letter to NRC, "Application to Revise Provisional Operating License and Technical Specifications," Indian Point Nuclear Generating Station Unit 1 (Letter No. NL-20-012), (ADAMS Accession No. ML20182A679), dated June 30, 2020

4. Entergy letter to NRC, “Notification of Permanent Cessation of Power Operations,” (Letter NL-17-021) (ADAMS Accession Number ML17044A004), dated February 8, 2017

5. NRC letter to Entergy, "Indian Point Nuclear Generating Unit Nos. 1, 2, and 3 – Issuance of Amendment Nos. 62, 293, and 268 Re: Changes to Emergency Plan for Post- Shutdown and Permanently Defueled Condition (EPID L-2019-LLA-0080)," (ADAMS Accession No. ML20078L140), dated April 15, 2020

6. Entergy letter to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No.2,"(Letter NL-20-042) (ADAMS Accession No. ML20133J902), dated May 12, 2020

7. Entergy letter to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No. 3," (Letter NL-20-033) (ADAMS Accession No. ML21131A157), dated May 11, 2021

8. 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

9. Letter from Holtec to US NRC, “Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” Attachment 1 “Holtec Topical Report, Method for Determining Spent Fuel Assembly Heat Up During a Theoretical Drain Down Event, HI-2200750(Proprietary),” (ADAMS Accession No. ML20280A524) dated September 29, 2020

10. Letter from Holtec to US NRC, “Response to Request for Additional Information—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML21148A289) dated May 28, 2021

11. Letter from Holtec to US NRC, “Revised Response to Request for Additional Information—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML21228A262) dated August 16, 2021

12. Letter from Holtec to US NRC, “Response to Request for Additional Information 10—Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report,” (ADAMS Accession No. ML21291A161) dated October 18, 2021

13. Letter from Holtec Decommissioning International, LLC (HDI) to US NRC, “Supplement to Holtec Decommissioning International, LLC (HDI) Request for Exemptions from Certain Emergency Planning Requirements of 10 CFR 50.47 and 10 CFR Part 50, Appendix E for Indian Point Unit Nos. 1, 2, and 3, Including Site-Specific Calculations” (Letter HDI-IPEC-22-013) (ADAMS Accession No. ML22032A027 & ML22032A017) dated February 1, 2022

14. NRC Letter to Holtec, “DRAFT SAFETY EVALUATION REPORT Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report, dated December 20, 2021

15. 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

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

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

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19. 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," (ADAMS Accession No. ML 13256A339), dated October 9, 2013

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

25. IP-CALC-19-00003, "Post-Permanent Shutdown Analyses of Fuel Handling, Waste Handling, and High Integrity Container Drop Accidents for Indian Point Units 2 and 3," Revision 0

26. IP-CALC-18-00066, "Shine Dose to Exclusion Area Boundary and Control Room from Spent Fuel Pool During SAFSTOR," Revision 1

27. IP-CALC-20-00022, "Evaluation of Unit 2 & Unit 3 Spent Fuel Pool (SFP) per NUREG-1738 Appendix B Seismic Checklist," Revision 0

28. NRC letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment No. 294 Re: Permanently Defueled Technical Specifications (EPID L-2019-LLA-0079)," (ADAMS Accession No. ML20081J402), dated April 28, 2020

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

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

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

33. 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," (ADAMS Accession No. ML100910006), dated May 2011

34. Engineering Report IP-RPT-12-00037, Rev. 1, Seismic Walkdown Submittal Report for Resolution of Fukushima Near Term Task Force Recommendation 2.3: Seismic, Unit 2

35. Engineering Report IP-RPT-12-00039, Rev. 1, Seismic Walkdown Submittal Report for Resolution of Fukushima Near Term Task Force Recommendation 2.3: Seismic, Unit 3

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

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