Evaluation of Request by Holtec Decommissioning International, LLC for Exemptions from Certain Emergency Planning Requirements for Indian Point Nuclear Generating Unit Nos. 1, 2, and 3

The following U.S. Nuclear Regulatory Commission (NRC) staff evaluation verifies that Holtec Decommissioning International, LLC (HDI, the licensee) provided the analyses described in Section 5.0, "Evaluation of Exemptions to EP [Emergency Planning] Regulations," of the Office of Nuclear Security and Incident Response (NSIR), Division of Preparedness and Response (DPR) Interim Staff Guidance (ISG) document NSIR/DPR-ISG-02, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," dated May 11, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ADAMS Accession No: ML14106A057). These analyses meet the criteria in the ISG to justify elimination of the requirement on the licensee to maintain the plume exposure pathway and ingestion pathway emergency planning zones (EPZs) and formal offsite radiological emergency preparedness (REP) plans. The discussion that follows lists each ISG criterion, followed by the NRC staff's evaluation of the licensee's consistency with that criterion for Indian Point Nuclear Generating Unit Nos. 1, 2, and 3 (IP1, IP2, and IP3, collectively referred to as the Indian Point Energy Center (IPEC)).

1. The licensee performed an analysis indicating that any radiological release from the applicable remaining design-basis accidents (DBAs) would be within the dose limits of section 50.67, "Accident source term," of Title 10 of the *Code of Federal Regulations* (10 CFR) and dose acceptance criteria in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No: ADAMS Accession No: ML003716792). The licensee evaluated the maximum 2-hour total effective dose equivalent (TEDE) to an individual located at the exclusion area boundary (EAB), and the 30-day TEDE to an individual at the outer boundary of the low population zone (LPZ) and the control room. The resulting doses would not approach the U.S. Environmental Protection Agency (EPA) early phase protective action guides (PAGs) recommendation for protection of the public.[1]

   Evaluation: HDI states that the irradiated fuel will be stored in the IP2 and IP3 spent fuel pools (SFPs) and an independent spent fuel storage installation (ISFSI). HDI further states, and the NRC staff agrees, that while spent fuel remains in the SFPs, the only postulated DBAs[2] that would remain applicable to the permanently defueled IPEC facility that could contribute a significant dose would be: (1) a fuel handling accident (FHA) in the fuel storage buildings (FSBs); (2) an accidental release of waste gas; and (3) an accidental release-recycle of waste liquid.

   FHA

   The NRC staff previously approved the revised DBA radiological analysis in License Amendments No. 294 (ADAMS Accession No: ML20081J402) for IP2 and License

[1] Use of the EPA early phase PAGs as a threshold is consistent with the planning basis for the 10-mile EPZ provided in NUREG-0396 (EPA 520/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (ADAMS Accession No: ML051390356), and endorsed by the Commission in a policy statement published on October 23, 1979 in the *Federal Register* (FR) ("Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents," 44 FR 61123).
[2] Since the crane used for cask handling at IP2 and IP3 is single-failure-proof, it is not necessary to postulate the drop of a spent fuel cask, as indicated in the previously approved amendments and in this exemption request.

Amendment No. 270 (ADAMS Accession No: ML21074A000) for IP3. In the information provided to support these license amendment requests (LARs), and, in the information provided as part of this exemption request, the licensee analyzed the FHA and found that the dose at the EAB following a FHA that occurred 30 days following permanent cessation of power operations is 0.47 rem, without crediting mitigation by any active safety systems or components.

Based on the permanent shutdown of IP3 on April 30, 2021, the 15-month decay time elapsed on August 1, 2022. The NRC staff notes that the doses from an FHA are dominated by the isotope Iodine-131. After 15 months of decay, the thyroid dose from an FHA would be negligible. With 15 months of decay, the only isotope remaining in significant amounts, among those postulated to be released in a DBA FHA, would be Krypton-85. Because Krypton-85 primarily decays by beta emission, the calculated skin dose from an FHA release would make an insignificant contribution to the TEDE, which is the parameter of interest in the determination of the EPA early phase PAGs for sheltering or evacuation.

Accidental Release of Waste Gas

HDI performed an analysis that 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. 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 calculated radiological consequences following a waste gas decay tank rupture without credit for any mitigating systems or the primary auxiliary building ventilation system post shutdown are as follows:

- In either control room – a whole body dose of 0.77 rem, which is under the limit of 5 rem;
- At the EAB – a whole body dose of 0.30 rem, which is under the limit of 0.5 rem; and
- At the LPZ – a whole body dose of 0.11 rem, which is under the limit of 0.5 rem.

HDI reevaluated the dose from an accidental release of waste gas to reflect the removal of the waste gas decay tank(s) from operation and to reevaluate the dose at 15 months after the shutdown of IP3. Based on the revised analysis, the radiological consequences of a postulated waste gas decay tank rupture was determined to be negligible because the tanks are removed from operation, and depressurized and vented to atmosphere.

Accidental Release-Recycle of Waste Liquid

Section 6.4, “Accidental Release-Recycle of Waste Liquid,” of the IP2 and IP3 Defueled Safety Analysis Reports (DSARs) (ADAMS Accession Nos: ML20259A199 and ML21270A059, respectively) addresses the accidental release of waste liquid by stating that the hazard from these releases is derived only from any volatized components. Thus, the release of liquid waste is evaluated in the analysis for an accidental release of waste gas.

The NRC staff reviewed the consequences of a FHA, waste gas release accident, and liquid tank failure accident in detail during the review of the previously approved license amendment requests and found them to be acceptable. Since this information has not changed for this exemption request the NRC staff relied on these previous LARs to conduct the review of this exemption request. The NRC staff notes that while the applicant continues to rely on the information from the previously approved LARs, the calculated doses would be expected to be lower when the exemption is implemented, due to additional decay time beyond the time assumed for the approved LARs. Since the dose at the EAB will not exceed the 1 rem limit, the NRC staff finds it acceptable to support approval of the exemption request.

2. The licensee performed an analysis demonstrating that after the spent fuel has decayed for 15 months, with a complete loss of SFP water inventory with no heat loss (i.e., adiabatic heat-up), a minimum of 10 hours would be available before any fuel cladding temperature reaches 900 degrees Celsius (°C) from the time all cooling is lost.

   Evaluation: The NRC staff evaluates the ability to mitigate beyond-design-basis events considering the time available to implement measures to maintain the fuel cool or, if necessary, implement an appropriate emergency response. The NRC staff uses an assessment of the adiabatic heat-up to determine the available time because adiabatic heat-up is generally the limiting condition. The heat-up time is calculated as the time to reach a temperature of 900°C, which correlates to 1,652 degrees Fahrenheit (°F) and the temperature where “runaway oxidation” (zirconium cladding fire) is expected to occur, as defined in NUREG1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” dated February 2001 (ADAMS Accession No: ML010430066).

   The 10-hour criterion, conservatively, does not consider the time to uncover the fuel and assumes instantaneous loss of cooling to the fuel. The 10-hour time period is also not intended to represent the time that it would take to repair all key safety systems or to repair a large SFP breach. The 10-hour criterion is a conservative period of time in which pre-planned mitigation measures to provide makeup water or spray to the SFP can be reliably implemented before the onset of a zirconium cladding ignition. In addition, in the unlikely event that a release is projected to occur, 10 hours would provide sufficient time for offsite agencies, if deemed warranted, to take appropriate action to protect the health and safety of the public.

   An analysis was performed by HDI which shows that 15 months after shutdown the spent fuel stored in the SFP will have decayed to the extent that the requested exemptions may be implemented at the IPEC without any additional compensatory actions. Given the permanent shutdown date of IP3 of April 30, 2021, and the fuel decay time of 15 months, the period in which the spent fuel could heat up to clad ignition temperature within 10 hours under adiabatic conditions ended on August 1, 2022. This analysis, “Holtec Spent Fuel Pool Heat Up Calculation Methodology Topical Report, Revision 2,” dated December 22, 2021 (ADAMS Accession No: ML21357A005 [non-public]) was submitted by HDI in support of the exemptions from certain EP requirements requested by letter dated December 22, 2021 (ADAMS Accession No: ML21356B693). HDI provided further information in Enclosure 1, “Indian Point Unit Nos. 2 and 3 Spent Fuel Pool Heat Up Calculations,” to HDI’s supplemental letter dated February 1, 2022 (ADAMS Accession No: ML22032A117 [non-public]). The analysis determined the decay time necessary to ensure at least a 10-hour heat-up time

considering the thermal capacity of the portion of the fuel assembly that heats uniformly and the decay heat rate of the fuel. The HDI analysis shows that after the spent fuel has decayed for 15 months, for beyond-design-basis events where the SFP is drained and air cooling is not possible, at least 10 hours would be available from the time spent fuel cooling is lost until the hottest fuel assembly reaches a temperature of 900°C. This 10-hour minimum threshold provides sufficient time for the licensee to take mitigative actions, or if government officials deem warranted, for offsite protective actions to be initiated using a comprehensive approach to EP.

The NRC staff reviewed the calculation to verify that important physical properties of materials were within acceptable ranges and that the results were accurate. The staff determined that physical properties were appropriate and completed independent confirmatory calculations that produced similar results. Therefore, the staff found that after 15 months of decay, at least 10 hours would be available before a significant offsite release could begin. The staff concluded that the adiabatic heat-up calculation provided an acceptable method for determining that a minimum of 10 hours would be available before any fuel cladding temperature reaches 900°C from the time all cooling is lost.

3. The licensee performed an analysis for a loss of SFP water inventory resulting in radiation exposure at the EAB and the control room (which indicates that any release would be less than the EPA early phase PAGs at the EAB).

   Evaluation: NUREG-0586, “Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,” Supplement 1 (ADAMS Accession No: ML023470327), section 4.3.9, “Radiological Accidents,” identifies that a SFP drain down event is beyond design basis. The SFP water and the concrete pool structure serve as radiation shielding. A loss of water shielding above the fuel could increase the offsite radiation levels because of the gamma rays streaming up out of the pool and being scattered back to a receptor at the site boundary. The radiation that is scattered due to interactions with air is sometimes referred to as sky-shine.

   Therefore, 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 primary purpose of this calculation is to determine the dose rates as a function of time at the EAB and in the control room due to loss of shielding for an event in which the spent fuel assemblies are uncovered following drain down. The dose rates determined by this calculation are due to direct and indirect radiation from spent fuel assemblies and does not consider a potential fire in the SFP for reasons discussed in the previous section above.

   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. 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 SFPs would be less than the EPA PAGs.

   Based on an annual analysis, HDI determined that 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 permanent shutdown are less than 11.55 millirem/hour and 0.0259 millirem/hour,

respectively. HDI concluded that the extended time 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.

The NRC staff notes that while the direct dose rate above the unshielded fuel would be high, radiation protection personnel would restrict access to ensure that no one was subjected to the direct dose from the unshielded fuel. Therefore, the primary concern becomes the dose rate from gamma and neutron radiation that is scattered from interactions with the air above the IP2 and IP3 SFPs. The licensee used appropriate methods to evaluate the effects of this source of radiation at the EAB and in the IP2 and IP3 control rooms. The analysis assumed one year of radioactive decay following operations. This is conservative and acceptable because it will be more than one year following shutdown of both units when the exemption is implemented.

The NRC staff reviewed the licensee's analysis description, performed an independent evaluation and agrees that appropriate methods were used to evaluate the effects of this source of radiation at the control room and the EAB. Therefore, the NRC staff concludes that the dose consequence from skyshine emitted from the SFP due to a loss of SFP normal cooling would not exceed a level that would warrant protective actions under the EPA early phase PAGs.

4. Considering the site-specific seismic hazard, the licensee has performed either an evaluation demonstrating a high confidence of a low probability (less than 1×10^{-5} per year) of seismic failure of the SFP storage structure, or an analysis demonstrating that the fuel has decayed sufficiently such that natural air flow in a completely drained pool would maintain peak cladding temperature below 565˚C (the point of incipient cladding damage).

   Evaluation: In 2012, the licensee of the IPEC (Entergy) conducted a seismic evaluation in response to an NRC letter to all power reactor licensees, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ADAMS Accession No: ML12053A340). This evaluation provided an assessment of earthquake probabilities at potentially damaging accelerations. The NRC accepted the results of this assessment indicating that the low seismic hazard screening criteria had been satisfied at IP2 and IP3 by letters dated March 20, 2014 (ADAMS Accession No: ML14071A198), and March 6, 2014 (ADAMS Accession No: ML14065A272), respectively.

   HDI developed an analysis demonstrating successful completion of the Enhanced Seismic Checklist provided in NUREG-1738 for the IP2 and IP3 SFPs demonstrating a high confidence of a low-probability (less than 1×10^{-5} per year) of seismic failure of the SFP structures. This analysis is summarized in Table 6, "Seismic Checklist for Commercial Nuclear Power Plants During Decommissioning," of the enclosure to HDI's letter dated December 22, 2021 (ADAMS Accession No: ML21356B693).

   Therefore, the NRC staff found reasonable assurance that Criterion 4 of NSIR/DPR-ISG-02 is satisfied with respect to demonstrating a high confidence in a low probability of seismic failure for the IPEC FSBs, including the SFP structures.

5. If the licensee is storing fuel in an SFP, the licensee should address for the decommissioning site the risk reduction measures identified in NUREG-1738 as industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs).[3] The IDCs and SDAs are a set of design characteristics and operational capabilities that either help prevent a substantial loss of coolant inventory or increase the likelihood of recovery from such an event.

   Evaluation: In accordance with the safety analysis in NUREG-1738, the beyond-design-basis event sequences that dominate risk at a decommissioning nuclear power reactor are large earthquake and cask-drop events. This is an important difference relative to an operating nuclear power reactor, where typically a large number of different initiating events make significant contributions to risk.

   Assurance that the results of the NUREG-1738 analysis are representative of the plant specific conditions at IPEC can be established by assessing the facility against certain design and operational characteristics that were assumed in the NUREG-1738 analysis. These characteristics were identified in the NUREG-1738 study as recovery, mitigation, and emergency response activities assumptions that were relied on to evaluate the likelihood of success in event sequences. In Section 5.4, “Comparison to NUREG-1738 Industry Decommissioning Commitments and Staff Decommissioning Assumptions,” of the enclosure to its letter dated December 22, 2021, HDI described the conformance of the IPEC facility and operations with the IDCs and the SDAs. In its discussion of the IDCs and SDAs, HDI addressed measures in place to minimize the potential risk from event sequences that dominate risk at a decommissioning reactor with fuel stored in an SFP (e.g., those IDCs and SDAs related to fuel cask handling activities and seismic events).

   The NRC staff evaluation focused on the licensee’s conformance with IDCs and SDAs that are related to the design and operation of structures, systems, and components associated with the IPEC SFPs. The summary below of the NRC staff’s findings is based on an assessment of the licensee’s IDC and SDA items:

   IDC #1: Cask-drop analyses will be performed or single-failure-proof cranes will be used for handling of heavy loads (i.e., phase II of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A36,” dated July 1980 (ADAMS Accession No: ML070250180) will be implemented).

   Evaluation: HDI states that 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 American Society of Mechanical Engineers NOG-1-2004, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” Appendix C of NUREG-0612 and NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants” (ADAMS Accession No: ML110450636). These single-failure-proof cranes are used to support spent fuel cask handling activities at IP2 and IP3.

[3] The Nuclear Energy Institute proposed IDCs in a letter to the NRC dated November 12, 1999 (ADAMS Accession No: ML993340413). The NRC identified several additional SDAs through the NRC staff’s risk assessment and evaluation of the safety principles in RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” for decommissioning plants. The IDCs and SDAs are summarized in Tables 4.2-1 and 4.2-2 to NUREG-1738.

Because the cranes are single-failure-proof, an accidental load drop is considered not to be a credible event such that condition 5.1.2(1) of NUREG-0612 is satisfied and analysis of cask-drop accidents in accordance with condition 5.1.2(4) of NUREG-0612 is not required. Therefore, the NRC staff finds that HDI satisfies NUREG-1738 IDC #1.

IDC #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.

Evaluation: HDI maintains IPEC procedures to ensure onsite and offsite resources can be brought to bear during an event. 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.

HDI states that 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 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 (ADAMS Accession No: ML19333B868). Periodic Emergency Plan drills and exercises are conducted with opportunities for offsite response organization (ORO) participation, to maintain proficiency in response to a plant event.

Therefore, the NRC staff concludes that HDI has adequate procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event to satisfy the conditions assumed in the NUREG-1738 analysis.

IDC #3: Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.

Evaluation: HDI maintains IPEC procedures that provide guidance to establish and maintain communications between onsite and offsite organizations during severe weather and seismic events. These procedures provide direction for additional actions and communications with onsite and offsite stakeholders if the event does not reach the threshold for entry into an emergency classification. If the severity of the event requires entry into an emergency classification, communications with onsite and offsite organizations will be directed by the IPEC Emergency Plan and associated procedures. The procedures (or equivalent) will be updated as necessary to reflect the permanently shutdown and defueled condition. These procedures 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. Therefore, it is not necessary for the communications procedures to be specifically referenced in the Emergency Plan. Equipment requirements are specified in the pertinent procedures.

Therefore, the NRC staff concludes that HDI has adequate procedures that provide guidance to establish and maintain communications between onsite and offsite

organizations during severe weather and seismic events to satisfy the conditions assumed in the NUREG-1738 analysis.

IDC #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.

Evaluation: HDI states that the IPEC has procedures in place to ensure that onsite and offsite resources can be brought to bear during an event. Once the IPEC is permanently shut down and defueled, the on-shift plant operators, including CFHs and non-certified operators, will be 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 the 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.

HDI maintains IPEC procedures that provide guidance to establish and maintain communications between onsite and offsite organizations during severe weather and seismic events. The IPEC has multiple portable pumps and portable emergency generators that meet the Extensive Damage Mitigation Guidelines. In addition, the IPEC Emergency Plan provides guidance for communicating with and obtaining offsite resources.

HDI maintains procedures and strategies for the movement of any necessary portable equipment that will be relied upon for mitigating the loss of SFP water. Events involving a loss of SFP cooling and/or water inventory can be addressed by implementation of SFP inventory makeup strategies required under 10 CFR 50.155(b)(2), “Mitigation of beyond-design-basis events.” These capabilities are maintained as a license condition. These diverse strategies provide defense-in-depth 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 beyond design basis events affecting the SFPs. This portable equipment can be used as required by abnormal procedures and event-based procedures may be used to support mitigation strategies for SFP damage and water supply.

Therefore, the NRC staff concludes that HDI has adequate procedures regarding effective use of onsite and offsite resources to respond to events affecting the SFPs to satisfy the conditions assumed in the NUREG-1738 analysis.

IDC #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.

Evaluation: HDI states that 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 NRC’s 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 the normal level, which is 93 feet-9 inches. The IP2 SFP

temperature is locally indicated by TIC-651 and high temperature is alarmed by TIC-651 in the IP2 control room at 125 ºF.

IP3 SFP water level is monitored via two independent level channels that were added to meet the NRC's post-Fukushima requirements. LS-6500A and LS-6500B indicate remotely in the IP3 plant auxiliary building (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-level 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 135 ºF in the IP3 control room.

Additionally, 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.

Therefore, the NRC staff concludes that HDI will maintain adequate SFP monitoring instrumentation to satisfy the conditions assumed in the NUREG-1738 analysis regarding monitoring events affecting the SFP.

IDC #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.

Evaluation: HDI states that neither SFP (IP2 or IP3) has gates with seals that could lead to fuel uncovery. However, a gate isolates the SFPs 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 seal 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.

The NRC staff finds that the described design features that limit the potential for drainage through the fuel transfer system and SFP cooling system are consistent with the assumptions used in the analysis presented in NUREG-1738.

IDC #7: Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of antisiphon devices will be periodically verified.

Evaluation: HDI states that design features and administrative controls that 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 a

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.

The IP2 DSAR section 3.3.3.2.2 “Spent Fuel Pit Cooling Loop” states in part: “In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit.” The IP3 DSAR section 3.5 “Fuel Handling System” reads in part:

> 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 accidentally;
> 2) The spent fuel pit pump discharge pipe terminates in the pool at elevation 74’ – 4 ¾”. This elevation is approximately five (5) feet above the top of the spent fuel assemblies; therefore, this pipe could not accidentally become a siphon to uncover the fuel;
> 3) The skimmer pump takes suction from, and discharges to the surface of the spent fuel pit; therefore, it could not accidentally 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, and
> 5) The spent fuel pit cooling loop was designed to seismic Class II/III and the cleanup equipment 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 NRC staff concludes that the physical configuration of inlet and outlet connections and use of anti-siphon devices provide adequate control to minimize the potential for rapid drainage through permanent systems and are consistent with the assumptions used in the analysis presented in NUREG-1738.

IDC #8: An onsite restoration plan will be in place to provide repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote alignment of the makeup source to the SFP without requiring entry to the refuel floor.

Evaluation: HDI states that repairs to equipment designated for the SFPs will be performed using the normal online work management system (or equivalent). 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 SFP buildings, and fire water using a temporary diesel pump from outside of the SFP buildings.

Procedures will remain in place to perform filling and loss of cooling recovery of the SFPs during an abnormal loss of cooling or level, as well as filling the SFPs in the event that access to the SFP floor is inaccessible. HDI states that there are multiple ways to add makeup water to the IP2 and IP3 SFPs with or without entry to the refuel floors. Sources of makeup to the IP2 and IP3 SFPs include the PWST water, fire water inside the SFP buildings, and fire water using a temporary diesel pump from outside of the SFP buildings.

The NRC staff finds that the planned SFP cooling and make-up water capability, with access to numerous sources of make-up inventory, conforms to the capabilities assumed for the NRC staff analysis presented in NUREG-1738.

IDC #9: Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

Evaluation: HDI states that 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.

IP2 procedure 2-DCS-009-GEN 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.

IP2 DSAR section 3.5.5.4 “Response to NUREG 0612, Phase I Elements” reads in part:

> Fuel Storage Building Ederer Crane
>
> The Ederer 110-ton design rated gantry crane is used to move spent fuel casks up to 110 tons into and out of the spent fuel pit by lifting a fully loaded Holtec HI-TRAC® 100 spent fuel transfer cask and its associated components. The HI-STORM® cask system utilizes the HI-TRAC® 100 transfer cask for transporting a MPC from the spent fuel pit, and for inter-cask MPC transfers required for on-site storage. However, this crane is restricted from handling casks over spent fuel in the spent fuel pit and will only be utilized for other loading activities in the FSB.
>
> Fuel Handling Crane
>
> No object weighing more than 2,000 pounds may be moved over any region of the spent fuel pit when the pit contains spent fuel, unless a technical analysis has been performed consistent with the requirements of NUREG-0612 establishing

> the necessary controls to assure that a load drop accident could damage no more than a single fuel assembly. Administrative and procedural controls to protect fuel and fuel racks may include path selection to prevent loads from passing over or near fuel. For cases in which very heavy loads (>30,000 pounds) are transported over the spent fuel pit, the loads cannot under any circumstances pass over irradiated fuel. In all cases where loads >2,000 pounds are carried over the pit, the ventilation system must be functional.

IP3 procedure 3-SOP-CM-002, "Fuel Storage Building Crane Operation," delineates the specific path that 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. IP3 DSAR section 3.10.4.1, "Response to NUREG 0612, Phase I Elements" reads in part:

> …the DSAR and facility procedures prevent any object weighing more than 2,000 pounds from being moved over any region of the spent fuel pit containing irradiated fuel, unless a technical analysis has been performed consistent with the requirements of NUREG-0612 establishing the necessary controls to assure that a load drop accident could damage no more than a single fuel assembly. Administrative and procedural controls to protect fuel and fuel racks may include a path selection to prevent loads from passing over or near fuel. For cases in which very heavy loads (>30,000 pounds) are transported over the spent fuel pit, the loads cannot under any circumstances pass over irradiated fuel. In all cases where loads >2,000 pounds are carried over the pit, the ventilation system must be functional.

Procedures will remain in place to perform filling and loss of cooling recovery of the SFPs during an abnormal loss of cooling or level, as well as filling the SFPs in the event that access to the SFP floor is inaccessible.

The NRC staff finds that the described procedures are consistent with the administrative controls considered in the NRC staff analysis presented in NUREG-1738.

IDC #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.

Evaluation: HDI states that 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.

The NRC staff finds that the described administrative controls conform to those considered in the NRC staff analysis presented in NUREG-1738.

SDA #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.

Evaluation: Section 1.7, "Design Criteria for Structures and Components," of the IP2 and IP3 DSARs states that the SFPs are designed as Seismic Category I structures, i.e., designed to withstand a safe shutdown earthquake. Both units' SFP cooling system designs have two independent trains of SFP cooling. Each train rejects its heat to the component cooling water system at each unit, and its heat, in turn, is rejected to the service water system at each unit, with its heat being rejected to the Hudson River.

The NRC staff finds that the described cooling and makeup capabilities are comparable to the capabilities considered in the NRC staff analysis presented in NUREG-1738.

SDA #2: Walkdowns 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.

Evaluation: HDI states that currently a walkdown of the SFP systems at IPEC is performed each shift (i.e., twice per day) and that 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 and permanent defueling of IP2 and IP3.

Procedures will remain in place to perform filling and loss of cooling recovery of the SFPs during an abnormal loss of cooling or level, as well as filling the SFPs in the event that access to the SFP floor is inaccessible. IP2 DSAR section 3.3.3.2.2 "Spent Fuel Pit Cooling Loop" states that with a complete loss of Spent Fuel Pit cooling and with no heat removal, the time for the spent fuel pit water to rise from 180°F to 212°F with a full core in storage is at least 1.8 hours. Makeup water can be supplied within this time from the PSWT and/or the fire protection system. The maximum required makeup rate for boiloff is 62 gallons per minute (gpm) (for a full core).

IP3 DSAR Table 3.3-2 "Spent Fuel Cooling Loop Component Data" indicates that in the absence of spent fuel pit cooling due to pit water inertia and no heat removal it could take as long as 8 ½ hours to heat from 150°F to 212°F. section 3.2.1 "System Design and Operation" indicates that the PWST has a volume of 165,000 gallons. DSAR Table 3.2-2 "Chemical And Volume Control System Principal Component Data Summary" indicates that the two Primary Water Makeup Pumps each have a capacity 150 gpm at a design head of 235 feet.

The NRC staff finds that the monitoring of the SFP systems is consistent with the NRC staff analysis presented in NUREG-1738 based on the improvements in SFP monitoring capability and reliability implemented since the publication of NUREG-1738, specifically in response to the events at Fukushima Dai-ichi in 2011.

SDA #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 an emergency.

Evaluation: HDI states that the IP2 SFP water level is monitored via two independent level channels that were added to meet the NRC's 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 is alarmed by TIC-651 in the IP2 control room at 125ºF.

The NRC staff finds that the SFP monitoring capability is consistent with the assumptions in the analysis presented in NUREG-1738.

SDA #4: The licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.

Evaluation: HDI states that 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.

The NRC staff concludes that the SFP design protections against drainage are consistent with the assumptions used in the analysis presented in NUREG-1738.

SDA #5: Load drop consequence analysis 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 enough confidence in the facility's ability to withstand a heavy load drop.

Evaluation: HDI states that 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.

Since the subject cranes are single-failure-proof, analysis of cask drop accidents is not required per condition 5.1.2(4) of NUREG-0612. The staff concurs with HDI's conclusion that a performance of load drop consequence analyses is not required.

The NRC staff finds that HDI's protection against heavy load drops is consistent with the assumptions considered in the analysis presented in NUREG-1738.

SDA #6: Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to NUREG1738. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant-specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in NUREG-1738 ($<1 \times 10^5$ per year including non-seismic events).

Evaluation: The licensee of the IPEC at the time (Entergy) conducted a seismic evaluation in response to an NRC letter to all power reactor licensees, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012. This evaluation provided an assessment of earthquake probabilities at potentially damaging accelerations. The NRC accepted the results of this assessment indicating that the low seismic hazard screening criteria had been satisfied at IP2 and IP3 by letters dated March 20, 2014, and March 6, 2014, respectively.

HDI developed an analysis demonstrating successful completion of the Enhanced Seismic Checklist provided in NUREG-1738 for the IP2 and IP3 SFPs demonstrating a high confidence of a low-probability (less than 1×10^{-5} per year) of seismic failure of the SFP structures. This analysis is summarized in Table 6 of the enclosure to HDI's letter dated December 22, 2021.

Therefore, the NRC staff finds that HDI satisfies NUREG-1738 SDA #6.

SDA #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.

Evaluation: HDI states that 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 that 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.

Therefore, the NRC staff finds that HDI satisfies NUREG-1738 SDA #7.

Based on the above evaluations, the NRC staff concludes that the design and operation of structures, systems, and components associated with SFP storage provide for safe storage of spent fuel and are consistent with the capabilities assumed in the analysis presented in NUREG-1738.

In addition to an evaluation against the specific NSIR/DPR-ISG-02 criteria above, Table 1, "Evaluation of Specific Exemptions to Emergency Planning Requirements," provides the NRC staff's evaluation of HDI's specific exemptions, shown as "strikethrough" text, requested from the requirements of 10 CFR 50.47, "Emergency plans," and Appendix , "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," based on the justification provided by HDI and evaluation criteria above.

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
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:
NRC Staff's Evaluation: The NRC requires a level of licensee EP commensurate with the potential consequences to public health and safety, and common defense and security at the licensee's site. HDI's exemption request included radiological analyses to show that, as of 30 days after the permanent cessation of power operations, the radiological consequences of the only remaining applicable DBAs would not exceed the limits of the EPA early phase PAGs at the EAB. HDI also performed an analysis which shows that, 15 months after the shutdown of IP3, the spent fuel stored in the SFPs will have decayed to the extent that in the unlikely event all cooling is lost to the spent fuel and a heat up under adiabatic conditions resulted, 10 hours would be available to take mitigative actions before the hottest fuel assembly reached 900˚C. NUREG-1738, and enhancements put into place as a result of the events of September 11, 2001, and the Fukushima Dai-ichi accident, support the NRC staff assumption that: only a highly unlikely, beyond-design-basis event (e.g., extreme earthquake or large aircraft impact) could result in an SFP fire. In addition, there would be a significant amount of time between the initiating event and the possible onset of conditions that could result in an SFP zirconium cladding fire. This time provides a substantial opportunity for event mitigation. Licensees are required to maintain effective strategies, sufficient resources, and adequately trained personnel to mitigate such an event. If State or local governmental officials determine that offsite protective actions are warranted, then sufficient time and capability would be available for ORO to implement these measures using a comprehensive emergency management plan or "all hazards" approach. Considering the very low probability of beyond-design-basis events affecting the SFPs and with the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel, and before the onset of a postulated zirconium cladding fire), formal offsite REP plans (in accordance with 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness,") are not considered necessary for a permanently shutdown and defueled nuclear power reactor.
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

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.
NRC Staff's Evaluation: NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," provides that emergency response plans should be useful for responding to any accident that would produce offsite radiological doses in excess of the EPA early phase PAGs. Additionally, it introduced the concept of generic plume exposure pathway zones as a basis for the planning of response actions, which would result in dose savings in the environs of nuclear facilities in the event of a serious power reactor accident. As previously discussed, HDI has provided radiological analyses, which show that as of 30 days after permanent cessation of power operations, the radiological consequences for the remaining applicable DBAs at IPEC will not exceed the limits of the EPA early phase PAGs at the EAB. In addition, reactor core melt (Class 9) scenarios, which were also considered in NUREG-0396, are no longer applicable to a permanently shutdown and defueled power reactor. Considering the very low probability of beyond-design-basis events affecting SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), formal offsite REP plans (in accordance with 44 CFR Part 350) are not considered necessary for a permanently shutdown and defueled nuclear power reactor. Therefore, designated plume exposure and ingestion pathway EPZs are no longer needed.
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.
NRC Staff's Evaluation: With the termination of reactor power operations at IPEC and the permanent removal of the fuel from the reactor vessels to the SFPs, most of the accident scenarios postulated for operating reactors are no longer possible. The spent fuel will be stored in the SFPs and the ISFSI and will remain onsite until it can be moved offsite for long-term storage or disposal. The reactor, reactor coolant system (RCS), and secondary systems are no longer in operation and have no function related to the storage of the spent fuel. Therefore, postulated accidents involving failure or malfunction of the reactor, RCS, or supporting systems are no longer applicable. During reactor decommissioning, the principal public safety concerns involve the radiological risks associated with the storage of spent fuel onsite. The emergency operations facility (EOF) is a support facility for the purpose of managing the overall licensee emergency response (including coordination with Federal, State, and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective actions. Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), formal

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
offsite REP plans (in accordance with 44 CFR Part 350) are not considered necessary for a permanently shutdown and defueled nuclear power reactor. Therefore, an EOF would not be needed to coordinate these types of assessments for determining public protective actions. 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. Onsite staff will continue to maintain and provide for communication and coordination capabilities with offsite authorities for the purpose of notification and for the level of support required for the only remaining applicable DBAs and the prompt implementation of mitigative actions in response to an event affecting the SFPs.
10 CFR 50.47(b)(4): A standard emergency classification and action level scheme, the basis 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~~.
NRC Staff's Evaluation: Considering the very low probability of beyond design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), formal offsite REP plans (in accordance with 44 CFR Part 350) are not needed. The IPEC Emergency Plan will continue to maintain arrangements for requesting and using assistance resources from offsite support organizations. Therefore, minimum initial offsite response measures are not required.
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~~.
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, a means to provide early notification and clear instruction to the populace within a designated plume exposure pathway EPZ is no longer required.
10 CFR 50.47(b)(6): Provisions exist for prompt communications among principal response organizations to emergency personnel~~ and to the public~~.
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, the requirement to provide prompt communication to the public within a designated plume exposure EPZ in regard to initial or pre-determined protective actions is no longer needed.

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
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.
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, the requirement to provide periodic information to the public within a designated plume exposure EPZ on how they will be notified and what their initial or predetermined protective actions should be in an emergency is not needed.
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.
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, the requirement for assessing or monitoring offsite consequences beyond the EAB is not needed.
10 CFR 50.47(b)(10): A range of protective actions has been developed for ~~the plume exposure pathway EPZ for~~ emergency workers and the public. ~~In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~
NRC Staff's Evaluation: HDI's analysis demonstrated that, as of 30 days after the permanent cessation of power operations at the IPEC, no credible events within the design basis would result in doses to the public that would exceed the EPA early phase PAGs at the EAB. Therefore, EPZs beyond the EAB and the associated protective actions developed from evacuation time estimates are no longer required. Additionally, in the unlikely event of an SFP accident, the iodine isotopes, which contribute to an offsite dose from an operating reactor power accident, are not present, so potassium iodide distribution would no longer serve as an effective or necessary supplemental protective action. As such, the NRC staff concludes that HDI provides for an acceptable level of EP at the IPEC in its permanently shutdown and defueled condition, and also provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the IPEC.

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
Although formal offsite REP plans (in accordance with 44 CFR Part 350) have typically been exempted for decommissioning sites, OROs will continue to be relied upon for firefighting, law enforcement, ambulance, and medical services in support of the licensee's (onsite) emergency plan. The licensee is responsible for providing protective measures for any emergency workers responding onsite. Additionally, the licensee is responsible for control of activities within the EAB, including public access. The licensee actions that are necessary to protect the health and safety of members of the public who are in the EAB may include, but are not limited to, evacuation, sheltering, and decontamination in the unlikely event of a release of radioactive materials.
10 CFR 50.47(c)(2): ~~Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.~~ The size of the EPZs ~~also~~ may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. ~~The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.~~
NRC Staff's Evaluation: Considering the very low-probability of beyond-design-basis events affecting the SFPs and with the time available to initiate mitigative actions consistent with plant conditions (i.e., between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), formal offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, an EPZ is not required. Section 50.47(c)(2) and footnote 1 to Appendix E to 10 CFR Part 50 both state, in part: "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 [megawatts] thermal." This provision is not applicable to the IPEC because it is not a gas-cooled reactor and it has permanently ceased power operations. Therefore, no exemption is required.
10 CFR Part 50, Appendix E, Section IV.1: The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery~~, and onsite protective actions during hostile action~~. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license under this Part, or for an early site permit (as applicable) or combined license under 10 CFR Part 52, shall contain information needed to demonstrate compliance with the standards described in § 50.47(b), and they will be evaluated against those standards.
NRC Staff's Evaluation: The 2011 EP Final Rule (76 FR 72560; November 23, 2011) made generically applicable the security-based response elements of NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005 (ADAMS Accession No: ML051740058). The enhancements of NRC Bulletin 2005-02 were not applicable to

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
holders of operating licenses for power reactors that had permanently ceased operations and had certified that fuel had been removed from the reactor vessel. Since HDI has certified that it has permanently ceased operations at the IPEC and that all fuel has been removed from the reactor vessels, the requirement for onsite protective actions during hostile action is not necessary for the IPEC. Additionally, the NRC excluded non-power reactors from the definition of "hostile action" at the time of the 2011 EP Final Rule because, as defined in 10 CFR 50.2, "Definitions," a non-power reactor is not considered a nuclear power reactor and a regulatory basis had not been developed to support the inclusion of non-power reactors in the definition of "hostile action." Like a non-power reactor, a decommissioning power reactor has a lower likelihood of a credible accident resulting in radiological releases requiring offsite protective measures than does an operating power reactor. For all of the above reasons, the NRC staff concludes that a decommissioning power reactor is not a facility that falls within the definition of "hostile action." Although this analysis provides a justification for exempting the IPEC from "hostile action" related requirements, some EP requirements for security-based events are maintained. The classification of security-based events, notification of offsite authorities, and coordination with offsite agencies are still required.
10 CFR Part 50, Appendix E, Section 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.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR 50.47(b)(10).
10 CFR Part 50, Appendix E, Section 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.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2.
10 CFR Part 50, Appendix E, Section IV.4: ~~Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2
10 CFR Part 50, Appendix E, Section IV.5: ~~During the years between decennial censuses, nuclear power reactor licensees shall~~

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
~~estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2.
10 CFR Part 50, Appendix E, Section IV.6: ~~If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC under § 50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2.
10 CFR Part 50, Appendix E, Section IV.A.1: A description of the normal plant ~~operating~~ organization.
Staff's Evaluation: Because the NRC docketed the certifications of permanent cessation of operations and permanent removal of fuel from the reactor vessels, the 10 CFR Part 50 licenses for the IPEC no longer authorizes operation of the IPEC reactors, or emplacement or retention of fuel into the reactor vessels, as specified in 10 CFR 50.82(a)(2), "Termination of license." Because HDI is no longer authorized to operate the reactors, the IPEC does not have a plant "operating" organization. Rather, the site is maintained by an on-shift staff responsible for safely managing and storing spent fuel. A description of the plant organization, as it relates to the requirements in Section IV.A.1 to Appendix E of 10 CFR Part 50 is still required.
10 CFR Part 50, Appendix E, Section IV.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.~~
NRC Staff's Evaluation: The number of staff at decommissioning sites is generally small but is commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. HDI furnished information concerning its SFP inventory makeup strategies that could be used in the event of a catastrophic loss of SFP water inventory and states that designated on-shift personnel will be trained to implement such strategies with equipment maintained onsite. The on-shift individuals will be able to implement the necessary tasks within 2 hours. As such, designation of specific licensee headquarters personnel is not necessary for the augmentation of the on-shift staffing and, therefore, is not described.

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10 CFR Part 50, Appendix E, Section IV.A.4: Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making ~~offsite~~ dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.
NRC Staff's Evaluation: HDI's analyses demonstrated that, as of 30 days after permanent cessation of power operations, no DBA would result in doses in excess of the EPA early phase PAGs to the public beyond the EAB. While it is unlikely that a beyond-DBA would result in doses in excess of the EPA early phase PAGs to the public beyond the EAB, the licensee still must be able to determine if a radiological release is occurring, thereby achieving the underlying purpose of this regulatory provision. If a release is occurring, then the licensee's staff should promptly communicate that information to offsite authorities for their consideration. The offsite authorities are responsible for deciding what, if any, protective actions should be taken that they consider appropriate to protect public health and safety. Considering the very low-probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), formal offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, offsite dose projections are not required.
10 CFR Part 50, Appendix E, Section IV.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.~~
NRC Staff's Evaluation: HDI furnished information concerning its SFP inventory makeup strategies that could be used in the event of a catastrophic loss of SFP water inventory and states that designated on-shift personnel are trained to implement such strategies with equipment maintained onsite. HDI will have on-shift individuals able to implement the necessary tasks within 2 hours. As such, additional employees or other persons with special qualifications are not anticipated. Considering the very low probability of beyond-design-basis events affecting the SFP and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, personnel with special qualifications, as directed in 10 CFR Part 50, Appendix E, Section IV.A.5, are not required.
10 CFR Part 50, Appendix E, Section IV.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

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that include the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force.
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.1.
10 CFR Part 50, Appendix E, Section IV.A.8: ~~Identification of the State and/or local officials responsible for planning for, ordering and controlling appropriate protective actions, including evacuations when necessary.~~
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, identification of the State and/or local officials responsible for detailed pre-planning for, and ordering appropriate protective actions, including evacuations when necessary, is no longer required.
10 CFR Part 50, Appendix E, Section IV.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.~~
NRC Staff's Evaluation: As part of the 2011 EP Final Rule, the NRC concluded 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 similarities with regard to risk profile between the IPEC and non-power reactors show that the IPEC facility should be treated in a similar fashion as a non-power reactor for purposes of EP. Therefore, the staff concludes that a detailed staffing analysis is not needed for a decommissioning reactor.
10 CFR Part 50, Appendix E, Section IV.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.
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time

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available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, a decommissioning reactor is not required to have emergency action levels (EALs) to determine protective measures offsite. With respect to EALs for hostile action, refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.1.
10 CFR Part 50, Appendix E, Section IV.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 [Federal Emergency Management Agency]-REP-1.
NRC Staff's Evaluation: For a permanently shutdown and defueled power reactor, containment pressure and emergency core cooling system are no longer required. Therefore, they would have no parameters indicating a potential emergency. Other indications, such as SFP level, SFP temperature, and area radiation monitors indicate the conditions at the IPEC. HDI's analysis demonstrated that, as of 30 days after the permanent cessation of power operations, no credible events within the DBA would reach the dose criteria for the declaration of a Site Area Emergency or a General Emergency. As discussed previously, the probability of a beyond-DBA condition that could reach emergency classifications of a Site Area Emergency or a General Emergency is very low. In the unlikely event of a severe beyond-DBA resulting in the loss of all cooling to the stored fuel, as of 15 months after the permanent cessation of power operations at IP3, it would take at least 10 hours from the time the fuel is uncovered until it reaches a temperature of 900˚C. During this time, HDI could initiate mitigative actions consistent with plant conditions. The need for offsite radiation monitoring systems in support of event classification above an Alert classification level is no longer required because of the very low probability of beyond-design-basis events occurring that would affect SFP structural integrity, as well as the time available to initiate SFP mitigative measures before the onset of a postulated zirconium cladding fire.
10 CFR Part 50, Appendix E, Section IV.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

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licensee to be necessary to protect public health and safety provided that any delay in declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.
NRC Staff's Evaluation: HDI states that it will maintain the capability to assess, classify, and declare an emergency condition within 30 minutes after the availability of indications to operators that an EAL threshold has been reached. In the 2011 EP Final Rule, non-power reactor licensees were not required to assess, classify, and declare an emergency condition within 15 minutes. Like non-power reactors, a decommissioning power reactor has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For these reasons, the NRC staff concludes that a decommissioning power reactor should not be required to assess, classify, and declare an emergency condition within 15 minutes.
10 CFR Part 50, Appendix E, Section IV.D.1: Administrative and physical means for notifying local, State, and Federal officials and agencies ~~and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary,~~ shall be described. This description shall include identification ~~of the appropriate officials, by title and agency,~~ of the State and local government agencies ~~within the EPZs~~.
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR 50.47(b) and 10 CFR 50.47(b)(10).
10 CFR Part 50, Appendix E, Section IV.D.2: ~~Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.D.1.
10 CFR Part 50, Appendix E, Section IV.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 informed by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at the 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 with 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 notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate~~

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~~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.~~
NRC Staff's Evaluation: HDI proposed in its exemption request to complete emergency notifications within 60 minutes after an emergency declaration or a change in classification to the State of New York and local government agencies. This timeframe is consistent with the 10 CFR 50.72(a)(3), "Immediate notification requirements for operating nuclear power reactors," notification time to the NRC and is appropriate because in the permanently defueled condition, the rapidly developing scenarios associated with events initiated during reactor operation are no longer credible. Also refer to basis for exemption from 10 CFR 50.47(b) and 10 CFR 50.47(b)(10).
10 CFR Part 50, Appendix E, Section IV.D.4: ~~If FEMA has approved a nuclear power reactor site's alert and notification design report, including the backup alert and notification capability, as of December 23, 2011, then the backup alert and notification capability requirements in Section IV.D.3 must be implemented by December 24, 2012. If the alert and notification design report does not include a backup alert and notification capability or needs revision to ensure adequate backup alert and notification capability, then a revision of the alert and notification design report must be submitted to FEMA for review by June 24, 2013, and the FEMA-approved backup alert and notification means must be implemented within 365 days after FEMA approval. However, the total time period to implement a FEMA-approved backup alert and notification means must not exceed June 22, 2015.~~
NRC Staff's Evaluation: Refer to the basis for exemption from 10 CFR Part 50, Appendix E, Section IV.D.3 regarding the alert and notification system requirements.
10 CFR Part 50, Appendix E, Section IV.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;
NRC Staff's Evaluation: The technical support center (TSC) is an area located close to the control room that provides plant management and technical support to the reactor operating personnel located in the control room during emergency conditions. It has technical data displays and plant records available to assist in the detailed analysis and diagnosis of abnormal plant conditions and any significant release of radioactivity to the environment. The TSC is also the primary communications center for the plant during an emergency. With the permanently shutdown and defueled status of the IPEC and the storage of the spent nuclear fuel in the SFPs and ISFSI, the TSC and EOF will no longer be required to meet their original purpose during an emergency or support

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initial SFP mitigation actions if needed. The basis for the EOF exemption is provided in the basis for exemption from 10 CFR 50.47(b)(3).
10 CFR Part 50, Appendix E, Section IV.E.8.a.(ii): ~~For nuclear power reactor licensees, a licensee onsite operational support center;~~
NRC Staff's Evaluation: The operations 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. The OSC should provide a location where plant logistic support can be coordinated during an emergency and restrict control room access to those support personnel specifically requested by the shift supervisor. With the permanently shutdown and defueled status of the IPEC and the storage of the spent nuclear fuel in the SFPs and ISFSI, an OSC will no longer be required to meet its original purpose during an emergency or support initial SFP mitigation actions if needed. HDI states that an onsite facility will continue to be maintained, from which effective direction can be given and effective control may be exercised during an emergency.
10 CFR Part 50, Appendix E, Section IV.E.8.b: ~~For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles from a nuclear power reactor site shall request prior Commission approval by submitting an application for an amendment to its license. For an emergency operations facility located more than 25 miles from a nuclear power reactor site, provisions must be made for locating NRC and offsite responders closer to the nuclear power reactor site so that NRC and offsite responders can interact face-to-face with emergency response personnel entering and leaving the nuclear power reactor site. Provisions for locating NRC and offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:~~ (1) ~~Space for members of an NRC site team and Federal, State, and local responders;~~ (2) ~~Additional space for conducting briefings with emergency response personnel;~~ (3) ~~Communication with other licensee and offsite emergency response facilities;~~ (4) ~~Access to plant data and radiological information; and~~ (5) ~~Access to copying equipment and office supplies;~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR 50.47(b)(3).

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10 CFR Part 50, Appendix E, Section IV.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;~~ (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~~ (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~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR 50.47(b)(3).
10 CFR Part 50, Appendix E, Section IV.E.8.d: ~~For nuclear power reactor licensees, an alternative facility (or facilities) that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and collectively having the following characteristics: the capability for communication with the emergency operations facility, control room, and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation, for use when onsite emergency facilities cannot be safely accessed during hostile action. The requirements in this paragraph 8.d must be implemented no later than December 23, 2014, with the exception of the capability for staging emergency response organization personnel at the alternative facility (or facilities) and the capability for communications with the emergency operations facility, control room, and plant security, which must be implemented no later than June 20, 2012.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.1 regarding "hostile action."
10 CFR Part 50, Appendix E, Section IV.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;~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR 50.47(b)(3).
10 CFR Part 50, Appendix E, Section IV.E.9.a: Provisions for communications with contiguous State/local governments ~~within the plume exposure pathway EPZ~~. Such communication shall be tested monthly.
NRC Staff's Evaluation: HDI will maintain the 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, HDI would no longer describe provisions to 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). A description of the communications systems and the testing frequencies will be provided.

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10 CFR Part 50, Appendix E, Section IV.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.~~
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. There is no need for a TSC, EOF, or offsite field assessment teams to meet the underlying purpose of the rule. With the elimination of the requirements for a TSC, EOF, and the field assessment teams, performing annual testing of communications among them is no longer required. Communications with State and local governments will be through the commercial phone system. Due to its frequency of use, the testing of that system is not necessary.
10 CFR Part 50, Appendix E, Section IV.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.
NRC Staff's Evaluation: Based on the smaller facility staff and the greatly reduced required interaction with State and local emergency response facilities, the NRC staff concludes that the functions of the control room, EOF, TSC, and the OSC may be combined into one or more locations. As discussed previously, there is no need for the TSC and EOF. As a result, communications between the EOF and TSC and the NRC, and monthly testing of these capabilities, are no longer needed. The Emergency Notification System used to communicate with the NRC will continue to be tested monthly.
10 CFR Part 50, Appendix E, Section IV.F.1: The program to provide for: (a) The training of employees and exercising, by periodic drills, of radiation 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 radiation 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: i. Directors and/or coordinators of the plant emergency organization; ii. Personnel responsible for accident assessment, including control room shift personnel; iii. Radiological monitoring teams; iv. Fire control teams (fire brigades); v. Repair and damage control teams; vi. First aid and rescue teams; vii. Medical support personnel; viii. ~~Licensee's headquarters support personnel;~~

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ix. Security personnel. 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~~.
NRC Staff's Evaluation: Decommissioning power reactor sites typically have a level of emergency response that does not require additional response by the licensee's headquarters personnel. HDI states that 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 plant procedures and will beregularly tested through drills and exercises. Therefore, the NRC staff considers exempting the licensee's headquarters personnel from training requirements to be reasonable. Due to the low probability of DBA or other credible events to exceed the EPA early phase PAGs, offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services, as appropriate. Local news media personnel are no longer need radiological orientation training since they will not be called upon to support the formal Joint Information Center. The term "Civil Defense" is no longer commonly used, so references to this term in the regulation are not needed.
10 CFR Part 50, Appendix E, Section IV.F.2: The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, ~~test the public alert and notification system,~~ and ensure that emergency organization personnel are familiar with their duties.
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.D.1.
10 CFR Part 50, Appendix E, Section IV.F.2.a: ~~A full participation exercise 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.~~ [**F.2.a.(i), (ii), and (iii)** are not applicable.]
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, conducting a full participation exercise with State and local agencies is not required.

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However, 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 the IPEC, to the extent defined by the IPEC Emergency Plan. The licensee would be exempt from 10 CFR Part 50, Appendix E, Section IV.F.2.a.(i)-(iii) because the licensee would be exempt from the umbrella provision of 10 CFR Part 50, Appendix E, Section IV.F.2.a.
10 CFR Part 50, Appendix E, Section IV.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, and 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 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.
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.F.2.a for the basis for exemption from requirements related to offsite actions. The basis for the TSC exemption is provided in the basis for exemption from 10 CFR Part 50, Appendix E, Section IV.E.8.a.(i). The basis for the OSC exemption is provided in the basis for exemption from 10 CFR Part 50, Appendix E, Section IV.E.8.a.(ii). The basis for the EOF exemption is provided in the basis for exemption from 10 CFR 50.47(b)(3).

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10 CFR Part 50, Appendix E, Section IV.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, then each licensee shall:~~ (1) ~~Conduct an exercise biennially of its onsite emergency plan;~~ (2) ~~Participate quadrennially in an offsite biennial full or partial participation exercise;~~ (3) ~~Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~ (4) ~~Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~ (5) ~~Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.F.2.a.
10 CFR Part 50, Appendix E, Section IV.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.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2.
10 CFR Part 50, Appendix E, Section IV.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.
NRC Staff's Evaluation: 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 the IPEC, to the extent defined by the IPEC Emergency Plan. Also see the basis for exemption from 10 CFR Part 50, Appendix E, Section IV.2.
10 CFR Part 50, Appendix E, Section IV.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

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protective measures can and will be taken in the event of a radiological emergency or (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response. ~~The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.~~
NRC Staff's Evaluation: Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, the requirement to conduct a full participation exercise with State and local agencies is not needed. Because the NRC staff previously concluded that full participation emergency plan exercises are not required, and FEMA does not have responsibilities related to onsite EP, NRC consultation with FEMA is not necessary.
10 CFR Part 50, Appendix E, Section IV.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 OROs.
NRC Staff's Evaluation: For decommissioning power reactor sites, there are limited events that could occur and, as such, the purpose of ensuring that responders do not get preconditioned to certain scenarios is not necessary to achieve the underlying purpose of this rule provision. Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, drills involving principle functional areas associated with formal offsite REP are not needed.
10 CFR Part 50, Appendix E, Section IV.F.2.j: ~~(i) The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section.~~ ~~(ii) Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center.~~ ~~(iii) In each 8-calendar-year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements:~~ ~~(1) Hostile action directed at the plant site;~~ ~~(2) No radiological release or an unplanned minimal radiological release that does not require public protective actions;~~

Table 1 **Evaluation of Specific Exemptions to EP Requirements**
~~(3) An initial classification of, or rapid escalation to, a Site Area Emergency or General Emergency;~~ ~~(4) Implementation of strategies, procedures, and guidance under § 50.155(b)(2); and~~ ~~(5) Integration of offsite resources with onsite response.~~ ~~(iv) The licensee shall maintain a record of exercises conducted during each 8-year exercise cycle that documents the content of scenarios used to comply with the requirements of section IV.F.2.j of this appendix.~~ ~~(v) Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015.~~ ~~(vi) The first 8-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under 10 CFR part 52, the first 8-year exercise cycle begins in the calendar year of the initial exercise required by section IV.F.2.a of this appendix.~~
NRC Staff's Evaluation: For decommissioning power reactor sites, there are limited events that could occur and, as such, the purpose of ensuring that responders do not get preconditioned to certain scenarios is not necessary to achieve the underlying purpose of this provision. Considering the very low probability of beyond-design-basis events affecting the SFPs and the time available to initiate mitigative actions consistent with plant conditions (i.e., the time between the loss of both water and air cooling to the spent fuel and the time before the onset of a postulated zirconium cladding fire), offsite REP plans (in accordance with 44 CFR Part 350) are not needed. Therefore, drills involving principle functional areas associated with formal offsite REP are not needed.
10 CFR Part 50, Appendix E, Section IV.I: ~~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.~~
NRC Staff's Evaluation: Refer to basis for exemption from 10 CFR Part 50, Appendix E, Section IV.E.8.d.