



HDI-IPEC-22-076

November 2, 2022

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

Subject: License Amendment Request to Revise Indian Point Nuclear Generating Unit Nos. 2 and 3 Renewed Facility Licenses and Permanently Defueled Technical Specifications and IP3 Appendix C Technical Specifications to Reflect Permanent Removal of Spent Fuel from the IP2 and IP3 Spent Fuel Pits

Indian Point Nuclear Generating Unit Nos. 2 and 3  
Docket Nos. 50-247 and 50-286  
Renewed Facility License No. DPR-26  
Renewed Facility License No. DPR-64

- Reference:
- 1) Letter, Entergy to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No. 2," dated May 12, 2020 (Letter NL-20-042) (ML20133J902)
  - 2) Letter, Entergy to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No. 3," dated May 11, 2021 (Letter NL-21-033) (ML21131A157)
  - 3) Letter, NRC to Entergy, "Indian Point Nuclear Generating Station, Unit Nos. 1, 2, and 3 - Order Approving Transfer of Licenses and Draft Conforming Administrative License Amendments, (EPID-L-2019-LLM-0003)," dated November 23, 2020 (ADAMS Accession No. ML20297A321)

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Indian Point 2, LLC (IP2) and Holtec Indian Point 3, LLC (IP3), requests U.S. Nuclear Regulatory Commission (NRC) review and approval of an amendment to Renewed Facility License No. DPR-26 and Renewed Facility License No. DPR-64. The proposed License Amendment would revise the 10 CFR Part 50 Renewed Facility Licenses (FLs) and Appendix A Permanently Defueled Technical Specifications (PDTSSs) and IP3 Appendix C, "Inter – Unit Fuel Transfer Technical Specifications," (Appendix C TSs) to reflect removal of all spent nuclear fuel from the IP2 and IP3 Spent Fuel Pits (SFPs) to dry cask storage within a site controlled Independent Spent Fuel Storage Installation (ISFSI).

In References 1 and 2, Entergy certified to the NRC, in accordance with 10 CFR 50.82(a)(1)(i), that power operations ceased at IP2 on April 30, 2020, and at IP3 on April 30, 2021. In addition, Entergy certified in accordance with 10 CFR 50.82(a)(1)(ii), that the fuel was permanently removed from the IP2 reactor vessel and placed in the IP2 spent fuel pit (SFP) on May 12,

2020, and that the fuel was permanently removed from the IP3 reactor vessel and placed in the IP3 SFP on May 11, 2021. Therefore, as specified in 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for IP2 and IP3 no longer authorize operation of the reactors or emplacement or retention of fuel in the reactor vessels.

On May 28, 2021, pursuant to the NRC Order (Reference 3), Entergy notified the NRC that IPEC ownership and the IPEC operating licenses were transferred to Holtec Indian Point 2, LLC (IP1 & IP2) and Holtec Indian Point 3, LLC (IP3).

HDI expects that transfer of the spent fuel from the IP2 and IP3 SFPs to dry storage within an ISFSI will be completed before February 2023 and December 2023, respectively. In support of these conditions, revisions to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS are proposed to comport with the requirements for a facility configuration with all spent nuclear fuel in dry storage within an ISFSI.

HDI requests review and approval of the proposed license amendments by November 15, 2023, and a thirty-day implementation period following HDI's notification to the NRC that all spent fuel assemblies have been transferred from the IP2 SFP and IP3 SFP to dry storage within an ISFSI.

The Enclosure to this letter provides a description and evaluation of the proposed changes to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS. The evaluation includes the regulatory evaluation, the no significant hazards consideration determination, and the environmental considerations.

Attachment 1 to the Enclosure contains markup pages of the IP2 FL and PDTs.

Attachment 2 to the Enclosure contains markup pages of the IP3 FL, PDTs pages and Appendix C TS.

Attachment 3 to the Enclosure contains the retyped IP2 and IP3 FL and the consolidated IP2 and IP3 ISFSI Only TS pages. The IP3 Appendix C TS is proposed to be deleted in their entirety; thus, no retyped Appendix C TSs are provided.

HDI has reviewed the proposed amendments in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92, and concludes that this change does not involve a significant hazards consideration. HDI has also determined that the proposed changes satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22(c)(10) and do not require an environmental review. Therefore, in accordance with 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required.

As required by 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b) copies of this application, with the enclosure, are being provided to the New York State Department of Health and Emergency Management Agency.

New regulatory commitments for IP2 and IP3 are described in Attachment 4 to Enclosure 1 of this letter.

Should you have any questions or require additional information, please contact Mr. Walter Wittich, IPEC Licensing at 914-254-7212.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 2, 2022.

Sincerely,

Jean A. Fleming  
Vice President, Licensing, Regulatory and PSA  
Holtec International

Enclosure: Description and Evaluation of Proposed Changes - License Amendment Request to Revise Indian Point Nuclear Generating Unit Nos. 2 and 3 Renewed Facility Licenses and Permanently Defueled Technical Specifications and IP3 Appendix C Technical Specifications to Reflect Permanent Removal of Spent Fuel from the IP2 and IP3 Spent Fuel Pits

Attachment 1: Markup Pages of the IP2 FL and PDTS

Attachment 2: Markup Pages of the IP3 FL, PDTS, and Appendix C TS

Attachment 3: Retyped IP2 and IP3 FLs and Consolidated IP2 and IP3 ISFSI Only TS Pages

Attachment 4: Regulatory Commitments

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

**HDI-IPEC-22-076**

**Enclosure**

**Description and Evaluation of Proposed Changes**

License Amendment Request to Revise Indian Point Nuclear Generating Unit Nos. 2 and 3  
Renewed Facility Licenses and Permanently Defueled Technical Specifications and  
IP3 Appendix C Technical Specifications to Reflect Permanent Removal of  
Spent Fuel from the IP2 and IP3 Spent Fuel Pits

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Attachment 1: Markup Pages of the IP2 FL and PDTS

Attachment 2: Markup Pages of the IP3 FL, PDTS, and Appendix C TS

Attachment 3: Retyped IP2 and IP3 FLs and Consolidated IP2 and IP3 ISFSI Only TS Pages

Attachment 4: Regulatory Commitments

## 1.0 INTRODUCTION AND DESCRIPTION

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Indian Point 2, LLC (IP2) and Holtec Indian Point 3, LLC (IP3), requests U.S. Nuclear Regulatory Commission (NRC) review and approval of an amendment to Renewed Facility License No. DPR-26 and Renewed Facility License No. DPR-64. The proposed License Amendment would revise the 10 CFR Part 50 Renewed Facility Licenses (FLs) and Appendix A Permanently Defueled Technical Specifications (PDTs) and IP3 Appendix C, "Inter – Unit Fuel Transfer Technical Specifications," (Appendix C TSs) to reflect removal of all spent nuclear fuel from the IP2 and IP3 Spent Fuel Pits (SFPs) to dry cask storage within a site controlled Independent Spent Fuel Storage Installation (ISFSI).

In References 1 and 2, Entergy certified to the NRC, in accordance with 10 CFR 50.82(a)(1)(i), that power operations ceased at IP2 on April 30, 2020, and at IP3 on April 30, 2021. In addition, Entergy certified in accordance with 10 CFR 50.82(a)(1)(ii), that the fuel was permanently removed from the IP2 reactor vessel and placed in the IP2 SFP on May 12, 2020, and that the fuel was permanently removed from the IP3 reactor vessel and placed in the IP3 SFP on May 11, 2021. Therefore, as specified in 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for IP2 and IP3 no longer authorize operation of the reactors or emplacement or retention of fuel in the IP2 and IP3 reactor vessels.

HDI expects that transfer of the spent fuel from the IP2 and IP3 SFPs to dry storage within an ISFSI will be completed before February 2023 and December 2023, respectively. In support of these conditions, revisions to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS are proposed to comport with the requirements for a facility configuration with all spent nuclear fuel in dry storage within an ISFSI.

HDI requests review and approval of the proposed license amendments by November 15, 2023, and a thirty-day implementation period following HDI's notification to the NRC that all spent fuel assemblies have been transferred from the IP2 SFP and IP3 SFP to dry storage within an ISFSI.

The existing IP2 and IP3 PDTs and Appendix C TSs contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe storage and management of irradiated fuel with spent fuel stored in the IP2 SFP or IP3 SFP. As such, the existing PDTs and IP3 Appendix C TS provide a level of control in excess of that needed for safe storage and management of irradiated fuel with all IP2 and IP3 spent fuel stored in an ISFSI. The remaining LCOs are only applicable when irradiated fuel assemblies are stored in the IP2 SFP or IP3 SFP. Once all IP2 and IP3 spent fuel assemblies have been transferred to one of the site controlled ISFSI locations, all remaining LCOs and associated Surveillance Requirements (SRs) will no longer be applicable and are proposed for deletion.

The proposed revisions to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS reflect the removal of all spent fuel from the IP2 and IP3 SFPs and will become applicable after the last spent fuel assembly has been removed from the IP2 and IP3 SFPs and stored in an ISFSI. The revised IP2 and IP3 PDTs will be consolidated into a single document and referred to as the IP2 and IP3 ISFSI Only Technical Specifications (IOTS).

### Pending Licensing Actions under NRC Review

On August 2, 2022, (Reference 3) HDI submitted a License Amendment Request (LAR) to revise the IP2 PDTs to modify staffing requirements following the transfer of spent fuel to dry storage.

In conjunction with this LAR, HDI intends to submit two additional LARs:

1. requesting approval of a proposed revision to the Permanently Defueled Emergency Plan (DPEP) to accommodate transition to an ISFSI Only Emergency Plan (IOEP);
2. requesting approval of a proposed revision to the Indian Point Physical Security Plan (PSP) to accommodate transition to an Indian Point ISFSI Only PSP (IOPSP).

## **2.0 PROPOSED CHANGES**

This LAR modifies the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS to comport to the condition of all IP2 and IP3 irradiated fuel stored in approved dry casks within either of the site controlled ISFSI storage locations. The proposed amendment would revise the IP2 and IP3 PDTs and IP3 Appendix C TS to eliminate operational requirements and certain design requirements that involve storage and protection of spent fuel in the IP2 SFP or IP3 SFP. The proposed IP2 and IP3 PDTs revisions include a new requirement that will prevent storage of spent fuel in the IP2 SFP and IP3 SFP. The revised IP2 and IP3 PDTs will be consolidated into a single document and referred to as the IP2 and IP3 IOTS.

The proposed revisions to the IP2 and IP3 PDTs also request relocating certain administrative specifications from IP2 and IP3 PDTs Section 5, "Administrative Controls," to the IP2 and IP3 Defueled Safety Analysis Reports (DSARs), and subsequently controlling future changes to those requirements in accordance with 10 CFR 50.59.

## **3.0 TECHNICAL EVALUATION**

### **3.1 General Analysis Applicable to the Proposed Changes**

The proposed amendment would modify the existing IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS by deleting requirements that are no longer applicable to a facility due to the revised plant configuration where no spent fuel is stored in the IP2 SFP or IP3 SFP, and all spent fuel is stored in approved dry casks in site controlled ISFSI locations. The proposed changes also involve revision and relocation of existing IP2 and IP3 PDTs administrative controls that are no longer required to be retained in the IP2 and IP3 IOTS. This proposed amendment will be implemented after NRC approval and within 30 days following HDI's notification to the NRC that all spent fuel assemblies have been transferred from the IP2 SFP and IP3 SFP to dry storage within an ISFSI.

HDI plans to use a decommissioning method in which most fluid systems are drained and IP2 and IP3 are left in a stable condition until final dismantlement. Administrative controls that are required to be in place when decontamination or dismantling activities of radioactive systems, structures, and components (SSCs) are being performed, are designed to minimize the likelihood of an off-normal or accident event, and thereby the consequences of such an event. The proposed changes to the existing IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS do

not have an adverse impact on these remaining decommissioning activities or any of their postulated radiological consequences.

IP2 and IP3 spent fuel will be stored in dry casks within an ISFSI, until it is shipped off site consistent with the schedules described in the Indian Point Units 1, 2, and 3 Post-Shutdown Decommissioning Activities Reports (Reference 4) and the Fuel Loading Plan for Indian Point Unit 2 and 3 (Reference 5).

During decommissioning with all IP2 and IP3 spent fuel in dry storage within an ISFSI, there are no installed plant SSCs relied upon for the safe storage of spent fuel. In these conditions, there are no credible accidents at IP2 or IP3 whose prevention or mitigation would need to be addressed by the IOTS. In addition, the NRC approved spent fuel storage casks and canisters to be used for spent fuel storage are subject to their own 10 CFR 72 Certificate of Compliance and Cask Technical Specifications (CTS).

The IP2 and IP3 DSARs describe the design basis accidents (DBAs) related to the IP2 and IP3 SFPs, respectively. Postulated accidents were predicated on spent fuel being stored in the IP2 SFP or IP3 SFP. With the removal of the spent fuel from the IP2 and IP3 SFPs, there are no remaining spent fuel assemblies to be monitored and there are no credible accidents that require the actions of a Certified Fuel Handler, Shift Manager, or a Non-certified Operator to prevent occurrence or mitigate the consequences of an accident.

10 CFR 50.2 defines safety-related structures, systems, and components (SSCs) as those that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant boundary,
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in §50.34(a)(1) or § 100.11 of this chapter, as applicable.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to IP2 and IP3, because they are in a permanently defueled condition.

The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. After the IP2 and IP3 nuclear spent fuel assemblies have been transferred to dry cask storage within an ISFSI, there are no longer any SSCs at IP2 or IP3 that are required to be relied upon for accident prevention or mitigation.

- Section 6.2 of the IP2 and IP3 DSARs describe the DBAs related to the IP2 SFP and IP3 SFP, respectively. These postulated accidents are predicated on spent fuel being stored in the IP2 or IP3 SFPs. Thus, a spent fuel handling accident is no longer applicable as a design basis accident once all spent fuel is offloaded from the IP2 SFP and IP3 SFP and transferred to dry storage within an ISFSI.
- Section 6.4 of the IP2 and IP3 DSARs addresses why the accidental release-recycle of waste liquid is not addressed.
- Section 6.5 of the IP2 and IP3 DSARs address an accidental drop of a High-Integrity Container. Compliance with an activity limit ensures that the release resulting from the High-Integrity Container drop event remains is bounded by the consequences of the

FHA and remains below the 1 rem Environmental Protection Agency Protective Action Guideline.

The proposed deletion of requirements in the IP2 and IP3 PDTs and IP3 Appendix C TS do not involve any SSCs credited in any accident analysis at IP2 or IP3 after the transfer of spent fuel from the IP2 and IP3 SFPs to an ISFSI.

10 CFR 50.36, "Technical Specifications," promulgates the regulatory requirements related to the content of Technical Specifications. As detailed in subsequent sections of this LAR, this regulation lists four criteria to define the scope of equipment and parameters that must be included in a plant's Technical Specifications. A discussion of the applicability of these four criteria in the permanently defueled condition with all fuel removed from the SFP is provided in Section 4.1 of this enclosure. In a permanently defueled condition with all IP2 and IP3 spent fuel in dry storage within the ISFSI, the scope of equipment and parameters that need be included in the IOTS is limited to a description of the site location and high radiation area administrative controls.

The proposed changes related to the relocation of certain administrative controls from the IP2 and IP3 PDTs to the IP2 and IP3 DSARs do not affect operating procedures or administrative controls that have the function of preventing or mitigating any accidents applicable to the safe management of irradiated fuel stored in approved dry casks or decommissioning of the permanently defueled facilities.

### **3.2 Detailed Discussion**

Each proposed change to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS is described and justified in the following section of this enclosure. Changes to the existing IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS are identified in the markups shown in Attachments 1 and 2 to this Enclosure, respectively. Retyped pages showing the revised IP2 and IP3 FLs and the consolidated IP2 and IP3 IOTS are provided in Attachment 3 to this Enclosure. No retyped pages are provided for the IP3 Appendix C TS, because they are proposed to be eliminated in their entirety.

Changes to the IP2 PDTs and IP3 PDTs Table of Contents are proposed to reflect the removal of sections that will be eliminated in this LAR. These proposed changes are administrative changes, and no further description or justification is provided.

#### **3.2.1 Proposed Changes to the IP2 and IP3 Renewed Facility Licenses**

##### **IP2 License Condition 1.J and IP3 License Condition 1.I**

IP2 License Condition 1.J and IP3 License Condition 1.I address the management of aging effects for IP2 and IP3 SSCs during the periods of extended operation (PEO) and time-limited aging analyses that require review. These IP2 and IP3 License Conditions are proposed to be deleted.

The requirements within 10 CFR 54.4 associated with the plant systems, structures, and components (SSG) are no longer applicable to a station with all fuel stored in the ISFSI.

The implementation scope for this requirement included:

"(a) Plant systems, structures, and components within the scope of this part are-

1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions-

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11 of this chapter, as applicable.

2) All non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in 10 CFR 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1)-(3) of this section."

All fire protection requirements associated with 10 CFR 50.48(f) are controlled within the fire protection program and do not require the separate aging requirements associated with 10 CFR 54.4.

Upon cessation of operations, IP2 and IP3 license renewal commitments for aging management were incorporated into Appendix A, "License Renewal" of the DSAR, which is updated in accordance with 10 CFR 50.71(e). Therefore, changes to these license renewal commitments continue to be evaluated pursuant to the criteria in 10 CFR 50.59.

There is no other equipment meeting the requirements of this standard that are needed in the ISFSI only condition. Therefore, none of the requirements associated with the scope 10 CFR 54.4 remain germane and the deletion of IP2 License Condition 1.J and IP3 License Condition 1.I are consistent with the requirements associated with decommissioning nuclear power plant(s).

IP2 License Condition 2.C.(2) and IP3 License Condition 2.C.(2), “Technical Specifications”

IP2 License Condition 2.C.(2) and IP3 License Condition 2.C.(2) provide a generic discussion of the IP2 Technical Specifications contained in Appendices A and B and IP3 Technical Specifications in Appendices A, B, and C.

<b>IP2 License Condition 2.C.(2) as modified by the LAR submitted on August 2, 2022</b>	<b>Proposed IP2 License Condition 2.C.(2)</b>
The Technical Specifications contained in Appendices A, and B, as revised through Amendment No. XXX, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.	The Technical Specifications contained in Appendices A, and B, as revised through Amendment No. XXX, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.

<b>IP3 License Condition 2.C.(2)</b>	<b>Proposed IP2 License Condition 2.C.(2)</b>
The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 271, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.	The Technical Specifications contained in Appendices A, <i>and B, and C,</i> as revised through Amendment No. <del>271</del> XXX, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.

The IP2 License Condition is modified to resolve a typographical error and to update the reference to the License Amendment number. The IP3 License Condition is modified to eliminate the reference to the IP3 Appendix C TSs (as discussed below) and to update the reference License Amendment number.

IP2 License Conditions 2.I through 2.M and IP3 License Conditions 2.H through 2.AB

IP2 License Conditions 2.I through 2.M and IP3 License Conditions 2.H through 2.AB were previously deleted. The historical references to the deleted License Conditions are proposed to be deleted. These are administrative changes.

IP2 License Condition 2.N and IP3 License Condition 2.AC, “Mitigation Strategy License Condition”

IP2 License Condition 2.N and IP3 License Condition 2.AC address requirements added to the FLs to assure that IP2 and IP3 would be capable of mitigating large fires and explosions. These License Conditions are proposed for deletion. After the IP2 and IP3 irradiated fuel is stored within an ISFSI, the mitigation strategy license condition is no longer required.

These License Conditions incorporated the requirements for Section B.5.b mitigation strategies of the Interim Compensatory Measures (ICM) Order EA-02-026 dated February 25, 2002 (Reference 6). Subsequently, 10 CFR 50.54(hh) became effective on May 26, 2009. These License Conditions provide mitigation strategies and other requirements for loss of large areas of IP2 or IP3 due to large fires and explosions. However, as stated in 10 CFR 50.54(hh)(2), this section of the regulation does not apply to a permanently defueled reactor that has submitted the certifications under 10 CFR 50.82(a) for permanent cessation of operations and permanent

removal of fuel from the reactor vessel. These have been submitted for IP2 and IP3 and docketed (References 1 and 2). By letter dated November 28, 2011, the NRC rescinded Item B.5.b of the ICM Order (Reference 7). Therefore, neither the ICM Order nor 10 CFR 50.54(hh) continue to apply to IP2 and IP3.

Additionally, 10 CFR 50.155, "Mitigation of Beyond-Design-Basis Events," provides guidance on mitigating capabilities at permanently shut down and defueled reactors. Specifically, 10 CFR 50.155(a)(2)(iv) states, "Holders of operating licenses or combined licenses for which the certifications described in § 50.82(a)(1) or § 52.110(a) of this chapter have been submitted need not meet the requirements of this section once all irradiated fuel has been permanently removed from the spent fuel pool(s)." IP2 and IP3 will implement these changes to their FLs once all spent fuel has been moved from the IP2 SFP and IP3 SFP to dry cask storage within an ISFSI; therefore, 10 CFR 50.155 will no longer apply to IP2 and IP3.

#### IP2 License Condition 2.O and IP3 License Condition 2.AD

IP2 License Condition 2.O and IP3 License Condition 2.AD were previously deleted. The historical references to the deleted License Conditions are proposed to be deleted. These are administrative changes.

#### IP2 License Condition 2.P and IP3 License Condition 2.AE

IP2 License Condition 2.P and IP3 License Condition 2.AE permit the transfer of IP3 spent fuel to the IP2 SFP subject to the conditions listed in the IP2 and IP3 Appendix C TSs. Additionally, these License Conditions permit the transfer of IP3 spent fuel into approved dry storage casks for onsite storage.

IP2 License Condition 2.P and IP2 Appendix C TS are proposed to be deleted in Reference 3.

Following the transfer of all of the spent fuel from the IP2 SFP to dry storage within an ISFSI, inter-unit transfer of spent fuel from IP3 to the IP2 SFP will no longer be permitted. In addition, the capability for IP3 to store spent fuel in onsite dry storage is permitted by the general license conditions defined in 10 CFR 72. Thus, IP3 License Condition 2.AE and IP3 Appendix C TS are no longer necessary, and their elimination is acceptable.

#### IP2 License Condition 2.Q and IP3 License Condition 2.AF, "License Renewal License Conditions"

IP2 License Condition 2.Q and IP3 License Condition 2.AF involve management of license renewal amendment (LRA) commitments. The purpose of these LRA commitments was to ensure that the aging effects of equipment important to the safe operation of the reactor are managed so that the functionality of SSCs is maintained during the facility's period of extended operation.

The IP2 and IP3 license renewal commitments for aging management are maintained in Appendix A, "License Renewal," of the IP2 and IP3 DSARs (References 8 and 9). The IP2 and IP3 DSARs are updated in accordance with 10 CFR 50.71(e). Changes to these license renewal commitments continue to be evaluated and controlled pursuant to the requirements of 10 CFR 50.59 and 10 CFR 50.71(e).

The spent fuel storage cask systems located in an ISFSI are subject to their own 10 CFR 72 Certificate of Compliance and Cask Technical Specification requirements. These cask

protection requirements are not referenced or identified in IP2 License Condition 2.Q and IP3 License Condition 2.AF. Based on the above, administrative controls for maintaining the IP2 and IP3 DSARs will be used to address and control license renewal commitments. Therefore, the deletions of IP2 License Condition 2.Q and IP3 License Condition 2.AF are acceptable.

### IP2 FL and IP3 FL Attachments

The proposed changes to the list of attachments in the IP2 and IP3 FLs:

- 1) Rename the IP2 PDTs and IP3 PDTs to ISFSI Only Technical Specifications; and
- 2) Eliminate the reference to the IP3 Appendix C TS.

The proposed changes to rename the IP2 and IP3 PDTs reflects that IP2 and IP3 are permanently defueled and shutdown, and all fuel has been removed from and permanently prevented from being stored in the IP2 and IP3 SFPs. The revisions to the IP2 and IP3 list of Attachments are consistent with the proposed changes to remove operational TSs associated with storage of spent fuel in the IP2 SFP or IP3 SFP.

Following the transfer of all of the spent fuel from the IP2 SFP to dry storage within an ISFSI, inter-unit transfer of spent fuel from IP3 to IP2 will no longer be permitted. Thus, the IP3 Appendix C TS are no longer necessary, and their elimination is acceptable. The IP2 Appendix C TS are proposed to be deleted in Reference 3.

The reference to the Appendix C TS in the IP2 FL is proposed to be deleted in Reference 3.

### **3.2.2 Proposed Changes to IP2 Appendix A and IP3 Appendix A, “Permanently Defueled Technical Specifications”**

#### IP2 PDTs and IP3 PDTs Cover Pages

The proposed changes to the IP2 and IP3 PDTs Cover Pages are administrative changes to:

- 1) Reflect that the IP2 and IP3 FLs are Renewed FLs;
- 2) Rename the IP2 PDTs and IP3 PDTs from Permanently Defueled Technical Specifications to ISFSI Only Technical Specifications; and
- 3) Eliminate the reference to the IP2 and IP3 Bases for the Technical Specifications.

The proposed changes reflect that IP2 and IP3 are permanently defueled and shutdown, and all fuel has been removed from and permanently prevented from being stored in the IP2 and IP3 SFPs. The revisions to the IP2 and IP3 PDTs cover pages are consistent with the proposed changes to remove operational Technical Specifications associated with storage of spent fuel in the IP2 SFP or IP3 SFP and the associated Bases for those Technical Specifications.

#### IP2 PDTs 1.1 and IP3 PDTs 1.1, "Definitions"

IP2 PDTs 1.1, including the term Actions, and IP3 PDTs 1.1, including the terms Actions, Certified Fuel Handler, and Non-certified Operator, are proposed to be eliminated. The purpose of these definitions is to provide uniform interpretation of frequently used terms in the IP2 and IP3 PDTs. The proposed changes to other requirements of the IP2 and IP3 PDTs, as reflected in this submittal, either eliminate or relocate the information that references these terms.

Since these terms are no longer needed after spent fuel has been removed from the IP2 and IP3 SFPs and transferred to an ISFSI, the deletion of these definitions from the IP2 and IP3 PDTs is acceptable.

Note: the terms Certified Fuel Handler and Non-certified Operator are proposed to be deleted from the IP2 PDTs in Reference 3.

#### IP2 PDTs 1.2 and IP3 PDTs 1.2, "Logical Connectors"

IP2 PDTs 1.2 and IP3 PDTs 1.2 are proposed to be eliminated. The purpose of these sections is to explain the meaning of logical connectors in Required Actions and Surveillances. The remaining Required Actions and Surveillances in the IP2 and IP3 PDTs are proposed to be eliminated as reflected in this submittal.

The rules of usage regarding logical connectors are no longer needed after spent fuel has been removed from the IP2 and IP3 SFPs and transferred to an ISFSI. Thus, the deletion of the discussion of logical connectors from the IP2 and IP3 PDTs is acceptable.

#### IP2 PDTs 1.3 and IP3 PDTs 1.3, "Completion Times"

IP2 PDTs 1.3 and IP3 PDTs 1.3 are proposed to be eliminated. The purpose of these sections was to establish the Completion Time convention and provide guidance for its use. The remaining Completion Times in the IP2 and IP3 PDTs are proposed to be eliminated as reflected in this submittal.

The rules and guidance regarding Completion Times are no longer needed after spent fuel has been removed from the IP2 and IP3 SFPs and transferred to an ISFSI, it is acceptable to delete the discussion of Completion Times from the IP2 and IP3 PDTs.

#### IP2 PDTs 1.4 and IP3 PDTs 1.4, "Frequency"

IP2 PDTs 1.4 and IP3 PDTs 1.4 are proposed to be eliminated. The purpose of these sections was to define the proper use and application of Frequency requirements. The remaining Frequencies in the IP2 and IP3 PDTs are proposed to be eliminated as reflected in this submittal.

The rules and guidance regarding Frequency are no longer needed after spent fuel has been removed from the IP2 and IP3 SFPs and transferred to an ISFSI. Thus, the deletion of the discussion of Frequency from the IP2 and IP3 PDTs is acceptable.

### IP2 PDTS 2.0 and IP3 PDTS 2.0

IP2 PDTS 2.0 and IP3 PDTS 2.0 were previously deleted from the IP2 and IP3 PDTSs. These proposed changes remove the pages that were retained after PDTS 2.0 of the IP2 and IP3 PDTSs were deleted. These are administrative changes to eliminate pages that no longer contain PDTS requirements.

### IP2 PDTS 3.0 and IP3 PDTS 3.0, "Limiting Condition for Operation (LCO) Applicability"

IP2 PDTS 3.0 and IP3 PDTS 3.0, "Limiting Condition for Operation (LCO) Applicability," provide guidance and requirements for use of PDTS LCOs. These requirements, including LCO 3.0.1 and LCO 3.0.2, are proposed for deletion. The proposed changes to other sections of the IP2 and IP3 PDTSs eliminate all LCOs. Since LCOs are no longer needed after the spent fuel has been removed from the IP2 and IP3 SFPs and transferred to the ISFSI, the deletion of these TS requirements in their entirety is acceptable, because it has no impact on continued safe storage and maintenance of spent fuel in an ISFSI.

### IP2 PDTS 3.0 and IP3 PDTS Section 3.0, "Surveillance Requirement (SR) Applicability"

IP2 PDTS 3.0 and IP3 PDTS Section 3.0, "Surveillance Requirement (SR) Applicability," provide guidance and requirements for use of PDTS SRs. These requirements including SR 3.0.1 through SR 3.0.4, are proposed for deletion. The proposed changes to other sections of the IP2 and IP3 PDTSs eliminate all SRs. Since SRs are no longer needed after the spent fuel has been removed from the IP2 and IP3 SFPs and transferred to an ISFSI, the deletion of these TS requirements in their entirety is acceptable, because it has no impact on continued safe storage and maintenance of spent fuel in an ISFSI.

### IP2 PDTS 3.7.11 and IP3 PDTS 3.7.14, "Spent Fuel Pit Water Level"

IP2 PDTS 3.7.11 and IP3 PDTS 3.7.14 provide requirements to ensure that the IP2 and IP3 SFP water levels are maintained at a specified level during movement of irradiated fuel assemblies in the IP2 or IP3 SFP, respectively. They are proposed to be deleted. The requirements in these PDTS are related to assuring the ability to safely store spent fuel in the IP2 and IP3 SFPs. IP2 PDTS 3.7.11 and IP3 PDTS 3.7.14 do not apply when there is no spent fuel stored in the IP2 and IP3 SFP, respectively. Without any spent fuel in the IP2 SFP or IP3 SFP, SFP water level has no impact on the continued safe storage and maintenance of spent fuel in an ISFSI. Thus, the deletion of these PDTS is acceptable.

### IP2 PDTS 3.7.12 and IP3 PDTS 3.7.15, "Spent Fuel Pit Boron Concentration"

IP2 PDTS 3.7.12 and IP3 PDTS 3.7.15 provide requirements to ensure that the IP2 and IP3 SFP boron concentrations are maintained at specified concentrations when fuel assemblies are stored in the IP2 or IP3 SFP, respectively. They are proposed to be deleted. The requirements in these PDTS are related to assuring the ability to safely store spent fuel in the IP2 and IP3 SFPs. IP2 PDTS 3.7.12 and IP3 PDTS 3.7.15 do not apply when there is no spent fuel stored in the IP2 and IP3 SFP, respectively. Without any spent fuel in the IP2 SFP or IP3 SFP, SFP boron concentration has no impact on the continued safe storage and maintenance of spent fuel in an ISFSI. Thus, the deletion of these PDTS is acceptable.

IP2 PDTS 3.7.13 and IP3 PDTS 3.7.16, “Spent Fuel Assembly Storage”

IP2 PDTS 3.7.13 and IP3 PDTS 3.7.16 provide requirements to ensure that the IP2 and IP3 fuel assemblies are properly classified and stored in appropriate locations within the IP2 and IP3 SFPs, respectively, when fuel assemblies are stored in the IP2 or IP3 SFP. They are proposed to be deleted. The requirements in these PDTS are related to assuring the ability to safely store spent fuel in the IP2 and IP3 SFPs. IP2 PDTS 3.7.13 and IP3 PDTS 3.7.16 do not apply when there is no spent fuel stored in the IP2 and IP3 SFP, respectively. Without any spent fuel in the IP2 SFP or IP3 SFP, the requirements regarding storage of fuel assemblies in the IP2 SFP or IP3 SFP has no impact on the continued safe storage and maintenance of spent fuel in the ISFSI. Thus, the deletion of these PDTS is acceptable.

IP2 PDTS 4.1 and IP3 PDTS 4.1, “Site Location”

IP2 PDTS 4.1 and IP3 PDTS 4.1 describe the IP2 and IP3 site locations and the minimum distances from the IP2 and IP3 reactor center-lines to the site exclusion area and the outer boundary of the low population zone. Additionally, IP2 PDTS 4.1 describes the “Restricted Area.”

The proposed changes to IP2 PDTS 4.1 and IP3 PDTS 4.1 replace the term “Indian Point 2” and “Indian Point 3” with “IP2” and “IP3,” respectively. These are administrative changes to reflect a preferred nomenclature.

The proposed changes to IP2 PDTS 4.1 and IP3 PDTS 4.1 delete the discussions of the minimum distance requirements regarding the IP2 and IP3 reactor center-line locations. The minimum distance from the centerline of the IP2 and IP3 reactor containment to the site exclusion area and the outer boundary of the low population zone is based on requirements contained in 10 CFR 100.3 regarding reactor accident dose analyses. Based on submittal of the 10 CFR 50.82(a)(2) certifications, the IP2 and IP3 FLs no longer authorize operation of the reactors or emplacement or retention of fuel in the IP2 and IP3 reactor vessels, respectively. As a result, IP2 PDTS 4.1 and IP3 PDTS 4.1 no longer need to describe the IP2 or IP3 reactor center-line location. Thus, the deletion of the discussions regarding the IP2 and IP3 reactor centerlines is acceptable.

Additionally, the proposed change to IP2 PDTS 4.1 deletes the discussion regarding the Restricted Area. The elimination of this information is consistent with the level of detail provided in IP3 PDTS 4.1. Restricted Areas will continue to be defined in accordance with 10 CFR 20 in the applicable site procedures.

IP2 PDTS 4.3 and IP3 PDTS 4.3, “Fuel Storage”

IP2 PDTS 4.3 and IP3 PDTS 4.3 and their subsections describe design features associated with spent fuel storage in the IP2 and IP3 SFPs, respectively.

The proposed change to IP2 PDTS 4.3 clarifies that the reference is to the IP2 SFP. This is a clarification that does not alter the intent. Other changes to IP2 TS 4.3 are proposed in Reference 3. These changes conform to the changes proposed for IP3 TS 4.3.

The proposed change to IP3 PDTS 4.3 eliminates the existing IP3 SFP protection requirements and replaces them with new prohibitions against the storage of spent fuel in the IP3 SFP. This new prohibition will permanently preclude storage of fuel in the IP3 SFP after the spent fuel is removed during the final off-load campaign. This is acceptable, because the requirements of IP3

PDTS 4.3 are no longer applicable following the permanent removal of all spent fuel from the IP3 SFP.

IP2 PDTS 5.1 and IP3 PDTS 5.1, "Responsibility"

IP2 PDTS 5.1 and IP3 PDTS 5.1 are proposed to be deleted. The requirements of IP2 PDTS 5.1.1 and IP3 PDTS 5.1.1 will be relocated to the IP2 and IP3 DSARs, respectively, and the requirements of IP3 PDTS 5.1.2 will be deleted. The requirements of IP2 PDTS 5.1.2 are proposed to be deleted in Reference 3.

IP2 PDTS 5.1.1 and IP3 PDTS 5.1.1 provide a description and requirements regarding certain key operational management responsibilities. The proposed change is to delete these sections from the IP2 and IP3 PDTS and relocate them to the IP2 and IP3 DSARs, respectively. In the IP2 and IP3 DSARs, a new administrative controls section will be created to retain the relocated Administrative Controls specifications. The title of the "plant manager" position in TS 5.1.1 will be revised to "manager responsible for overall operational activities" when it is relocated to the IP2 and IP3 DSARs. This title change does not change any requirements, qualifications, or responsibilities of the individual in this position, and is strictly an administrative change to provide flexibility in the event of organization changes.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for responsibility requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letters 95-06 and 96-04 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

IP3 PDTS 5.1.2 requires the shift manager to be responsible for the shift command function. This requirement is proposed to be eliminated. As described in the existing IP3 PDTS 5.1.2 and 5.2.2, the shift command function is focused on operations involving the storage or movement of spent nuclear fuel within the IP3 SFP. After all of the spent fuel is permanently removed from the IP3 SFP, the need for the shift manager and shift command function for spent fuel management no longer exists. The position of shift manager as described in the PDTS is a holdover from the control room function of supervising multiple functions of an operating nuclear power plant. With the limited requirements for supervision of the passive dry fuel storage system utilized within an ISFSI or with respect to the decommissioning of the former power generation facilities, the shift manager position and shift command function are no longer required, and the proposed deletion of IP3 PDTS 5.1.2 is acceptable.

IP2 PDTS 5.1.2 requirement for shift manager is proposed to be deleted in Reference 3.

IP2 PDTS 5.2.1 and IP3 PDTS 5.2.1, "Onsite and Offsite Organizations"

IP2 PDTS 5.2.1 and IP3 PDTS 5.2.1 provide administrative controls regarding organizational lines of authority, responsibilities, and organizational freedom for certain personnel including those performing health physics or quality assurance functions. The administrative controls provided in the IP2 PDTS 5.2.1 and IP3 PDTS 5.2.1 will be relocated to the IP2 and IP3 DSARs verbatim, with the following exceptions:

- The IP2 PDTS 5.2.1.(b) and IP3 PDTS 5.2.1.(b) reference to "plant manager" will be revised to "manager responsible for overall operational activities." This does not change any requirements, qualifications, or responsibilities of the individual in this position, and

is strictly an administrative change. This change allows HDI specific titles to be identified in licensee-controlled documents such as the IP2 and IP3 DSARs.

The IP2 PDTS 5.2.1.(d) and IP3 PDTS 5.2.1.(d) reference to individuals who “train the Certified Fuel Handlers” will not be incorporated into the IP2 and IP3 DSARs. After all fuel is stored within an ISFSI and storage of spent fuel in the IP3 SFP is prohibited, there will no longer be an organizational need for Certified Fuel Handlers or the associated training program. Therefore, this proposed change will have no impact on continued safe storage and maintenance of spent fuel stored in the ISFSI. IP2 PDTS 5.2.1.(d) is proposed to be deleted in Reference 3.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for organizational requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letters 95-06 and 96-04 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

#### IP2 PDTS 5.2.2 and IP3 PDTS 5.2.2, “Facility Staff”

IP3 PDTS 5.2.2 is proposed to be deleted in its entirety. These administrative controls pertain to the facility staff organization and requirements when spent fuel is stored or moved within the IP3 SFP. With the removal of the spent fuel from the IP3 SFP, there are no remaining spent fuel assemblies to be monitored in the IP3 SFP and there are no credible accidents at IP3 that require the actions of a Certified Fuel Handler, Shift Manager, or a Non-certified Operator to prevent occurrence or mitigate the consequences of an accident. After all spent fuel is transferred from the IP3 SFP to dry storage within an ISFSI, storage of spent fuel in the IP3 SFP is prohibited, and inter-unit spent fuel transfer from IP3 to the IP2 SFP is no longer permitted, it is unnecessary to retain these spent fuel handling administrative controls. Therefore, the deletion of IP3 PDTS 5.2.2 after the fuel has been moved from the IP3 SFP to an ISFSI will have no impact on safe storage and maintenance of spent fuel in the ISFSI, and is acceptable.

IP2 PDTS 5.2.2 is proposed to be deleted in Reference 3.

#### IP2 PDTS 5.3 and IP3 PDTS 5.3, “Facility Staff Qualifications”

The requirements of IP2 PDTS 5.3.1 and IP3 PDTS 5.3.1 are proposed to be deleted from the TSs and relocated to the IP2 and IP3 DSARs, respectively. After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for facility staff qualification requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letter 95-06 and 94-06 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

IP3 PDTS 5.3 requires the maintenance of an NRC approved training and retraining program for Certified Fuel Handlers. As described in the existing IP3 PDTS 5.3.2, the IP3 shift command function is focused on operations involving the storage or movement of spent nuclear fuel within the IP3 SFP. With the removal of the spent fuel from the IP3 SFP, there are no remaining spent fuel assemblies to be monitored in the IP3 SFP and there are no credible accidents at IP3 that require the actions of a Certified Fuel Handler, Shift Manager, or a Non-certified Operator to

prevent occurrence or mitigate the consequences of an accident. Following the transfer of all spent fuel from the IP3 SFP to dry storage within an ISFSI, storage of spent fuel in the IP3 SFP and transfers of spent fuel from IP3 to the IP2 SFP will be prohibited upon implementation of this License Amendment. There will no longer be a need for Certified Fuel Handlers or the associated training programs. Therefore, this proposed deletion will have no impact on safe storage and maintenance of spent fuel in the ISFSI, and is acceptable.

IP2 PDTS 5.3.2 is proposed to be deleted in Reference 3.

#### IP2 PDTS 5.4 and IP3 PDTS 5.4, "Procedures"

The requirements of IP2 PDTS 5.4.1.(a) and IP3 PDTS 5.4.1.(a) which involve fuel handling operations in the IP2 or IP3 SFP are proposed to be deleted from the TS. Additionally, the requirements of IP2 PDTS 5.4.1 and IP3 PDTS 5.4.1 are proposed to be deleted from the TSs and relocated to the IP2 and IP3 DSARs, respectively.

As discussed above, following the transfer of the spent fuel from the IP2 SFP and IP3 SFP to the ISFSI, the proposed changes to IP2 PDTS 4.3 and IP3 PDTS 4.3 will prohibit the storage of spent fuel in the IP2 SFP or IP3 SFP.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for facility staff qualification requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letter 95-06 and 94-06 (References 10 and 114) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

#### IP2 PDTS 5.5 and IP3 PDTS 5.5, "Programs and Manuals"

IP2 PDTS 5.5 and IP3 PDTS 5.5 address the programs and manuals that are required to be established and maintained. This section identifies that the Offsite Dose Calculation Manual (ODCM), the Radioactive Effluent Controls Program, and the Explosive Gas and Storage Tank Radioactivity Monitoring Program are required to be maintained.

#### IP2 PDTS 5.5.1 and IP3 PDTS 5.5.1, "Offsite Dose Calculation Manual (ODCM)"

IP2 PDTS 5.5.1 and IP3 PDTS 5.5.1 specify how to document, review, and approve changes to the IP2 and IP3 ODCM, respectively. The proposed change is to delete IP2 PDTS 5.5.1 and IP3 PDTS 5.5.1 and relocate the requirements verbatim to the IP2 and IP3 DSARs with the following exceptions:

- References to Technical Specification 5.6.2 and Technical Specification 5.6.3 in IP2 PDTS 5.5.1.b and IP3 PDTS 5.5.1.b will be deleted. These are administrative changes, because these requirements will no longer reside in the PDTSs once relocated to the IP2 and IP3 DSARs and the references are not necessary to describe the administrative requirements.
- References to the "plant manager" position in IP2 PDTS 5.5.1.c.2 and IP3 PDTS 5.5.1.c.2 will be revised to "manager responsible for overall operational activities". This administrative change does not alter any requirements, qualifications, or responsibilities of the individual in this position, and is strictly an administrative change.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for facility staff qualification requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letter 95-06 and 94-06 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

#### IP2 PDTS 5.5.2 and IP3 PDTS 5.5.2

IP2 PDTS 5.5.2 and IP3 PDTS 5.5.2 were previously deleted from the IP2 and IP3 PDTSs. These proposed changes remove the existing references to the deleted subsections that were retained after PDTS 5.5.2 of the IP2 and IP3 PDTSs were deleted. These are administrative changes to eliminate references to subsections that no longer contain PDTS requirements.

#### IP3 PDTS 5.5.3

IP3 PDTS 5.5.3 was previously deleted from the IP3 PDTS. The proposed change removes the existing reference to the deleted subsection that was retained after IP3 PDTS 5.5.3 was deleted. This is an administrative change to eliminate a reference to a subsection that no longer contains PDTS requirements.

#### IP2 PDTS 5.5.3 and IP3 PDTS 5.5.4, "Radioactive Effluent Controls Program"

IP2 PDTS 5.5.3 and IP3 PDTS 5.5.4 specify administrative requirements for the program to control radioactive effluents and for maintaining doses to the public to within the specified limits. The proposed changes delete IP2 PDTS 5.5.3 and IP3 PDTS 5.5.4 and relocate the requirements to the IP2 and IP3 DSARs with the following exceptions:

- References to iodine-131 in IP2 PDTS 5.5.3.g.2 and 5.5.4.i and IP3 PDTS 5.5.4.g.2 and 5.5.4.i will not be relocated to the IP2 and IP3 DSARs, respectively, due to the radioactive decay and short half-lives (approximately 8 days) and time since permanent cessation of IP2 and IP3 reactor operation.
- The requirements regarding noble gases in IP2 PDTS 5.5.3.g.1 and 5.5.3.h and IP3 PDTS 5.5.4.g.1 and 5.5.4.h will not be relocated to the IP2 and IP3 DSARs, respectively, since after all spent fuel is transferred from the IP2 SFP and IP3 SFP to an ISFSI and contained within dry storage casks, there will no longer be a requirement to monitor for noble gases released from IP2 and IP3.
- References to "gaseous" effluents and monitoring will be revised to "airborne" effluents and monitoring will be incorporated based on conditions applicable to post fuel removal from the IP2 SFP and IP3 SFP to dry casks located in the ISFSI. This proposed change does not alter monitoring requirements previously identified and is strictly a clarification.
- References to IP2 PDTS SR 3.0.2 and SR 3.0.3 and IP3 PDTS SR 3.0.2 and SR 3.0.3 will not be relocated to the IP2 and IP3 DSARs. As previously discussed, IP2 PDTS SR 3.0.2 and SR 3.0.3 are proposed to be deleted. Therefore, they are no longer applicable.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for the Radioactive Effluents Control Program to the IP2 and IP3 DSARs is consistent

with NRC Administrative Letter 95-06 and 96-04 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

IP2 PDTs 5.5.4 through 5.5.9 and IP3 PDTs 5.5.5 through 5.5.10

IP2 PDTs 5.5.4 through 5.5.9 and IP3 PDTs 5.5.5 through 5.5.10 were previously deleted from the IP2 and IP3 PDTs. These proposed changes remove the existing references to the deleted subsections that were retained after the associated requirements were deleted from IP2 and IP3 PDTs. These are administrative changes to eliminate references to subsections that no longer contain PDTs requirements.

IP2 PDTs 5.5.10 and IP3 PDTs 5.5.11, "Explosive Gas and Storage Tank Radioactivity Monitoring Program"

IP2 PDTs 5.5.10 and IP3 PDTs 5.5.11 specify administrative controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The proposed changes delete IP2 PDTs 5.5.10 and IP3 PDTs 5.5.11 and relocate the requirements to the IP2 and IP3 DSARs with the following exception:

- References to IP2 PDTs SR 3.0.2 and SR 3.0.3 and IP3 PDTs SR 3.0.2 and SR 3.0.3 will not be relocated to the IP2 and IP3 DSARs. As previously discussed, IP2 PDTs SR 3.0.2 and SR 3.0.3 are proposed to be deleted. Therefore, they are no longer applicable.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for the Explosive Gas and Storage Tank Radioactivity Monitoring Program to the IP2 and IP3 DSARs is consistent with NRC Administrative Letter 95-06 and 96-04 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

IP2 PDTs 5.5.11 and IP3 PDTs 5.5.12

IP2 PDTs 5.5.11 and IP3 PDTs 5.5.12 were previously deleted from the IP2 and IP3 PDTs. These proposed changes remove the existing references to the deleted subsections that were retained after the associated requirements were deleted from the TS. These are administrative changes to eliminate references to subsections that no longer contain TS requirements.

IP2 PDTs 5.5.12 and IP3 PDTs 5.5.13, "Technical Specifications (TS) Bases Control Program"

IP2 PDTs 5.5.12 and IP3 PDTs 5.5.13 provides a means for processing changes to the Bases for the IP2 PDTs. The IP2 and IP3 PDTs Bases are associated with the requirements in PDTs Section 3.0 Limiting Condition for Operation (LCO) applicability, PDTs Section 3.0 Surveillance Requirement (SR) applicability, and the PDTs requirements regarding storage of irradiated fuel in the IP2 or IP3 SFPs. Specifically, the requirements in IP2 PDTs 3.0, 3.7.11 through 3.7.13 and IP3 PDTs 3.0, 3.7.14 through 3.7.16 are proposed to be eliminated as previously described. Following transfer of all irradiated fuel from the IP2 SFP and IP3 SFP to an ISFSI, the IP2 and IP3 SFPs will no longer be used for irradiated fuel storage. Since all of the IP2 and IP3 TS Bases will be deleted, there will no longer be a need for an IP2 or IP3 TS Bases Control Program. Therefore, deleting IP2 PDTs 5.5.12 and IP3 PDTs 5.5.13 after all the irradiated fuel is transferred from the IP2 SFP and IP3 SFP to an ISFSI is acceptable.

### IP2 PDTS 5.6 and IP3 PDTS 5.6, "Reporting Requirements"

IP2 PDTS 5.6 and IP3 PDTS 5.6 address radiological reporting requirements. The proposed changes delete IP2 TS 5.6 and IP3 TS 5.6 and relocate the requirements verbatim to the IP2 and IP3 DSARs, with the following exception:

- IP2 PDTS 5.6.1 and IP3 PDTS 5.6.1 were previously deleted from the IP2 and IP3 PDTSs. These proposed changes remove the existing references to the deleted subsections that were retained after the associated requirements were deleted from the TS. These are administrative changes to eliminate references to subsections that no longer contain TS requirements.

After these administrative controls are incorporated into the IP2 and IP3 DSARs, any future changes will be controlled in accordance with 10 CFR 50.59. This will provide adequate control for IP2 and IP3 with all spent fuel located within the ISFSI. The relocation of administrative controls for the reporting requirements to the IP2 and IP3 DSARs is consistent with NRC Administrative Letter 95-06 and 96-04 (References 10 and 11) guidance and will have no impact on safe storage and maintenance of spent fuel in the ISFSI.

### IP2 PDTS 5.7.2 and IP3 PDTS 5.7.2, "High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation"

IP2 PDTS 5.7.2.a.1 and IP3 PDTS 5.7.2.a.1 require maintenance of the door and gate keys for high radiation areas under the control of the IP3 shift manager (i.e., the shift supervisor as referenced in the IP3 PDTS), radiation protection manager, or his or her designee. These PDTSs are proposed to be revised by replacing the references to the "IP3 shift manager" and "shift supervisor," in the IP2 and IP3 PDTS, respectively, with a reference to the "lead licensee representative on shift." This proposed change provides flexibility regarding the title for the individual that represents the licensee on shift at the ISFSI. This is an administrative change to reflect that there will no longer be a "shift supervisor" as previously referred to following the transfer of all spent fuel from the IP2 SFP and IP3 SFP to dry storage within an ISFSI.

After all of the spent fuel is permanently removed from the IP2 and IP3 SFPs, the need for the shift manager (i.e., shift supervisor) and shift command function for spent fuel management no longer exists. As previously discussed, with the limited requirements for supervision of the passive dry fuel storage system utilized within an ISFSI or with respect to the decommissioning of the former power generation facilities, the shift manager position and shift command function are no longer required, and the proposed deletion of TS 5.1.2 is acceptable.

### 3.2.3 Proposed Changes to IP3 Appendix C, "Inter – Unit Fuel Transfer Technical Specifications"

The Appendix C TS defined requirements that permitted the transfer of spent fuel from the IP3 SFP to the IP2 SFP. The IP3 Appendix C TS are proposed to be deleted in their entirety.

Following the transfer of all of the spent fuel from the IP2 SFP and IP3 SFP to dry storage within an ISFSI, inter-unit transfer of spent fuel from IP3 to IP2 will no longer be permitted. Thus, the IP3 Appendix C TS will be unnecessary and their elimination is acceptable.

Note: The IP2 Appendix C TS are proposed for deletion in Reference 3.

#### **4.0 REGULATORY EVALUATION**

##### **4.1 Applicable Regulatory Requirements/Criteria**

###### 10 CFR 50.2, Definitions. Safety-Related Structures, Systems and Components

10 CFR 50.2 defines safety-related SSCs as those SSCs that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11 of 10 CFR, as applicable.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to IP2 and IP3, because they are in a permanently defueled condition.

The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. After all nuclear spent fuel assemblies have been transferred from the IP2 SFP and IP3 SFP to dry cask storage within an ISFSI, none of the SSCs at IP2 or IP3 are required to be relied on for accident mitigation. Therefore, none of the SSCs at IP2 or IP3 meet the definition of a safety-related SSC stated in 10 CFR 50.2. The proposed deletion of requirements in the PDTs does not affect systems credited in any accident analysis at IP2 and IP3.

###### 10 CFR 50.36. Technical Specifications

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

The final Commission Policy Statement established four criteria to define the scope of equipment and parameters to be included in the improved Standard Technical Specifications. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design bases accidents or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those SSCs shown to be significant to public health and safety. These criteria, which were subsequently codified in changes to 10 CFR 50.36 (60 FR 36953), also

pertain to the TS requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for “installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.” Since the certifications of 10 CFR 50.82(a)(1) have been docketed for IP2 and IP3, under 10 CFR 50.82(a)(2) the IP2 and IP3 FLs no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel. Therefore, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a “process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” The purpose of this criterion is to capture those process variables, design features, or operating restrictions that involve an initial condition assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. IP2 and IP3 are no longer licensed to operate. The design basis accidents for IP2 and IP3 are fuel handling accidents predicated on the storage of spent fuel in the IP2 SFP or IP3 SFP. After the IP2 and IP3 spent fuel is stored in the ISFSI, there are no remaining design basis accidents which are credible. Therefore, this criterion is not applicable.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The intent of this criterion is to capture into TS only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. Since fuel will have been removed from the IP2 SFP and IP3 SFP prior to implementation of this License Amendment, this criterion is not applicable.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. Since fuel will have been removed from the IP2 SFP and IP3 SFP prior to implementation of this License Amendment, this criterion is not applicable.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they “... are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.” The particular administrative controls to be included in the TS, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations. Accordingly, the NRC staff determined that administrative control requirements that are not specifically required under 10 CFR 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program Manual, DSAR, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS

administrative controls is specified in 10 CFR 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations including 10 CFR 50.59 and Appendix B to 10 CFR Part 50 provide adequate regulatory control.

10 CFR 50.36(c)(6), "Decommissioning," applies only to nuclear power reactor facilities that have submitted the certifications required by 10 CFR 50.82(a)(1). For such facilities, TS involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

#### 10 CFR 50.51. Continuation of License

10 CFR 50.51 (b) states: "Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated. During such period of continued effectiveness, the licensee shall:

- (1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and
- (2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

#### 10 CFR 50.82. Termination of License

10 CFR 50.82(a)(2) states: "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

#### Administrative Letter (AL) 95-06 and 96-04

The Defueled Safety Analysis Report is an appropriate candidate for relocation of administrative controls due to the controls imposed by 10 CFR 50.59 and requirements to review changes, tests, and experiments.

NRC AL 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance" (Reference 10), and NRC AL 96-04, "Efficient Adoption of Improved Standard Technical Specifications" (Reference 11), provide guidance to licensees requesting amendments that relocate administrative controls to licensee-controlled documents for which there are regulatory requirements for future change review (i.e., 50.54(a) for the Quality Assurance Program Manual; and 50.59 for the Final Safety Analysis Report).

### **5.0 No Significant Hazards Consideration Determination**

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Holtec Decommissioning International, LLC (HDI), on behalf of Holtec Indian Point 2, LLC (IP2) and Holtec Indian Point 3, LLC (IP3), requests U.S. Nuclear Regulatory Commission (NRC) review and approval

of an amendment to Renewed Facility License No. DPR-26 and Renewed Facility License No. DPR-64. The proposed License Amendment would revise the 10 CFR Part 50 Renewed Facility Licenses (FLs) and Appendix A Permanently Defueled Technical Specifications (PDTSSs) and IP3 Appendix C, "Inter – Unit Fuel Transfer Technical Specifications," (Appendix C TSs) to reflect removal of all spent nuclear fuel from the IP2 and IP3 Spent Fuel Pits (SFPs) to dry cask storage within a site controlled Independent Spent Fuel Storage Installation (ISFSI).

In References 1 and 2, Entergy certified to the NRC, in accordance with 10 CFR 50.82(a)(1)(i), that power operations ceased at IP2 on April 30, 2020, and at IP3 on April 30, 2021. In addition, Entergy certified in accordance with 10 CFR 50.82(a)(1)(ii), that the fuel was permanently removed from the IP2 reactor vessel and placed in the IP2 SFP on May 12, 2020, and that the fuel was permanently removed from the IP3 reactor vessel and placed in the IP3 SFP on May 11, 2021.

HDI expects that transfer of the spent fuel from the IP2 and IP3 SFPs to dry storage within an ISFSI will be completed before February 2023 and December 2023, respectively. In support of these conditions, revisions to the IP2 and IP3 FLs and PDTSSs, and IP3 Appendix C TS are proposed to comport with the requirements for a facility configuration with all spent nuclear fuel in dry storage within an ISFSI.

In accordance with 10 CFR 50.92, HDI has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration because the proposed changes satisfy the criteria in 10 CFR 50.92(c). These criteria require that operation of IP2 and IP3 in accordance with the proposed License Amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The discussion below addresses each of these criteria and demonstrates that the proposed License Amendment for IP2 and IP3 does not constitute a significant hazard.

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

Response: No.

The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2 states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant boundary;
- (2) The capability to shutdown the reactor and maintain it in a safe shutdown condition;  
or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 100.11.

The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to IP2 and IP3, because they are in a permanently

defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after all nuclear spent fuel assemblies have been transferred to dry cask storage within an ISFSI, none of the SSCs at IP2 or IP3 are required to be relied on for accident mitigation. Therefore, none of the SSCs at IP2 or IP3 meet the definition of a safety-related SSC stated in 10 CFR 50.2. The proposed deletion of requirements in the TS does not affect systems credited in any accident analysis at IP2 and IP3.

Section 6.2 of the IP2 and IP3 Defueled Safety Analysis Reports (DSARs) describe the design basis accidents (DBAs) related to the IP2 SFP and IP3 SFP, respectively. These postulated accidents are predicated on spent fuel being stored in the IP2 or IP3 SFPs. With the removal of the spent fuel from the IP2 SFP and IP3 SFP, there are no remaining spent fuel assemblies to be monitored in the IP2 SFP and IP3 SFP and there are no credible accidents at IP2 that require the actions of a Certified Fuel Handler, Shift Manager, or a Non-certified Operator to prevent occurrence or mitigate the consequences of an accident.

The proposed changes modify the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS to be commensurate with the hazards associated with permanently shutdown and defueled facilities that have transferred all spent fuel from the IP2 and IP3 SFPs to dry storage within an ISFSI. After the removal of the spent fuel from the IP2 SFP and IP3 SFP and transfer to the ISFSI, no spent fuel assemblies will remain in the IP2 SFP or IP3 SFP. Coupled with a prohibition against storage of fuel in the IP2 SFP or IP3 SFP and the elimination of the allowance to transfer IP3 spent fuel to the IP2 SFP, the potential for fuel related accidents is eliminated.

The proposed changes do not have an adverse impact on the remaining decommissioning activities or any of their postulated consequences. The proposed changes related to the relocation of certain administrative requirements do not affect operating procedures or administrative controls that have the function of preventing or mitigating any accidents applicable to the safe management of spent fuel or decommissioning of the facilities. Therefore, the License Amendment Request (LAR) does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes eliminate the operational requirements and certain design requirements associated with the storage of the spent fuel in the IP2 SFP and IP3 SFP and relocate certain administrative controls to the IP2 and IP3 DSARs.

After the removal of the spent fuel from the IP2 SFP and IP3 SFP and transfer to the ISFSI, there are no spent fuel assemblies that remain in the IP2 SFP and IP3 SFP. Coupled with a prohibition against storage of fuel in the IP2 SFP and IP3 SFP, the potential for fuel related accidents is removed. The proposed changes do not introduce any new failure modes.

The proposed changes do not involve installation of new equipment or modification of existing equipment that could create the possibility of a new or different kind of accident. Hence, the proposed changes do not result in a change to the way the facilities or equipment are operated in a manner which could cause a new or different kind of accident initiator to be created.

Therefore, the LAR does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The removal of all spent nuclear fuel from the IP2 SFP and IP3 SFP into storage in dry casks within an ISFSI, coupled with a prohibition against future storage of fuel within the IP2 SFP and IP3 SFP, removes the potential for fuel related accidents. The design basis and accident assumptions within the IP2 and IP3 DSARs and the IP2 and IP3 PDTs relating to safe management and safety of spent fuel in the IP2 SFP and IP3 SFP are no longer applicable. The proposed changes do not affect remaining plant operations, structures, systems, or components supporting decommissioning activities.

The requirements for SSCs that have been removed from the IP2 and IP3 TSs are not credited in the existing accident analysis for any applicable postulated accident; and as such, do not contribute to the margin of safety associated with the accident analysis.

Therefore, the LAR does not involve a significant reduction in a margin of safety.

Based on the above, HDI concludes that the proposed changes to the IP2 and IP3 FLs and PDTs, and IP3 Appendix C TS present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of “no significant hazards consideration” is justified.

### **5.1 Precedent**

This proposed License Amendment is consistent with recently approved License Amendments issued for Pilgrim Nuclear Power Station on August 24, 2021 (Reference 12) and Duane Arnold Energy Center on January 21, 2022 (Reference 13).

### **5.2 Conclusion**

Based on the analyses and considerations described above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22(c)(9) as follows:

- (i) The LAR involves no significant hazard consideration.

As described in Section 5.0 of this evaluation, the proposed amendment involves no significant hazards consideration.

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

The proposed amendment does not involve any physical alterations to the configuration that could lead to a change in the type or amount of effluent release offsite.

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

The proposed amendment does not involve any physical alterations to the IP2 or IP3 configuration and does not involve any changes to regulatory requirements or programs and procedures related to controls for limiting radiation exposure that could lead to a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, HDI concludes that the LAR meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this License Amendment.

## 7.0 REFERENCES

1. Letter, Entergy to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No. 2," dated May 12, 2020 (Letter NL-20-042) (ML20133J902)
2. Letter, Entergy to NRC, "Certifications of Permanent Cessation of Power Operations and Permanent Removal of Fuel from the Reactor Vessel, Indian Point Nuclear Generating Unit No. 3," dated May 11, 2021 (Letter NL-21-033) (ML21131A157)
3. Letter, Entergy to NRC, "License Amendment Request to Revise Indian Point Nuclear Generating Unit No. 2 Permanently Defueled Technical Specifications to Modify Staffing Requirements following Transfer of Spent Fuel to Dry Storage," dated August 2, 2022 (Letter HDI-IPEC-22-057) (ML22214A128)
4. Holtec letter to U.S. Nuclear Regulatory Commission (NRC), "Post Shutdown Decommissioning Activities Report including Site-Specific Decommissioning Cost Estimate for Indian Point Nuclear Generating Units 1, 2 and 3," dated December 19, 2019, (ML19354A698)

5. HI-2210651, Revision 8, Fuel Loading Plan for Indian Point Unit 2 and 3 dated October 12, 2022.
6. Interim Compensatory Measures (ICM) Order EA-02-026, dated February 25, 2002 (ML020500299)
7. Letter, USNRC to Holders of Licenses for Operating Power Reactors as Listed in the Enclosure, "Rescission or Partial Rescission of Certain Power Reactor Security Orders Applicable to Nuclear Power Plants," dated November 28, 2011 (ML111220447)
8. Letter, HDI to NRC, "Biennial Defueled Safety Analysis Report Update, and Regulatory Commitment Change Summary – September 2020 to September 2022," dated September 14, 2022, transmitting IP2 Defueled Safety Analysis Report, Revision 1 (ML22257A127)
9. Letter, HDI to NRC, "Indian Point Unit No. 3 10 CFR 50.71(e) Submittal," dated September 21, 2021, transmitting IP3 Defueled Safety Analysis Report, Revision 0 (ML21270A055)
10. NRC Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995 (ML031110271)
11. NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," dated October 9, 1996 (ML031110087)
12. Letter, USNRC to HDI LLC, "Pilgrim Nuclear Power Station – Issuance of Amendment to Revise the Permanently Defueled Technical Specifications to Align to the Requirements for Permanent Removal of Spent Fuel from the Spent Fuel Pool (EPID NO. L-2021-LLA-0049)," dated August 24, 2021 (ML21203A047)
13. Letter, USNRC to Next Era Energy, Inc, "Duane Arnold Energy Center – Issuance of Amendment No. 315 to Renewed Facility Operating License No DPR-49 to Revise the Permanently Defueled Technical Specifications to Align to the Requirements for Permanent Removal of Spent Fuel from the Spent Fuel Pool (EPID L-2021-LLA-0029)," dated January 21, 2022 (ML21172A217)

**HDI-IPEC-22-076**

**Enclosure - Attachment 1**

Markup Pages of the IP2 FL and PDTS

- H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed Facility License No. DPR-26, subject to the conditions for the protection of the environment set forth herein, is in accordance with 10 CFR Part 51, Appendix B, of the Commission's regulations and all applicable requirements of said Appendix B have been satisfied;
- I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and
- ~~J. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.~~

2. Renewed Facility License No. DPR-26 is hereby issued to Holtec IP2 and HDI to read as follows:

- A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 2, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by Holtec IP2 and maintained by HDI. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the Defueled Safety Analysis Report, as supplemented and amended, and the Environmental Report, as amended.
- B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:
  - (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec IP2 to possess and use, and (b) HDI to possess and use, the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;
  - (2) HDI pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended.

- (3) HDI pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use, at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required.
  - (4) HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components.
  - (5) HDI pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect. and is subject to the additional conditions specified or incorporated below:
- (1) Deleted per Amendment No. 294.
  - (2) Technical Specifications XXX  
The Technical Specifications contained in Appendices A, and B, as revised through Amendment No. ~~XXX~~, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.
  - (3) Deleted per Amendment No. 294.

HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The HDI CSP was approved by License Amendment No. 266, as supplemented by changes approved by License Amendment Nos. 279, 284, and 286.

HDI has been granted Commission authorization to use "stand alone preemption authority" under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. HDI shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

~~I. Deleted per Amdt. 133, 7-6-88.~~

~~J. Deleted per Amdt. 133, 7-6-88.~~

~~K. Deleted per Amendment No. 294.~~

~~L. Deleted per Amendment 238.~~

~~M. Deleted per Amendment 238.~~

~~N. Mitigation Strategy License Condition~~

~~The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:~~

~~(a) Fire fighting response strategy with the following elements:~~

- ~~1. Pre-defined coordinated fire response strategy and guidance~~
- ~~2. Assessment of mutual aid fire fighting assets~~
- ~~3. Designated staging areas for equipment and materials~~
- ~~4. Command and control~~
- ~~5. Training of response personnel~~

~~(b) Operations to mitigate fuel damage considering the following:~~

- ~~1. Protection and use of personnel assets~~
- ~~2. Communications~~
- ~~3. Minimizing fire spread~~
- ~~4. Procedures for implementing integrated fire response strategy~~
- ~~5. Identification of readily available pre-staged equipment~~
- ~~6. Training on integrated fire response strategy~~

- (e) ~~Actions to minimize release to include consideration of:~~
  - 1. ~~Water spray scrubbing~~
  - 2. ~~Dose to onsite responders~~

~~O. Deleted per Amendment No. 294.~~

~~P. HDI is authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by HDI and Holtec Indian Point 3, LLC.~~

~~Q. License Renewal License Conditions~~

- (1) ~~The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3," (SER) and supplements to the SER, are collectively the "License Renewal UFSAR Supplement." The UFSAR Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in the UFSAR Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.~~
- (2) ~~The License Renewal UFSAR Supplement, as defined in license condition Q(1) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).~~
  - a. ~~The licensee shall implement those new programs and enhancements to existing programs no later than the date specified in the License Renewal UFSAR Supplement.~~
  - b. ~~The licensee shall complete those activities no later than the date specified in the License Renewal UFSAR Supplement.~~
  - c. ~~The licensee shall notify the NRC in writing within 30 days after having accomplished item (2)a above and include the status of those activities that have been or remain to be completed in item (2)b above.~~

3. Deleted

(a) Deleted

(b) Provisional Trust:

- (i) The provisional trust agreement must be in a form acceptable to the NRC.
- (ii) Investments in the securities or other obligations of Holtec International or its affiliates, subsidiaries, successors, or assigns are and shall be prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are and shall be prohibited.

XXX

Amendment No. ~~XXX~~

- (iii) The provisional trust agreement must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The provisional trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.
- (iv) The provisional trust agreement must provide that the agreement cannot be amended in any material respect, or terminated, without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.
- (v) The appropriate section of the provisional trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.
- (vi) Use of assets in the provisional trust, in the first instance, shall be limited to the expenses related to decommissioning IP2 or IP1 as defined by the NRC in its regulations and issuances, and as provided in this license and any amendments thereto.

(c) Deleted

4. Deleted

5. Deleted

6. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

*/RA/*

Ho K. Nieh, Director  
Office of Nuclear Reactor Regulation

Attachments:

SFSI Only

Appendix A – ~~Permanently Defueled~~ Technical Specifications

Appendix B – Environmental Technical Specification Requirements

Date of Issuance: September 17, 2018

Amendment No. ~~XXX~~ XXX

APPENDIX A

TO

RENEWED



FACILITY LICENSE DPR-26

FOR

HOLTEC NUCLEAR INDIAN POINT 2, LLC AND  
HOLTEC DECOMMISSIONING INTERNATIONAL, LLC

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

DOCKET NO. 50-247

ISFSI ONLY



~~PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS AND BASES~~

XXX



Amendment No. ~~XXX~~

NOTE: This page is from the LAR submitted on August 2, 2022

RENEWED

FACILITY LICENSE No. DPR-26

Appendix A – ~~Permanently Defueled~~ Technical Specifications

ISFSI Only

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~~5.5.13 Deleted~~

~~5.5.14 Deleted~~

~~5.5.15 Deleted~~

XXX

**FACILITY LICENSE No. DPR-26**  
**Appendix A – Permanently Defueled Technical Specifications**

**Table of Contents (continued)**

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<del>5.6</del>	<del>Reporting Requirements</del>
<del>5.6.1</del>	<del>Not Used</del>
<del>5.6.2</del>	<del>Annual Radiological Environmental Operating Report</del>
<del>5.6.3</del>	<del>Radioactive Effluent Release Report</del>
5.7	High Radiation Area

---

XXX



~~1.0 USE AND APPLICATION~~

~~1.1 Definitions~~

---

~~NOTE~~

~~The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.~~

---

Term

Definition

~~ACTIONS~~

~~ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.~~

---

1.0 USE AND APPLICATION

1.2 Logical Connectors

---

**PURPOSE** The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Required Actions and Surveillances. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

---

**BACKGROUND** Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Surveillance, only the first level of logic is used, and the logical connector is left justified with the statement of the Surveillance.

---

**EXAMPLE** The following example illustrates the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify ... <u>AND</u> A.2 Restore ...	

---

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

---

1.0 USE AND APPLICATION

1.3 Completion Times

**PURPOSE** The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

**BACKGROUND** Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe handling and storage of spent nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

**DESCRIPTION** The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

**EXAMPLE** The following example illustrates the use of Completion Times with different Required Actions.

EXAMPLE 1.3.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent Fuel Pit boron concentration not within limit	A.1 Suspend movement of fuel assemblies in the Spent Fuel Pit	Immediately
	<u>AND</u> A.2 Initiate action to restore Spent Fuel Pit boron concentration to within limit.	Immediately

~~1.3 Completion Times~~

---

EXAMPLE (continued)

~~Condition A has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition A is entered.~~

~~The Required Actions of Condition A are to immediately suspend movement of fuel assemblies in the Spent Fuel Pit and initiate action to restore Spent Fuel Pit boron concentration to within limit.~~

---

~~IMMEDIATE~~      ~~When "Immediately" is used as a Completion Time, the Required~~  
~~COMPLETION TIME~~ ~~Action should be pursued without delay and in a controlled manner.~~

---

1.0 USE AND APPLICATION

1.4 Frequency

---

**PURPOSE**                    ~~The purpose of this section is to define the proper use and application of Frequency requirements.~~

---

**DESCRIPTION**            ~~Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.~~

~~The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0.2, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.~~

~~The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.~~

---

**EXAMPLE**                    ~~The following example illustrates the type of Frequency statement that appears in the Technical Specifications (TS).~~

EXAMPLE 1.4.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<del>Verify level is within limits.</del>	12 hours

---

~~Example 1.4.1 contains the type of SR encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is~~

---

1.4 Frequency

---

EXAMPLE (continued)

~~exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.~~

~~If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the specified condition or the LCO is considered not met (in accordance with SR 3.0.1).~~

---

~~2.0 DELETED~~

---

~~3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY~~

---

~~LCO 3.0.1 LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.~~

---

~~LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.~~

~~If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.~~

---

~~3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY~~

---

~~SR 3.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on variables outside specified limits.~~

---

~~SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance.~~

---

~~SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.~~

~~If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

~~When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

---

~~SR 3.0.4 Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.~~

---

~~3.7 SPENT FUEL PIT REQUIREMENTS~~

~~3.7.11 Spent Fuel Pit Water Level~~

~~LCO 3.7.11 The Spent Fuel Pit water level shall be  $\geq$  23 ft over the top of irradiated fuel assemblies seated in the storage racks.~~

~~APPLICABILITY: During movement of irradiated fuel assemblies in the Spent Fuel Pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Spent Fuel Pit water level not within limit.</del>	<del>A.1 Suspend movement of irradiated fuel assemblies in the Spent Fuel Pit.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.11.1 Verify the Spent Fuel Pit water level is <math>\geq</math> 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.</del>	<del>7 days</del>

~~3.7 SPENT FUEL PIT REQUIREMENTS~~

~~3.7.12 Spent Fuel Pit Boron Concentration~~

~~LCO 3.7.12 The Spent Fuel Pit boron concentration shall be  $\geq$  2000 ppm.~~

~~APPLICABILITY: When fuel assemblies are stored in the Spent Fuel Pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Spent Fuel Pit boron concentration not within limit.</del>	<del>A.1 Suspend movement of fuel assemblies in the Spent Fuel Pit.</del>	<del>Immediately</del>
	<del><u>AND</u></del>	
	<del>A.2 Initiate action to restore Spent Fuel Pit boron concentration to within limit.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.12.1 Verify the Spent Fuel Pit boron concentration is within limit.</del>	<del>7 days</del>

~~3.7 SPENT FUEL PIT REQUIREMENTS~~

~~3.7.13 Spent Fuel Pit Storage~~

~~LCO 3.7.13 IP2 fuel assemblies stored in the Spent Fuel Pit shall be categorized in accordance with Table 3.7.13-1 or, if pre-categorized, Table 3.7.13-2.~~

~~IP3 fuel assemblies stored in the Spent Fuel Pit shall be categorized in accordance with Table 3.7.13-1 or, if pre-categorized, Table 3.7.13-3.~~

~~IP2 and IP3 fuel assembly storage locations within the Spent Fuel Pit shall be restricted to locations allowed by Figure 3.7.13-1 and its associated notes.~~

~~NOTE~~

~~Regarding Category 5 fuel assemblies that are required by Figure 3.7.13-1 to contain a full length RCCA. The RCCA must not be placed in or removed while the assembly is in an RCCA required location unless all 8 adjacent cells are empty.~~

~~APPLICABILITY: Whenever any fuel assembly is stored in the Spent Fuel Pit.~~

~~ACTIONS~~

~~NOTE~~

~~Separate Condition entry is allowed for each fuel assembly.~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Requirements of the LCO not met.</del>	<del>A.1 Initiate action to move the noncomplying fuel assembly to an acceptable location.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.13.1</del> <del>Verify by administrative means that the IP2 fuel assembly has been categorized in accordance with Table 3.7.13-1 or, if pre-categorized, Table 3.7.13-2 and meets the requirements for the intended storage location.</del>  <del><u>OR</u></del> <del>Verify by administrative means that the IP3 fuel assembly has been categorized in accordance with Table 3.7.13-1 or, if pre-categorized, Table 3.7.13-3 and meets the requirements for the intended storage location.</del>	<del>Prior to storing the fuel assembly in the Spent Fuel Pit.</del>  <del>Prior to storing the fuel assembly in the Spent Fuel Pit.</del>

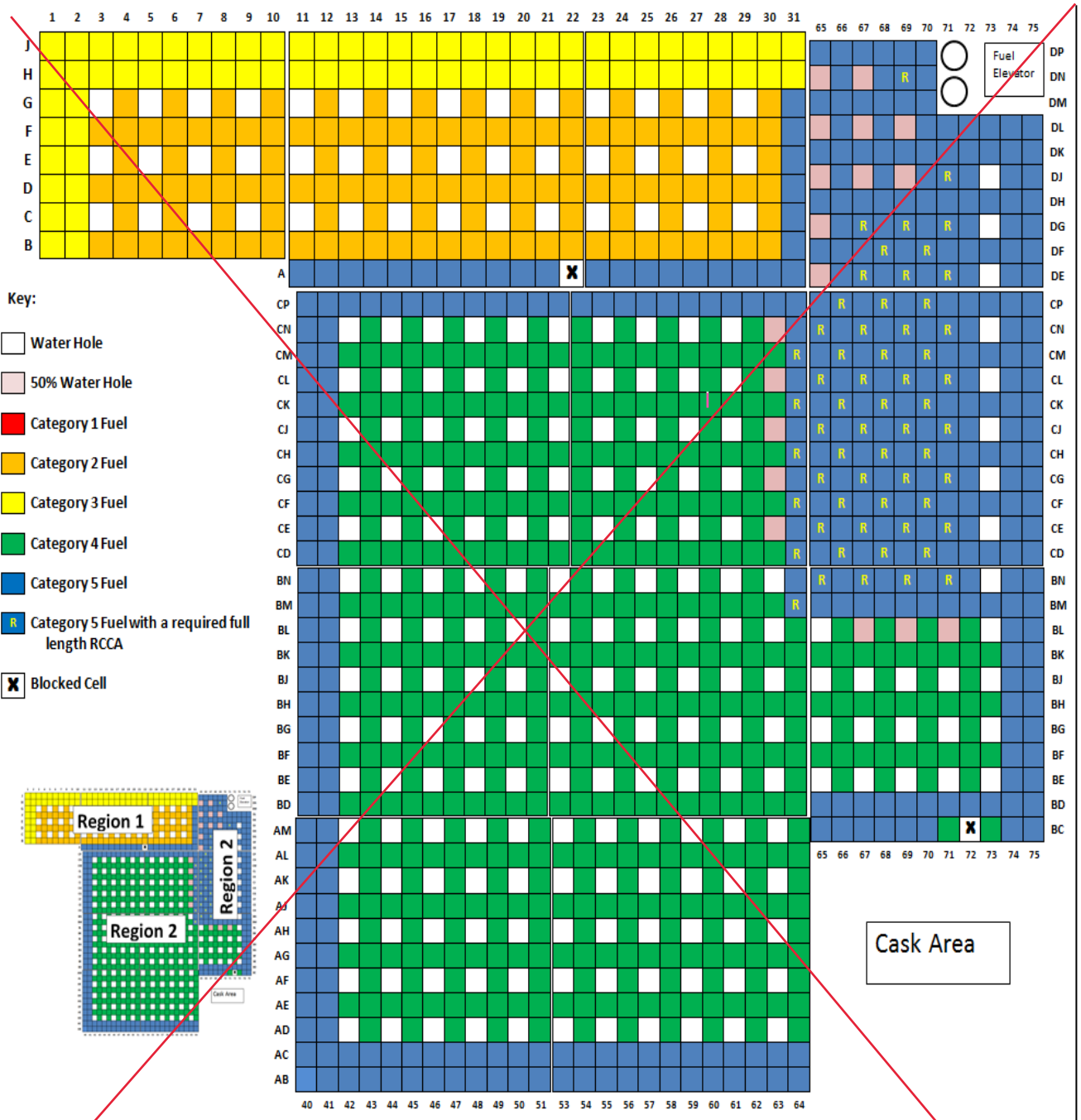
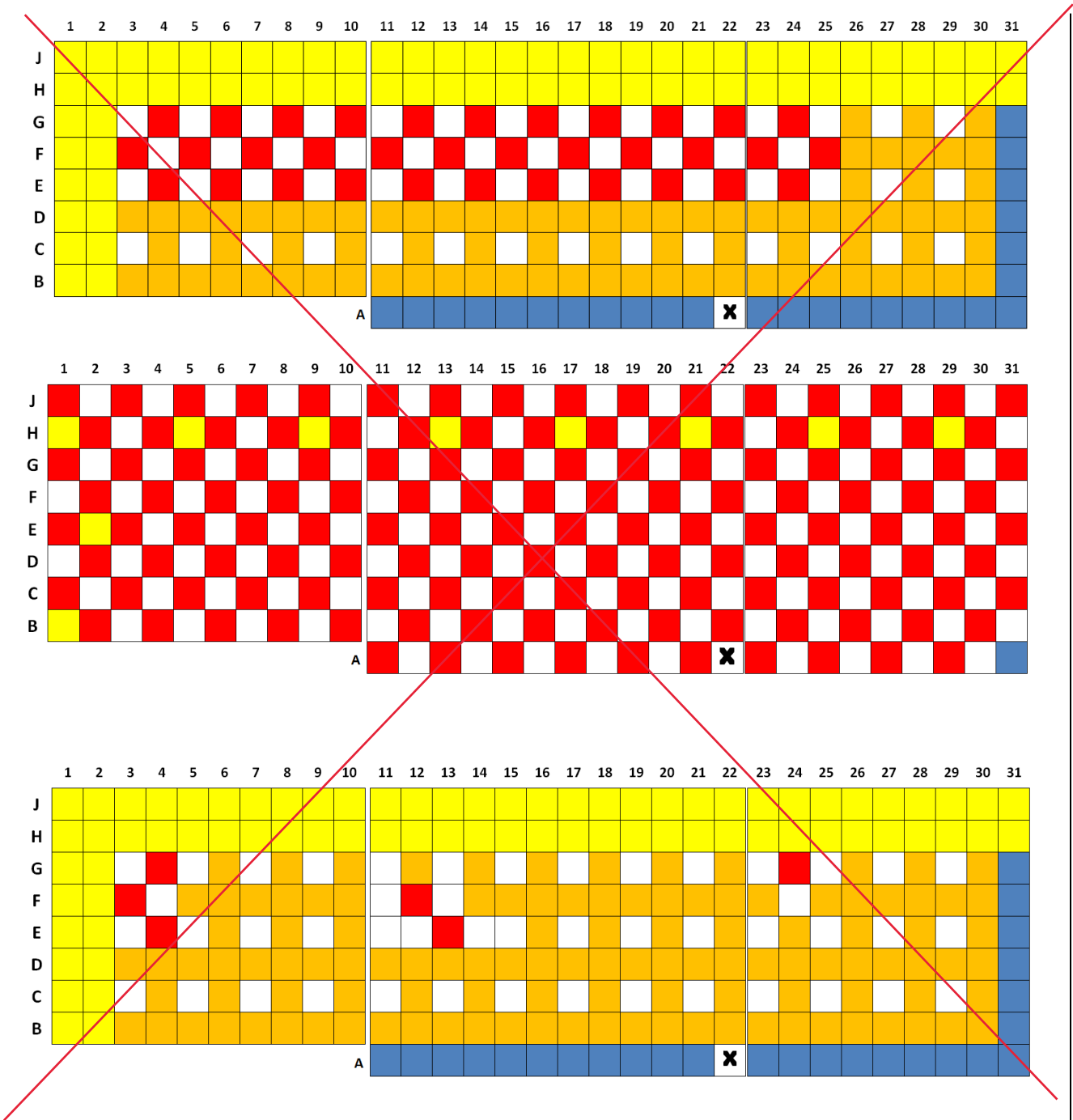


Figure 3.7.13-1 (page 1 of 2)  
Allowable Spent Fuel Pit Storage Locations for Category 1 through Category 5 Fuel Assemblies in Regions 1 and 2

~~-Notes-~~

- ~~1. Fuel assembly Categories are ranked in order of relative reactivity, from Category 1 to 5. Category 1 fuel assemblies have the highest reactivity, and Category 5 fuel assemblies have the lowest.~~
- ~~2. Fuel assembly categorization for assembly IDs after X for IP2 and after AA for IP3 must be performed in accordance with Table 3.7.13-1.~~
- ~~3. Fuel assembly Categories for IP2 assembly IDs A through X are located in Table 3.7.13-2.~~
- ~~4. Fuel assembly Categories for IP3 assembly IDs A through AA are located in Table 3.7.13-3.~~
- ~~5. Fuel assemblies of any higher numbered Category can be stored in any cell location that allows for a lower numbered Category. For example, a Category 5 fuel assembly can be stored in Category 1, 2, 3, 4, and 5 cells. Any cell may be empty.~~
- ~~6. Category 1 fuel assemblies that contain a full length RCCA may be stored in any Category 4, 3, 2, or 1 cell.~~
- ~~7. Category 2, 3 or 4 fuel assemblies that contain a full length RCCA may be stored in any Category 5 cell that does not require an inserted RCCA or in any Category 4, 3, 2, or 1 cell.~~
- ~~8. A Water Hole may contain up to 50% of absorber material by volume in the active fuel area. Stainless steel and Inconel meet the definition of absorber material. There is no restriction for non-actinide material outside of the active fuel area.~~
- ~~9. A 50% Water Hole may contain up to 50% of any non-actinide material by volume in the active fuel area. Zirconium meets the definition of non-actinide material. There is no restriction for non-actinide material outside of the active fuel area.~~
- ~~10. A Blocked Cell has the same requirements as a Water Hole.~~
- ~~11. A checkerboard area consists of every other cell being a Water Hole.~~
- ~~12. An area of Category 1 fuel assemblies may be formed in Region 1. The Category 1 area must be formed by replacing the Region 1 arrangement shown in this figure with an area of Category 1 fuel assemblies in accordance with the following criteria (see examples in Figure 3.7.13-2):~~
  - ~~a) Category 1 fuel assemblies must be face adjacent to at least three Water Holes and not face adjacent to another Category 1 assembly.~~
  - ~~b) Category 2 fuel assemblies must not be face adjacent to more than one Category 1 fuel assembly.~~
  - ~~c) Category 3 and Category 5 locations in Figure 3.7.13-1 may not be moved.~~
- ~~13. A checkerboard area of Category 1 fuel assemblies may be formed in Region 2. All four sides of the checkerboard area must be rows of Water Holes.~~
- ~~14. The edge of Region 2 next to the pool wall or cask loading area can be considered to be a row of Water Holes.~~

~~Figure 3.7.13-1 (page 2 of 2)~~



~~Figure 3.7.13-2~~  
~~Examples of Allowable Spent Fuel Pit Storage Locations for Category 1 Fuel Assemblies in~~  
~~Region 1~~

~~Table 3.7.13-1  
Fuel Assembly Reactivity Categorization for Assembly IDs after X for IP2 and after AA for IP3~~

Reactivity Category	Minimum Required Burnup (MRB) (GWd/T) <sup>(a)(b)(c)</sup>
1	0 <sup>(d)</sup>
2	21
3	28.5
4	$B_{1.2} = (a1 + a2 * E + a3 * E^2) \times \exp[-(a4 + a5 * E + a6 * E^2) \times CT] + a7 + a8 * E + a9 * E^2$ $B_{0.8} = (b1 + b2 * E + b3 * E^2) \times \exp[-(b4 + b5 * E + b6 * E^2) \times CT] + b7 + b8 * E + b9 * E^2$ $MRB = B_{0.8} + (B_{1.2} - B_{0.8}) \times (PF - 0.8) / 0.4$
5	MRB for Category 4 plus 11

Where:

E is enrichment in wt% U-235<sup>(e)</sup>,

CT is cooling time in years<sup>(f)</sup>, and

PF is the average peaking factor defined by the fuel assembly burnup divided by the sum of the cycle burnups for the cycles the fuel assembly was in the core.

and:

Coefficient	Value	Coefficient	Value
a1	-6.26824	b1	15.1405
a2	5.29367	b2	-4.81133
a3	-0.37154	b3	0.753855
a4	0.129582	b4	0.121252
a5	-0.0204918	b5	-0.0150991
a6	0.00205596	b6	0.00127009
a7	-0.13331	b7	-16.2293
a8	6.9037	b8	14.0159
a9	0.122068	b9	-0.687054

~~(a) 2 GWd/T must be added to the MRB for any fuel assembly that had a Hafnium insert.~~

~~(b) 4 GWd/T must be added to the MRB for any fuel assembly that was reconstituted without replacing removed fuel rods with stainless steel rods.~~

~~(c) 0.2, 0.3, 0.6, and 0.9 GWd/T must be added to the MRB for Categories 2, 3, 4, and 5, respectively, if the multi-cycle burnup averaged soluble boron concentration of 950 ppm is exceeded.~~

~~(d) With 64 IFBA rods or more. Assemblies with enrichments less than or equal to 4.5, 4.0, 3.5, and 3.0 require only 48, 32, 16, and 0 IFBA rods, respectively.~~

- ~~(e) Fuel assemblies at enrichments less than 4.2 wt% U-235 must use 4.2 wt% U-235 in the Category 4 equation.~~
- ~~(f) Fuel assemblies with cooling times of more than 25 years must use 25 years in the Category 4 equation.~~

~~Table 3.7.13-2 (page 1 of 3)  
Fuel Assembly Reactivity Categorization for Assembly IDs A through X for IP2~~

<del>Indian Point Unit 2 Fuel</del>					
<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>
<del>A01-A65</del>	<del>4</del>	<del>E43-E55</del>	<del>4</del>	<del>K01-K13</del>	<del>4</del>
		<del>E56</del>	<del>3</del>	<del>K14-K15</del>	<del>5</del>
<del>B01-B07</del>	<del>4</del>	<del>E57-E60</del>	<del>4</del>	<del>K16-K57</del>	<del>4</del>
<del>B08-B13</del>	<del>5</del>			<del>K58</del>	<del>5</del>
<del>B14-B23</del>	<del>4</del>	<del>F01</del>	<del>3</del>	<del>K59-K68</del>	<del>4</del>
<del>B24-B26</del>	<del>5</del>	<del>F02-F20</del>	<del>4</del>		
<del>B27-B64</del>	<del>4</del>	<del>F21</del>	<del>3</del>	<del>L01-L07</del>	<del>4</del>
		<del>F22-F30</del>	<del>4</del>	<del>L08-L10</del>	<del>5</del>
<del>C01-C04</del>	<del>4</del>	<del>F31-F34</del>	<del>5</del>	<del>L11-L63</del>	<del>4</del>
<del>C05-C06</del>	<del>5</del>	<del>F35</del>	<del>4</del>	<del>L64</del>	<del>3</del>
<del>C07-C12</del>	<del>4</del>	<del>F36</del>	<del>3</del>	<del>L65-L68</del>	<del>4</del>
<del>C13</del>	<del>5</del>	<del>F37-F39</del>	<del>4</del>		
<del>C14</del>	<del>4</del>	<del>F40</del>	<del>3</del>	<del>M01-M04</del>	<del>4</del>
<del>C15-C18</del>	<del>5</del>	<del>F41-F49</del>	<del>4</del>	<del>M05</del>	<del>5</del>
<del>C19-C28</del>	<del>4</del>	<del>F50</del>	<del>3</del>	<del>M06-M08</del>	<del>4</del>
<del>C29</del>	<del>5</del>	<del>F51-F60</del>	<del>4</del>	<del>M09</del>	<del>5</del>
<del>C30-C64</del>	<del>4</del>	<del>F61</del>	<del>3</del>	<del>M10-M12</del>	<del>4</del>
		<del>F62-F64</del>	<del>4</del>	<del>M13-M14</del>	<del>5</del>
<del>D01-D25</del>	<del>4</del>	<del>F65</del>	<del>3</del>	<del>M15-M20</del>	<del>4</del>
<del>D26</del>	<del>5</del>	<del>F66</del>	<del>4</del>	<del>M21</del>	<del>5</del>
<del>D27-D60</del>	<del>4</del>	<del>F67-F68</del>	<del>5</del>	<del>M22-M23</del>	<del>4</del>
<del>D61-D68</del>	<del>5</del>			<del>M24</del>	<del>5</del>
<del>D69-D72</del>	<del>4</del>	<del>G01-G05</del>	<del>4</del>	<del>M25-M27</del>	<del>4</del>
		<del>G06</del>	<del>5</del>	<del>M28</del>	<del>5</del>
<del>E01-E14</del>	<del>4</del>	<del>G07-G37</del>	<del>4</del>	<del>M29-M30</del>	<del>4</del>
<del>E15</del>	<del>3</del>	<del>G38</del>	<del>5</del>	<del>M31</del>	<del>5</del>
<del>E16-E19</del>	<del>5</del>	<del>G39-G72</del>	<del>4</del>	<del>M32-M34</del>	<del>4</del>
<del>E20</del>	<del>4</del>			<del>M35</del>	<del>5</del>
<del>E21-E24</del>	<del>5</del>	<del>H01-H38</del>	<del>4</del>	<del>M36-M37</del>	<del>4</del>
<del>E25-E27</del>	<del>4</del>	<del>H39-H51</del>	<del>5</del>	<del>M38-M44</del>	<del>5</del>
<del>E28-E31</del>	<del>5</del>	<del>H52-H54</del>	<del>4</del>	<del>M45</del>	<del>3</del>
<del>E32-E33</del>	<del>4</del>	<del>H55</del>	<del>5</del>	<del>M46</del>	<del>4</del>
<del>E34-E35</del>	<del>5</del>	<del>H56</del>	<del>4</del>	<del>M47-M48</del>	<del>5</del>
<del>E36-E40</del>	<del>4</del>			<del>M49-M50</del>	<del>4</del>
<del>E41-E42</del>	<del>5</del>	<del>J01-J68</del>	<del>4</del>	<del>M51-M52</del>	<del>5</del>

~~Table 3.7.13-2 (page 2 of 3)  
Fuel Assembly Reactivity Categorization for Assembly IDs A through X for IP2~~

<del>Indian Point Unit 2 Fuel</del>					
<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>
<del>M53-M54</del>	<del>4</del>	<del>Q71-Q73</del>	<del>4</del>	<del>T42-T43</del>	<del>4</del>
<del>M55-M56</del>	<del>5</del>	<del>Q74-Q76</del>	<del>5</del>	<del>T44-T46</del>	<del>5</del>
<del>M57</del>	<del>4</del>	<del>Q77</del>	<del>4</del>	<del>T47</del>	<del>4</del>
<del>M58-M59</del>	<del>5</del>	<del>Q78</del>	<del>5</del>	<del>T48</del>	<del>5</del>
<del>M60</del>	<del>4</del>	<del>Q79-Q80</del>	<del>4</del>	<del>T49-T51</del>	<del>4</del>
<del>M61</del>	<del>3</del>			<del>T52-T53</del>	<del>5</del>
<del>M62-M63</del>	<del>4</del>	<del>R01-R07</del>	<del>5</del>	<del>T54</del>	<del>4</del>
<del>M64</del>	<del>3</del>	<del>R08</del>	<del>4</del>	<del>T55</del>	<del>5</del>
<del>M65</del>	<del>4</del>	<del>R09-R38</del>	<del>5</del>	<del>T56-T72</del>	<del>4</del>
<del>M66</del>	<del>5</del>	<del>R39</del>	<del>4</del>	<del>T73-T80</del>	<del>5</del>
<del>M67</del>	<del>3</del>	<del>R40-R43</del>	<del>5</del>		
<del>M68</del>	<del>5</del>	<del>R44-R50</del>	<del>4</del>		
<del>M69-M71</del>	<del>4</del>	<del>R51-R69</del>	<del>5</del>	<del>U01-U04</del>	<del>5</del>
<del>M72</del>	<del>5</del>	<del>R70</del>	<del>4</del>	<del>U05</del>	<del>4</del>
		<del>R71-R72</del>	<del>5</del>	<del>U06-U13</del>	<del>5</del>
<del>N01-N08</del>	<del>4</del>	<del>R73-R74</del>	<del>4</del>	<del>U14</del>	<del>4</del>
<del>N09-N12</del>	<del>5</del>	<del>R75-R79</del>	<del>5</del>	<del>U15-U16</del>	<del>5</del>
<del>N13-N14</del>	<del>4</del>	<del>R80-R81</del>	<del>4</del>	<del>U17-U21</del>	<del>4</del>
<del>N15-N16</del>	<del>5</del>	<del>R82</del>	<del>5</del>	<del>U22</del>	<del>5</del>
<del>N17-N23</del>	<del>4</del>	<del>R83-R85</del>	<del>4</del>	<del>U23</del>	<del>4</del>
<del>N24-N32</del>	<del>5</del>			<del>U24-U49</del>	<del>5</del>
<del>N33-N47</del>	<del>4</del>	<del>S01-S44</del>	<del>5</del>	<del>U50</del>	<del>4</del>
<del>N48</del>	<del>5</del>	<del>S45</del>	<del>4</del>	<del>U51</del>	<del>5</del>
<del>N49-N80</del>	<del>4</del>	<del>S46-S47</del>	<del>5</del>	<del>U52</del>	<del>4</del>
		<del>S48</del>	<del>4</del>	<del>U53-U61</del>	<del>5</del>
<del>P01-P02</del>	<del>4</del>	<del>S49-S61</del>	<del>5</del>	<del>U62-U64</del>	<del>4</del>
<del>P03</del>	<del>3</del>	<del>S62</del>	<del>4</del>	<del>U65</del>	<del>5</del>
<del>P04-P47</del>	<del>4</del>	<del>S63-S65</del>	<del>5</del>	<del>U66-U68</del>	<del>4</del>
<del>P48</del>	<del>5</del>	<del>S66</del>	<del>4</del>	<del>U69-U73</del>	<del>5</del>
<del>P49-P60</del>	<del>4</del>	<del>S67-S77</del>	<del>5</del>		
<del>P61-P72</del>	<del>5</del>			<del>V01-V16</del>	<del>5</del>
				<del>V17-V29</del>	<del>4</del>
<del>Q01-Q65</del>	<del>5</del>	<del>T01-T32</del>	<del>5</del>	<del>V30-V35</del>	<del>5</del>
<del>Q66</del>	<del>4</del>	<del>T33-T34</del>	<del>4</del>	<del>V36</del>	<del>4</del>
<del>Q67-Q68</del>	<del>5</del>	<del>T35-T36</del>	<del>5</del>	<del>V37-V38</del>	<del>5</del>
<del>Q69</del>	<del>4</del>	<del>T37</del>	<del>3</del>	<del>V39</del>	<del>4</del>
<del>Q70</del>	<del>5</del>	<del>T38-T41</del>	<del>5</del>	<del>V40-V41</del>	<del>5</del>

~~Table 3.7.13-2 (page 3 of 3)  
Fuel Assembly Reactivity Categorization for Assembly IDs A through X for IP2~~

<del>Indian Point Unit 2 Fuel</del>					
<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>	<del>Assembly ID</del>	<del>Category</del>
<del>V42-V43</del>	<del>4</del>	<del>W21</del>	<del>5</del>	<del>X01-X02</del>	<del>3</del>
<del>V44-V49</del>	<del>5</del>	<del>W22</del>	<del>4</del>	<del>X03-X04</del>	<del>5</del>
<del>V50</del>	<del>4</del>	<del>W23</del>	<del>5</del>	<del>X05-X37</del>	<del>4</del>
<del>V51-V54</del>	<del>5</del>	<del>W24</del>	<del>4</del>	<del>X38</del>	<del>5</del>
<del>V55-V57</del>	<del>4</del>	<del>W25</del>	<del>5</del>	<del>X39-X49</del>	<del>4</del>
<del>V58-V61</del>	<del>5</del>	<del>W26</del>	<del>4</del>	<del>X50-X51</del>	<del>5</del>
<del>V62</del>	<del>4</del>	<del>W27</del>	<del>5</del>	<del>X52-X53</del>	<del>4</del>
<del>V63</del>	<del>5</del>	<del>W28-W34</del>	<del>4</del>	<del>X54-X55</del>	<del>5</del>
<del>V64-V65</del>	<del>4</del>	<del>W35</del>	<del>5</del>	<del>X56-X58</del>	<del>4</del>
<del>V66-V67</del>	<del>5</del>	<del>W36-W38</del>	<del>4</del>	<del>X59-X60</del>	<del>5</del>
<del>V68</del>	<del>4</del>	<del>W39</del>	<del>5</del>	<del>X61-X62</del>	<del>4</del>
<del>V69-V77</del>	<del>5</del>	<del>W40</del>	<del>4</del>	<del>X63</del>	<del>5</del>
<del>V78-V79</del>	<del>4</del>	<del>W41-W43</del>	<del>5</del>	<del>X64-X65</del>	<del>4</del>
<del>V80-V81</del>	<del>5</del>	<del>W44-W45</del>	<del>4</del>	<del>X66</del>	<del>5</del>
<del>V82</del>	<del>4</del>	<del>W46</del>	<del>5</del>	<del>X67</del>	<del>4</del>
<del>V83</del>	<del>5</del>	<del>W47</del>	<del>4</del>	<del>X68-X69</del>	<del>5</del>
<del>V84</del>	<del>4</del>	<del>W48-W49</del>	<del>5</del>	<del>X70-X73</del>	<del>4</del>
<del>V85</del>	<del>5</del>	<del>W50</del>	<del>4</del>	<del>X74</del>	<del>5</del>
<del>V86</del>	<del>4</del>	<del>W51</del>	<del>5</del>	<del>X75</del>	<del>4</del>
<del>V87-V88</del>	<del>5</del>	<del>W52-W55</del>	<del>4</del>	<del>X76</del>	<del>5</del>
<del>V89</del>	<del>4</del>	<del>W56-W58</del>	<del>5</del>	<del>X77</del>	<del>4</del>
<del>V90-V91</del>	<del>5</del>	<del>W59-W60</del>	<del>4</del>	<del>X78</del>	<del>5</del>
<del>V92</del>	<del>4</del>	<del>W61</del>	<del>5</del>	<del>X79</del>	<del>4</del>
		<del>W62</del>	<del>4</del>	<del>X80-X93</del>	<del>5</del>
<del>W01-W10</del>	<del>4</del>	<del>W63-W67</del>	<del>5</del>	<del>X94-X95</del>	<del>4</del>
<del>W11</del>	<del>5</del>	<del>W68</del>	<del>4</del>	<del>X96</del>	<del>5</del>
<del>W12-W15</del>	<del>4</del>	<del>W69-W71</del>	<del>5</del>		
<del>W16</del>	<del>5</del>	<del>W72</del>	<del>4</del>	<del>FRSB<sup>1</sup></del>	<del>4</del>
<del>W17</del>	<del>4</del>	<del>W73-W83</del>	<del>5</del>		
<del>W18-W19</del>	<del>5</del>	<del>W84</del>	<del>4</del>		
<del>W20</del>	<del>4</del>	<del>W85-W93</del>	<del>5</del>		

~~1 FRSB is the Fuel Rod Storage Basket~~

~~Table 3.7.13-3  
Fuel Assembly Reactivity Categorization for Fuel Assembly IDs A through AA for IP3~~

Indian Point Unit 3 Fuel					
Assembly ID	Category	Assembly ID	Category	Assembly ID	Category
V43	3	V48	3		
All other Fuel Assembly IDs A through AA are Category 4					

## 4.0 DESIGN FEATURES

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### 4.1 Site Location

IP2

~~Indian Point 2~~ is located on the East bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York. The site is approximately 24 miles north of the New York City boundary line. The nearest city is Peekskill which is 2.5 miles northeast of Indian Point.

~~The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone, as defined in 10 CFR 100.3, is 520 meters and 1100 meters, respectively. For the purpose of satisfying 10 CFR Part 20, the "Restricted Area" is the same as the "Exclusion Area" shown in the Defueled Safety Analysis Report (DSAR), Figure 2.2.2.~~

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### 4.2 Deleted

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### 4.3 Fuel Storage

IP2

Spent fuel shall not be stored in the ~~Spent Fuel Pit~~.

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XXX

Note: TS 5.1.1 will be relocated to the DSAR.

5.0 ~~ADMINISTRATIVE CONTROLS~~

5.1 ~~Responsibility~~

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5.1.1 ~~The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.~~

~~The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~

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Note: TS 5.2.1 will be relocated to the DSAR.

NOTE: This page is from the LAR submitted on August 2, 2022

## 5.0 ADMINISTRATIVE CONTROLS

### 5.2 Organization

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#### 5.2.1 Onsite and Offsite Organizations

~~Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.~~

- ~~a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all decommissioning organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the facility specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the DSAR,~~
- ~~b. The plant manager shall be responsible for overall safe maintenance of the facility and shall have control over those onsite activities necessary for storage and maintenance of nuclear fuel.~~
- ~~c. The corporate officer with direct responsibility for IP2 shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel, and~~
- ~~d. The individuals who train the individuals that carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.~~

Note: TS 5.3.1 will be relocated to the DSAR.

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.3 Facility Staff Qualifications~~

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~~5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the IPEC Quality Assurance Program Manual (QAPM).~~

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Note: TS 5.4.1 will be relocated to the DSAR

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.4 Procedures~~

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- ~~5.4.1~~ Written procedures shall be established, implemented, and maintained covering the following activities:
- ~~a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the DSAR;~~
  - ~~b. Deleted;~~
  - ~~c. Quality assurance for effluent and environmental monitoring;~~
  - ~~d. Deleted;~~
  - ~~e. All programs specified in Technical Specification 5.5; and~~
  - ~~f. Personnel radiation protection consistent with the requirements of 10 CFR 20.~~
-

Note: TS 5.5.1 will be relocated to the DSAR.

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

---

~~The following programs shall be established, implemented, and maintained.~~

~~5.5.1 Offsite Dose Calculation Manual (ODCM)~~

- ~~a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program, and~~
- ~~b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Technical Specification 5.6.2 and Technical Specification 5.6.3.~~
- ~~c. Licensee initiated changes to the ODCM:
  - ~~1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
    - ~~a. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and~~
    - ~~b. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,~~~~
  - ~~2. Shall become effective after the approval of the plant manager, and~~
  - ~~3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.~~~~

## 5.5 Programs and Manuals

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### 5.5.2 Deleted

### 5.5.3 Radioactive Effluent Controls Program

~~This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:~~

- ~~a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,~~
- ~~b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001–20.2402,~~
- ~~c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,~~
- ~~d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit/facility to unrestricted areas, conforming to 10 CFR 50, Appendix I,~~
- ~~e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,~~
- ~~f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,~~
- ~~g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:~~
  - ~~1. For noble gases: a dose rate  $\leq$  500 mrem/yr to the whole body and a dose rate  $\leq$  3000 mrem/yr to the skin and~~

~~5.5 Programs and Manuals~~

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~~5.5.3 Radioactive Effluent Controls Program (continued)~~

Note: TS 5.5.3 will be relocated to the DSAR.

- ~~2. For iodine 131, tritium, and all radionuclides in particulate form with half lives greater than 8 days: a dose rate  $\leq$  1500 mrem/yr to any organ,~~
- ~~h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,~~
- ~~i. Limitations on the annual and quarterly doses to a member of the public from iodine 131, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and~~
- ~~j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.~~

~~5.5.4 Deleted~~

~~5.5.5 Deleted~~

~~5.5.6 Deleted~~

~~5.5.7 Deleted~~

~~5.5.8 Deleted~~

~~5.5.9 Deleted~~

~~5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program~~

Note: TS 5.5.10 will be relocated to the DSAR.

~~This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11.5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."~~

Note: TS 5.5.10 will be relocated to the DSAR.

~~5.5 Programs and Manuals~~

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~~5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)~~

~~The program shall include:~~

- ~~a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);~~
- ~~b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of  $\geq 0.5$  rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents, and~~
- ~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.~~

~~5.5.11 Deleted~~

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~~5.5 Programs and Manuals~~

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~~5.5.12 Technical Specifications (TS) Bases Control Program~~

~~This program provides a means for processing changes to the Bases of these Technical Specifications.~~

- ~~a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.~~
  - ~~b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
    - ~~1. A change in the TS incorporated in the license or~~
    - ~~2. A change to the DSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.~~~~
  - ~~c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the DSAR.~~
  - ~~d. Proposed changes that meet the criteria of Technical Specification 5.5.12b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).~~
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~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.6 Reporting Requirements~~

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~~The following reports shall be submitted in accordance with 10 CFR 50.4.~~

~~5.6.1 Not Used~~

~~5.6.2 Annual Radiological Environmental Operating Report~~

~~NOTE~~

~~A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station.~~

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~~The Annual Radiological Environmental Operating Report covering the operation of the unit/facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.~~

~~A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.~~

~~5.6.3 Radioactive Effluent Release Report~~

~~NOTE~~

~~A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station; however, for units/facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit/facility.~~

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~~The Radioactive Effluent Release Report covering the operation of the unit/facility in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.~~

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Note: TS 5.6.2 will be relocated to the DSAR.

Note: TS 5.6.3 will be relocated to the DSAR.

5.7 High Radiation Area

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5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
  - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
  - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
  1. All such door and gate keys shall be maintained under the administrative control of the ~~IP3 shift manager~~, radiation protection manager, or his or her designee.
  2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

lead licensee  
representative on shift

XXX

**HDI-IPEC-22-076**

**Enclosure - Attachment 2**

Markup Pages of the IP3 FL, PDTS, and Appendix C TS Pages

H. The issuance of this renewed license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

~~I. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.~~

2. Accordingly, Renewed Facility License No. DPR-64 is hereby issued to Holtec IP3 and HDI to read as follows:

A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by Holtec IP3 and maintained by HDI. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Defueled Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended.

B. Subject to the conditions and requirements incorporated herein, the Commission licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec IP3 to possess and use, and (b) HDI to possess and use the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;

(2) HDI pursuant to the Act and 10 CFR Part 70, to possess, at any time, special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;

(3) HDI pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;

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- (4) HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components;
  - (5) HDI pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:
- (1) Deleted per Amendment No. 270
  - (2) Technical Specifications  
The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 271, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.
- D. (DELETED)
- E. (DELETED)
- F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.
- G. HDI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and to the authority of 10 CFR 50.90 and CFR 50.54(p). The combined set of plans<sup>1</sup> for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

<sup>1</sup> The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The HDI CSP was approved by License Amendment No. 243, as supplemented by changes approved by License Amendment Nos. 254, 260, and 263.

HDI has been granted Commission authorization to use “stand alone preemption authority” under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. HDI shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

~~H. Deleted per Amendment No. 270~~

~~I. DELETED~~

~~J. DELETED~~

~~K. DELETED~~

~~L. DELETED~~

~~M. DELETED~~

~~N. DELETED~~

~~O. Deleted per Amendment No. 270~~

~~P. Deleted~~

~~Q. DELETED~~

~~R. DELETED~~

~~S. DELETED~~

~~T. DELETED~~

~~U. DELETED~~

~~V. DELETED~~

~~W. Deleted~~

~~X. Deleted~~

~~AA. Deleted per Amendment No. 270~~

~~AB. Deleted per Amendment No. 270~~

~~AC. Mitigation Strategy License Condition~~

~~The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:~~

- ~~(a) Fire fighting response strategy with the following elements:~~
  - ~~1. Pre defined coordinated fire response strategy and guidance~~
  - ~~2. Assessment of mutual aid fire fighting assets~~
  - ~~3. Designated staging areas for equipment and materials~~
  - ~~4. Command and control~~
  - ~~5. Training of response personnel~~
  
- ~~(b) Operations to mitigate fuel damage considering the following:~~
  - ~~1. Protection and use of personnel assets~~
  - ~~2. Communications~~
  - ~~3. Minimizing fire spread~~
  - ~~4. Procedures for implementing integrated fire response strategy~~
  - ~~5. Identification of readily available pre staged equipment~~
  - ~~6. Training on integrated fire response strategy~~
  - ~~7. Spent fuel pool mitigation measures~~
  
- ~~(c) Actions to minimize release to include consideration of:~~
  - ~~1. Water spray scrubbing~~
  - ~~2. Dose to onsite responders~~

~~AD. Deleted per Amendment No. 270~~

~~AE. HDI may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. HDI is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by HDI and Holtec IP3.~~

~~AF. License Renewal License Conditions~~

- ~~(1) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3," (SER) and supplements to the SER, are collectively the "License Renewal UFSAR Supplement." The UFSAR Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in the UFSAR Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.~~
- ~~(2) The License Renewal UFSAR Supplement, as defined in license condition AF(1) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).~~
- ~~a. The licensee shall implement those new programs and enhancements to existing programs no later than the date specified in the License Renewal UFSAR Supplement.~~
  - ~~b. The licensee shall complete those activities no later than the date specified in the License Renewal UFSAR Supplement.~~
3. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

*/RA/*

Ho K. Nieh, Director  
Office of Nuclear Reactor Regulation

ISFSI Only

Attachments:

- Appendix A - ~~Permanently Defueled~~ Technical Specifications
- Appendix B - Environmental Technical Specification Requirements
- Appendix C - ~~Inter Unit Fuel Transfer~~ Technical Specifications

Date of Issuance: September 17, 2018

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Amendment No. 271

APPENDIX A

TO

RENEWED

FACILITY LICENSE DPR-64

~~PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS AND BASES~~

ISFSI ONLY

FOR THE

INDIAN POINT 3 NUCLEAR GENERATING STATION UNIT NO. 3

WESTCHESTER COUNTY, NEW YORK

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3)

AND HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

DOCKET NO. 50-286

Date of Issuance:  
April 15, 1976

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Amendment No. ~~271~~

RENEWED

Facility License No. DPR-64

Appendix A – ~~Permanently Defueled~~ Technical Specifications

ISFSI ONLY

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1.0 USE AND APPLICATION

1.1 Definitions

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NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

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<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER (CFH)	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER training and retraining program required by TS 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.

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1.0 USE AND APPLICATION

1.2 Logical Connectors

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**PURPOSE**            ~~The purpose of this section is to explain the meaning of logical connectors.~~

~~Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Required Actions and Surveillances. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.~~

---

**BACKGROUND**       ~~Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.~~

~~When logical connectors are used to state a Surveillance, only the first level of logic is used, and the logical connector is left justified with the statement of the Surveillance.~~

---

**EXAMPLE**            ~~The following example illustrates the use of logical connectors.~~

EXAMPLE 1.2-1

ACTIONS

<u>CONDITION</u>	<u>REQUIRED ACTION</u>	<u>COMPLETION TIME</u>
<del>A. LCO not met...</del>	<del>A.1    Verify...  <u>AND</u>  A.2    Restore...</del>	

~~In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.~~

---

1.0 USE AND APPLICATION

1.3 Completion Times

---

PURPOSE	<del>The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.</del>
BACKGROUND	<del>Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe handling and storage of spent nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).</del>
DESCRIPTION	<del>The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO.</del>  <del>Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.</del>
EXAMPLE	<del>The following example illustrates the use of Completion Times with different Required Actions.</del>

---

1.3 Completion Time

EXAMPLE (continued)

EXAMPLE 1.3-1

ACTIONS

<u>CONDITION</u>	<u>REQUIRED ACTION</u>	<u>COMPLETION TIME</u>
A. Spent fuel pit boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pit.	Immediately
	<u>AND</u> A.2 Initiate action to restore spent fuel pit boron concentration to within limit.	Immediately

Condition A has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion time is referenced to the time that Condition A is entered.

The Required Actions of Condition A are to immediately suspend movement of fuel assemblies in the spent fuel pit and initiate action to restore spent fuel pit boron concentration within limit.

IMMEDIATE COMPLETION TIME When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

**PURPOSE**            ~~The purpose of this section is to define the proper use and application of Frequency requirements.~~

**DESCRIPTION**       ~~Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.~~

~~The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.~~

**EXAMPLE**            ~~The following example illustrates the type of Frequency statement that appears in the Technical Specifications (TS).~~

EXAMPLE 1.4.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify level is within limits.	12 hours

~~Example 1.4.1 contains the type of SR encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.~~

~~If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the specified condition or the LCO is considered not met (in accordance with SR 3.0.1).~~

~~2.0 DELETED~~

---

~~3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY~~

---

---

~~LCO 3.0.1 LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.~~

---

~~LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.~~

~~If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.~~

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---

### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

~~SR 3.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on variables outside specified limits.~~

---

~~SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance.~~

---

~~SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.~~

~~If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

~~When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

---

~~SR 3.0.4 Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.~~

---

~~3.7 SPENT FUEL PIT REQUIREMENTS~~

~~3.7.14 Spent Fuel Pit Water Level~~

~~LCO 3.7.14            The spent fuel pit water level shall be  $\geq$  23 ft over the top of irradiated fuel assemblies seated in the storage racks.~~

~~APPLICABILITY:      During movement of irradiated fuel assemblies in the spent fuel pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Spent fuel pit water level not within limit.</del>	<del>A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pit.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.14.1    Verify the spent fuel pit water level is <math>\geq</math> 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.</del>	<del>7 days</del>

~~3.7 SPENT FUEL PIT REQUIREMENTS~~

~~3.7.15 Spent Fuel Pit Boron Concentration~~

~~LCO 3.7.15 The spent fuel pit boron concentration shall be  $\geq$  1000 ppm.~~

~~NOTE~~

~~During inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration."~~

~~APPLICABILITY: When fuel assemblies are stored in the spent fuel pit and a spent fuel pit verification has not been performed since the last movement of fuel assemblies in the spent fuel pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Spent fuel pit boron concentration not within limit.</del>	<del>A.1 Suspend movement of fuel assemblies in the spent fuel pit.</del>	<del>Immediately</del>
	<del>AND</del>	
	<del>A.2.1 Initiate action to restore spent fuel pit boron concentration to within limit.</del>	<del>Immediately</del>
	<del>OR</del>	
	<del>A.2.2 Initiate action to perform a spent fuel pit verification.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.15.1 Verify the spent fuel pit boron concentration is within limit.</del>	<del>31 days</del>

### ~~3.7 SPENT FUEL PIT REQUIREMENTS~~

#### ~~3.7.16 Spent Fuel Assembly Storage~~

~~LCO 3.7.16~~

~~Fuel assemblies stored in the spent fuel pit shall be classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup; and,~~

~~Fuel assembly storage location within the spent fuel pit shall be restricted based on the Figure 3.7.16-1 classification as follows:~~

- ~~a. Fuel assemblies classified as Type 2 may be stored in any location in either Region 1 or Region 2;~~
- ~~b. Fuel assemblies classified as Type 1A, 1B or 1C shall be stored in Region 1;~~
- ~~c. Fuel assembly storage location within Region 1 shall be restricted as follows:~~
  - ~~1. Type 1A assemblies may be stored anywhere in Region 1;~~
  - ~~2. Type 1B assemblies may be stored anywhere in Region 1, except a Type 1B assembly shall not be stored face adjacent to a Type 1C assembly;~~
  - ~~3. Type 1C assemblies shall not be stored in Row 64 or in Column ZZ; and~~
  - ~~4. Type 1C assemblies shall be stored in Region 1 locations where all face adjacent locations are as follows:~~
    - ~~a) occupied by Type 2 or Type 1A assemblies, or~~
    - ~~b) occupied by non fuel components, or~~
    - ~~c) empty.~~

~~APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Requirements of the LCO not met.</del>	<del>A.1 Initiate action to move fuel to restore compliance with LCO 3.7.16.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.7.16.1 Verify by administrative means the initial enrichment and burnup of each fuel assembly and that the storage location meets LCO 3.7.16 requirements.</del>	<del>Prior to storing the fuel assembly in the spent fuel pit</del>

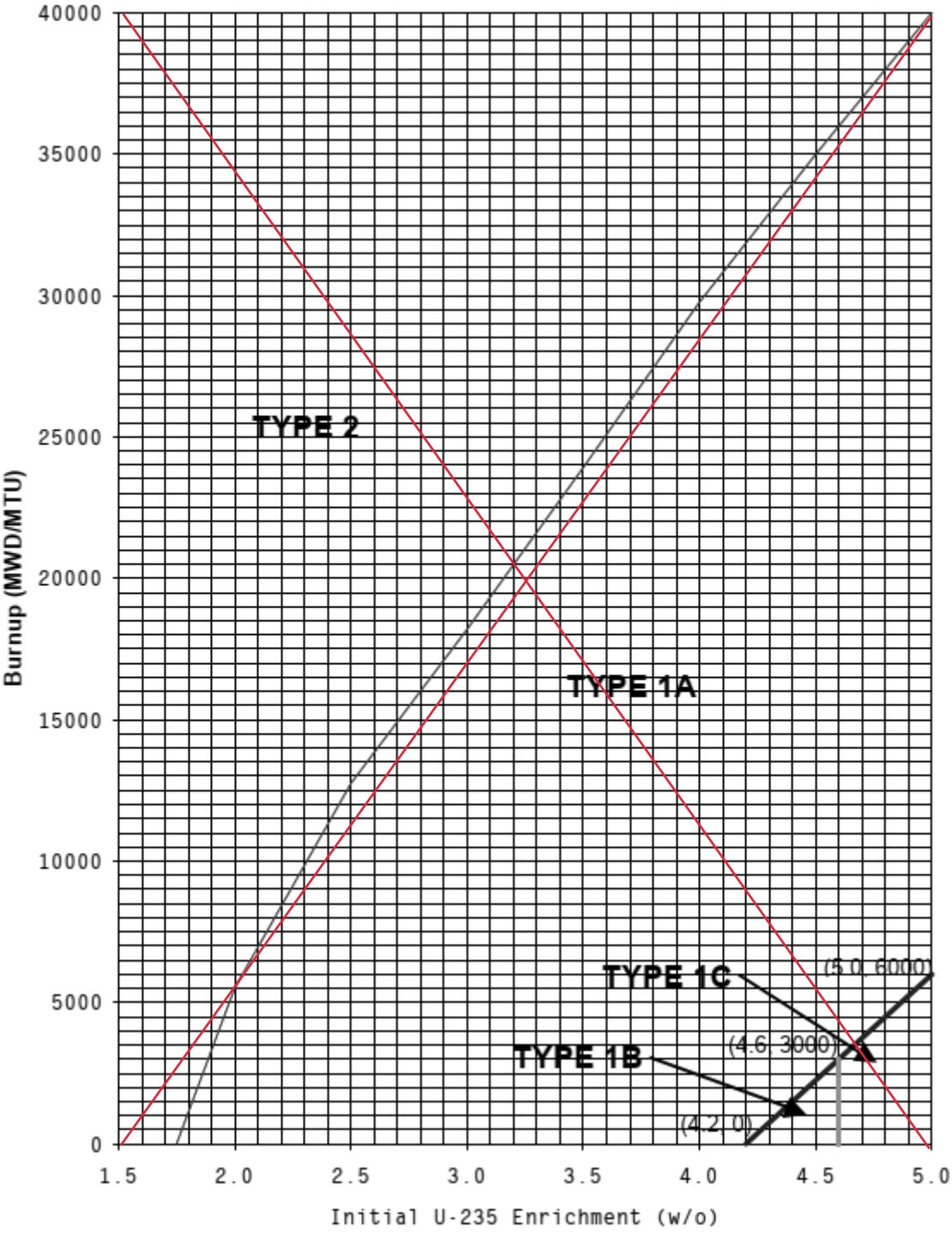


Figure 3.7.16-1 (Page 1 of 1)

Fuel Assembly Classification for Storage in the Spent Fuel Pit

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## 4.0 DESIGN FEATURES

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### 4.1 Site Location

IP3

~~Indian Point 3~~ is located on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York. The site is approximately 24 miles north of the New York City boundary line. The nearest city is Peekskill which is 2.5 miles northeast of Indian Point.

~~The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 350 meters and 1100 meters, respectively.~~

---

### 4.2 Deleted

---

### 4.3 Fuel Storage

Spent fuel shall not be stored in the IP3 Spent Fuel Pit.

#### ~~4.3.1 Criticality~~

- ~~4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:~~
- ~~a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;~~
  - ~~b.  $k_{eff} \leq 0.95$  if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage;~~
  - ~~c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II);~~
  - ~~d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I).~~

#### ~~4.3.2 Drainage~~

~~The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below a nominal elevation of 88 ft.~~

#### ~~4.3.3 Capacity~~

~~The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 1345 fuel assemblies.~~

---

Note: TS 5.1.1 will be relocated to the DSAR.

Responsibility  
5.1

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.1 Responsibility~~

---

~~5.1.1 The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.~~

~~The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~

~~5.1.2 The shift manager (SM) shall be responsible for the shift command function.~~

---

Note: TS 5.2.1 will be relocated to the DSAR.

5.0 ~~ADMINISTRATIVE CONTROLS~~

5.2 ~~Organization~~

---

5.2.1 ~~Onsite and Offsite Organizations~~

~~Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.~~

- ~~a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all decommissioning organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the facility specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the DSAR and Quality Assurance Plan, as appropriate;~~
- ~~b. The plant manager shall be responsible for overall safe maintenance of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of nuclear fuel;~~
- ~~c. The corporate officer with direct responsibility for IP3 shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel; and~~
- ~~d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.~~

5.2 Organization

---

5.2.2 Facility Staff

The facility staff organization shall include the following:

- a. ~~Each duty shift shall be composed of at least one shift manager and one NON CERTIFIED OPERATOR. The NON CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.~~

~~At least one person qualified to stand watch in the control room (NON CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in the spent fuel pool.~~

- b. ~~Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:~~

- ~~1) No fuel movements are in progress;~~
- ~~2) No movement of loads over fuel are in progress; and~~
- ~~3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.~~

- c. ~~An individual qualified in radiation protection procedures shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.~~

- d. ~~Not Used.~~

- e. ~~The shift manager shall be a CERTIFIED FUEL HANDLER.~~

- f. ~~Deleted.~~
-

Note: TS 5.3.1 will be relocated to the DSAR.

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.3 Facility Staff Qualifications~~

---

- ~~5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the HDI Quality Assurance Program Manual (QAPM).~~
  - ~~5.3.2 An NRC approved training and retraining program for CERTIFIED FUEL HANDLERS shall be maintained.~~
-

Note: TS 5.4.1 will be relocated to the DSAR

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.4 Procedures~~

---

~~5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:~~

- ~~a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the DSAR;~~
  - ~~b. Deleted;~~
  - ~~c. Quality assurance for effluent and environmental monitoring;~~
  - ~~d. Deleted; and~~
  - ~~e. All programs specified in Specification 5.5.~~
-

Note: TS 5.5.1 will be relocated to the DSAR.

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

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~~The following programs shall be established, implemented, and maintained.~~

~~5.5.1 Offsite Dose Calculation Manual (ODCM)~~

- ~~a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and~~
- ~~b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.~~
- ~~c. Licensee-initiated changes to the ODCM:
  - ~~1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
    - ~~(a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s); and~~
    - ~~(b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;~~~~
  - ~~2. Shall become effective after the approval of the plant manager; and~~
  - ~~3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.~~~~

~~5.5.2 Deleted~~

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Note: TS 5.5.4 will be relocated to the DSAR.

5.5 Programs and Manuals

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5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

~~This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:~~

- ~~a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;~~
- ~~b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;~~
- ~~c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;~~
- ~~d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit/facility to unrestricted areas, conforming to 10 CFR 50, Appendix I;~~
- ~~e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;~~
- ~~f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;~~

Note: TS 5.5.4 will be relocated to the DSAR.

5.5 Programs and Manuals

---

5.5.4 ~~Radioactive Effluent Controls Program (continued)~~

- g. ~~Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:~~
  - a. ~~For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and~~
  - b. ~~For iodine 131, tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to dose rate of 1500 mrem/yr to any organ.~~
- h. ~~Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;~~
- i. ~~Limitations on the annual and quarterly doses to a member of the public from iodine 131, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and~~
- j. ~~Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.~~

5.5.5 through ~~Deleted~~  
5.5.10

Note: TS 5.5.11 will be relocated to the DSAR.

5.5 Programs and Manuals

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5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

~~This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."~~

~~The program shall include:~~

- ~~a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);~~
- ~~b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank shall be limited to less than the amount that would result in a whole body exposure of  $\geq 0.5$  rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and~~
- ~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.~~

5.5.12 Deleted

5.5 Programs and Manuals

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5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. ~~Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.~~
  - b. ~~Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:~~
    - 1. ~~a change in the TS incorporated in the license; or~~
    - 2. ~~a change to the DSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.~~
  - c. ~~The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the DSAR.~~
  - d. ~~Proposed changes that do not meet the criteria of Specification 5.5.13.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).~~
-

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.6 Reporting Requirements~~

---

~~The following reports shall be submitted in accordance with 10 CFR 50.4.~~

~~5.6.1 Not Used~~

~~5.6.2 Annual Radiological Environmental Operating Report~~

Note: TS 5.6.2 will be relocated to the DSAR.

~~NOTE~~

~~A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station.~~

---

~~The Annual Radiological Environmental Operating Report covering the operation of the unit/facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.~~

~~A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.~~

~~5.6.3 Radioactive Effluent Release Report~~

Note: TS 5.6.3 will be relocated to the DSAR.

~~NOTE~~

~~A single submittal may be made for a multiple unit/facility station. The submittal shall combine sections common to all units/facilities at the station; however, for units/facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit/facility.~~

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~~The Radioactive Effluent Release Report covering the operation of the unit/facility in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.I.~~

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5.7 High Radiation Area

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5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
  1. All such door and gate keys shall be maintained under the administrative control of the ~~shift supervisor~~, radiation protection manager, or his or her designee.
  2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following facility radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
  1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
  2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

lead licensee  
representative on shift

~~APPENDIX C~~

~~TO~~

~~FACILITY LICENSE~~

~~FOR~~

~~HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND  
HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)~~

~~INDIAN POINT NUCLEAR~~

~~GENERATING UNIT No. 3~~

~~INTER UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS~~

~~PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM~~

~~FACILITY LICENSE NO. DPR 64~~

~~DOCKET NO. 50-286~~

~~Amendment No. 271~~

~~SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM~~

~~1.0 DESCRIPTION~~

~~The spent fuel transfer system consists of the following components: (1) a spent fuel shielded transfer canister (STC), which contains the fuel; (2) a transfer cask (HI TRAC 100D) (hereafter referred to as HI TRAC), which contains the STC during transfer operations; and (3) a bottom missile shield.~~

~~The STC and HI TRAC are designed to transfer irradiated nuclear fuel assemblies from the Indian Point 3 (IP3) spent fuel pit to the Indian Point 2 (IP2) spent fuel pit. A fuel basket within the STC holds the fuel assemblies and provides criticality control. The shielded transfer canister provides the confinement boundary, water retention boundary, gamma radiation shielding, and heat rejection capability. The HI TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The STC contains up to 12 fuel assemblies.~~

~~The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are  $\frac{3}{4}$  inch steel,  $2\frac{3}{4}$  inch lead and  $\frac{3}{4}$  inch steel, respectively. The canister closure incorporates two O-ring seals to ensure its confinement function. The confinement system consists of the canister inner shell, bottom plate, top flange, top lid, top lid O-ring seals, vent port seal and cover plate, and drain port seal and coverplate. The fuel basket, for the transfer of 12 Pressurized Water Reactor (PWR) fuel assemblies, is a fully welded, stainless steel, honeycomb structure with neutron absorber panels attached to the individual storage cell walls under stainless steel sheathing. The maximum gross weight of the fully loaded STC is 40 tons.~~

~~The HI TRAC is a multi-layer steel and lead cylinder with a bolted bottom (or pool) and top lid. For the fuel transfer operation the HI TRAC is fitted with a solid top lid, an STC centering assembly, and a bottom missile shield. The inner shell of the transfer cask forms an internal cylindrical cavity for housing the STC. The outer surface of the cask inner shell is buttressed with intermediate lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are  $\frac{3}{4}$  inch steel,  $2\frac{7}{8}$  inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.~~

~~The HI TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The bottom missile shield is attached to the bottom of the HI TRAC and provides tornado missile protection of the pool lid bolted joint. The HI TRAC can withstand a tornado missile in other areas without the need for additional shielding. The STC centering assembly provides STC position control within the HI TRAC and also acts as an internal impact limiter in the event of a non-mechanistic tipover accident.~~

## ~~2.0 CONDITIONS~~

### ~~2.1 OPERATING PROCEDURES~~

~~Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, maintenance, and recovery from off normal conditions such as crane hang up. The written operating procedures shall be consistent with the technical basis described in Chapter 10 of the Licensing Report (Holtec International Report HI-2094289).~~

### ~~2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM~~

~~Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 8 of the Licensing Report (Holtec International Report HI-2094289).~~

### ~~2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE~~

~~A training exercise of the loading, closure, handling/transfer, and unloading, of the equipment shall be conducted prior to the first transfer. The training exercise shall not be conducted with irradiated fuel. The training exercise may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The training exercise shall include, but is not limited to the following:~~

- ~~a) Moving the STC into the IP3 spent fuel pool.~~
- ~~b) Preparation of the HI TRAC for STC loading.~~
- ~~c) Selection and verification of specific fuel assemblies and non-fuel hardware to ensure type conformance.~~
- ~~d) Loading specific assemblies and placing assemblies into the STC (using a single dummy fuel assembly), including appropriate independent verification.~~
- ~~e) Remote installation of the STC lid and removal of the STC from the spent fuel pool.~~
- ~~f) Placement of the STC into the HI TRAC with the STC centering assembly.~~
- ~~g) STC closure, establishment of STC water level with steam, verification of STC water level, STC leakage testing, and operational steps required prior to transfer, as applicable.~~
- ~~h) Establishment and verification of HI TRAC water level.~~
- ~~i) Installation of the HI TRAC top lid.~~
- ~~j) HI TRAC closure, leakage testing, and operational steps required prior to transfer, as applicable.~~
- ~~k) Movement of the HI TRAC with STC from the IP3 fuel handling building to the IP2 fuel handling building along the haul route with designated devices.~~
- ~~l) Moving the STC into the IP2 spent fuel pool.~~
- ~~m) Manual crane operations for bare STC movements including demonstration of recovery from a crane hang up with the STC suspended from the crane.~~

~~APPENDIX C~~

~~TO~~

~~FACILITY LICENSE~~

~~FOR~~

~~HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND~~

~~HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)~~

~~INDIAN POINT NUCLEAR~~

~~GENERATING UNIT No. 3~~

~~INTER UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS~~

~~PART II: TECHNICAL SPECIFICATIONS~~

~~FACILITY LICENSE NO. DPR-64~~

~~DOCKET NO. 50-286~~

~~Amendment No. 271~~

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1.0 USE AND APPLICATION

1.1 Definitions

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
<del>ACTIONS</del>	<del>ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.</del>
<del>INTACT FUEL ASSEMBLIES</del>	<del>INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).</del>
<del>LOADING OPERATIONS</del>	<del>LOADING OPERATIONS include all licensed activities on an STC while it is being loaded with fuel assemblies and while the STC is being placed in the HI TRAC. LOADING OPERATIONS begin when the first fuel assembly is placed in the STC and end when the HI TRAC is suspended from or secured on the TRANSPORTER.</del>
<del>NON FUEL HARDWARE (NFH)</del>	<del>NFH is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs), Hafnium Flux Suppressors, and Instrument Tube Tie Rods (ITTRs).</del>
<del>TRANSFER OPERATIONS</del>	<del>TRANSFER OPERATIONS include all licensed activities performed on a HI TRAC loaded with one or more fuel assemblies when it is being moved after LOADING OPERATIONS or before UNLOADING OPERATIONS. TRANSFER OPERATIONS begin when the HI TRAC is first suspended from or secured on the TRANSPORTER and end when the TRANSPORTER is at its destination and the HI TRAC is no longer secured on or suspended from the TRANSPORTER.</del>
<del>TRANSPORTER</del>	<del>TRANSPORTER is the device or vehicle which moves the HI TRAC. The TRANSPORTER can either support the HI TRAC from underneath or the HI TRAC can be suspended from it.</del>

(continued)

1.1 Definitions (continued)

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<u>Term</u>	<u>Definition</u>
UNLOADING OPERATIONS	<del>UNLOADING OPERATIONS include all licensed activities on an STC or HI-TRAC while it is being unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the HI-TRAC is no longer suspended from or secured on the TRANSPORTER and end when the last fuel assembly is removed from the STC.</del>
ZR	<del>ZR means any zirconium-based fuel cladding authorized for use in a commercial nuclear power plant reactor.</del>

---

1.0 USE AND APPLICATION

1.2 Logical Connectors

---

**PURPOSE**            The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

---

**BACKGROUND**        Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

---

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2.1

ACTIONS

<u>CONDITION</u>	<u>REQUIRED ACTION</u>	<u>COMPLETION TIME</u>
A. LCO not met.	A.1 VERIFY ... <u>AND</u> A.2 Restore ...	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES  
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop ...  OR A.2.1 Verify ...  AND A.2.2.1 Reduce ...  OR A.2.2.2 Perform ...  OR A.3 Remove ...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

---

<b>PURPOSE</b>	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
<b>BACKGROUND</b>	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
<b>DESCRIPTION</b>	<p>The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the Spent Fuel Shielded Transfer Canister and Transfer Cask System is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the Spent Fuel Shielded Transfer Canister and Transfer Cask System is not within the LCO Applicability.</p> <p>Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.</p>

---

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	<u>AND</u> B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 ~~Completion Times (continued)~~

~~EXAMPLES  
(continued)~~

~~EXAMPLE 1.3-2~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. One system not within limit.</del>	<del>A.1 Restore system to within limit.</del>	<del>7 days</del>
<del>B. Required Action and associated Completion Time not met.</del>	<del>B.1 Complete action B.1.</del>	<del>12 hours</del>
	<del>AND B.2 Complete action B.2.</del>	<del>36 hours</del>

~~When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.~~

~~(continued)~~

1.3 Completion Times (continued)

EXAMPLES  
(continued)

~~EXAMPLE 1.3.3~~

~~ACTIONS~~

~~NOTE~~

~~Separate Condition entry is allowed for each component.~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. LCO not met.</del>	<del>A.1 Restore compliance with LCO.</del>	<del>4 hours</del>
<del>B. Required Action and associated Completion Time not met.</del>	<del>B.1 Complete action B.1.</del> <del>AND</del> <del>B.2 Complete action B.2.</del>	<del>6 hours</del>  <del>12 hours</del>

~~The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.~~

~~The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.~~

~~IMMEDIATE  
COMPLETION  
TIME~~

~~When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.~~

1.0 USE AND APPLICATION

1.4 Frequency

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PURPOSE	<del>The purpose of this section is to define the proper use and application of Frequency requirements.</del>
DESCRIPTION	<del>Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.</del> <del>The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.</del> <del>Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.</del>

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(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4.1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency (continued)

EXAMPLES  
(continued)

~~EXAMPLE 1.4.2~~

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>Verify flow is within limits.</del>	<del>Once within 12 hours prior to starting activity  AND 24 hours thereafter</del>

~~Example 1.4.2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4.1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.~~

~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.~~

~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.~~

~~2.0 NOT USED~~

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~~This section is intentionally left blank~~

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~~3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY~~

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<del>LCO 3.0.1</del>	<del>LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.</del>
<del>LCO 3.0.2</del>	<del>Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5.</del>  <del>If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.</del>
<del>LCO 3.0.3</del>	<del>Not applicable.</del>
<del>LCO 3.0.4</del>	<del>When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an STC.</del>
<del>LCO 3.0.5</del>	<del>Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.</del>

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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~~SR 3.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.~~

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~~SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.~~

~~For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance.~~

~~Exceptions to this Specification are stated in the individual Specifications.~~

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~~SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.~~

~~If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

~~When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.~~

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~~SR 3.0.4 Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an STC.~~

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~~3.1 INTER-UNIT FUEL TRANSFER~~

~~3.1.1 Boron Concentration~~

~~LCO 3.1.1            The boron concentration of the water in the Spent Fuel Pit and the STC shall be  $\geq$  2000 ppm.~~

~~APPLICABILITY:    Whenever one or more fuel assemblies are in the STC.~~

~~NOTE~~  
~~Only applicable to the spent fuel pit when the STC is in the spent fuel pit~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. Boron concentration not within limit.</del>	<del>A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS.</del>	<del>Immediately</del>
	<del>AND</del>	
	<del>A.2 Suspend positive reactivity additions.</del>	<del>Immediately</del>
	<del>AND</del>	
	<del>A.3 Initiate action to restore boron concentration to within limit.</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>NOTE</del>	<del>Once, within 4 hours prior to entering the Applicability of this LCO.</del>
<del>This surveillance is only required to be performed if the STC is submerged in water in the spent fuel pool or if water is added to, or recirculated through, the STC when the STC is in the HI TRAC. Any added water must meet the boron concentration requirement of LCO 3.1.1.</del>	<del>AND</del>
<del>SR 3.1.1.1            Verify the boron concentration is within limit using two separate measurements.</del>	<del>Once per 48 hours thereafter.</del>

### ~~3.1 INTER-UNIT FUEL TRANSFER~~

#### ~~3.1.2 Shielded Transfer Canister (STC) Loading~~

~~LCO 3.1.2 INTACT FUEL ASSEMBLIES placed into the Shielded Transfer Canister (STC) shall be classified in accordance with Table 3.1.2-1 based on initial enrichment and burnup and shall be restricted based on the following:~~

- ~~a. INTACT FUEL ASSEMBLIES classified as Type 2 may be placed in the STC basket (see Figure 3.1.2-1) with the following restrictions:~~
- ~~1. Post irradiation cooling time, initial enrichment, and allowable average burnup shall be within the limits for the cell locations as specified in Table 3.1.2-3;~~
  - ~~2. Decay heat including NON FUEL HARDWARE  $\leq 1.2$  kW (any cell);~~
  - ~~3. Total STC Decay heat from all cell locations including NON FUEL HARDWARE  $\leq 0.621$  kW;~~
  - ~~4. Post irradiation cooling time and the maximum average burnup of NON FUEL HARDWARE shall be within the cell locations and limits specified in Table 3.1.2-2. In accordance with Table 3.1.2-2 RCCAs and Hafnium Flux Suppressors cannot be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket.~~

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~~-NOTE-~~

~~If one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, AND 4 must be empty, with a cell blocker installed that prevents inserting fuel assemblies and/or NON FUEL HARDWARE.~~

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- ~~b. INTACT FUEL ASSEMBLIES classified as Type 1 or Type 2 may be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket (see Figure 3.1.2-1) with the following restrictions:~~
- ~~1. Post irradiation cooling time, initial enrichment, and allowable average burnup shall be within the limits for the cell locations as specified in Table 3.1.2-3;~~
  - ~~2. Decay heat including NON FUEL HARDWARE  $\leq 1.2$  kW;~~
  - ~~3. Post irradiation cooling time and the maximum average burnup of NON FUEL HARDWARE shall be within the cell locations and limits specified in Table 3.1.2-2. In accordance with Table 3.1.2-2 RCCAs and Hafnium Flux Suppressors cannot be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket.~~
- ~~c. Only INTACT FUEL ASSEMBLIES with initial average enrichment  $\leq 4.4$  wt% U-235 and discharged prior to IP3 Cycle 12 shall be placed in the STC basket. IP3 fuel assemblies V43 and V48 shall not be selected for transfer.~~

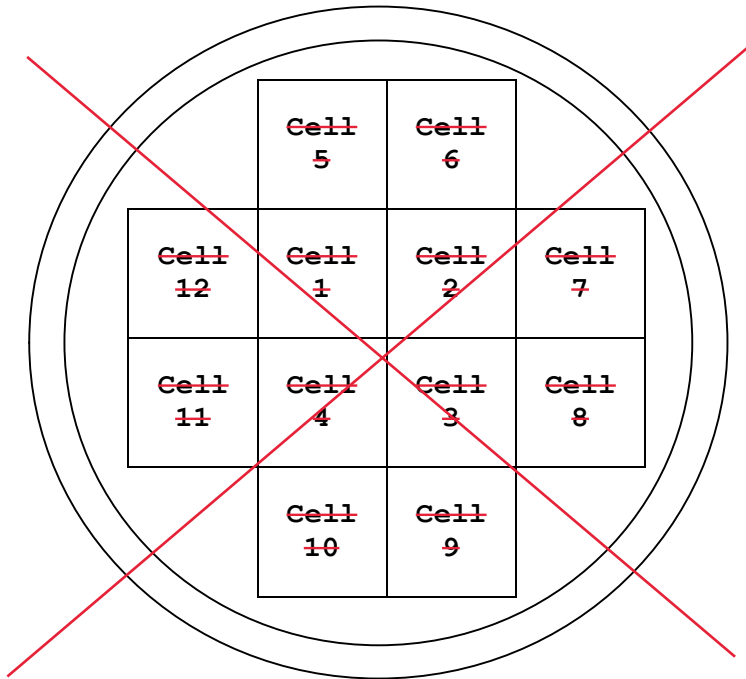
~~APPLICABILITY: Whenever one or more fuel assemblies are in the STC.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<p><del>A. One or more fuel assemblies or NON FUEL HARDWARE in the STC do not meet the LCO limits.</del></p>	<p><del>A.1.1 Initiate action to restore compliance with LCO 3.1.2.</del></p> <p><del>OR</del></p> <p><del>A.1.2 Initiate action to move fuel to the IP3 spent fuel pit in accordance with IP3 Appendix A Technical Specification LCO 3.7.16.</del></p>	<p><del>Immediately</del></p>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<p><del>SR 3.1.2.1 Verify by administrative means that the fuel assembly and NON FUEL HARDWARE meets the requirements specified in the LCO for placement in the STC.</del></p>	<p><del>Prior to placing the fuel assembly in the STC.</del></p>
<p><del>SR 3.1.2.2 Verify by visual inspection that a cell blocker which prevents inserting fuel assemblies and/or NON FUEL HARDWARE into cells 1, 2, 3, and 4 of the STC is installed.</del></p>	<p><del>Prior to placing a Type 1 fuel assembly in the STC.</del></p>



~~Figure 3.1.2-1~~  
~~Shielded Transfer Canister Layout~~  
~~(Top View)~~

~~Table 3.1.2-1  
Minimum Burnup Requirements at Varying Initial Enrichments<sup>(a)</sup>~~

<del>Maximum Assembly Initial Enrichment<sup>(f)(g)</sup> (wt% U235)</del>	<del>Configuration A<sup>(e)</sup> Minimum Assembly Average Burnup (MWD/MTU)<sup>(b)</sup></del>	<del>Configuration B<sup>(d)</sup> Minimum Assembly Average Burnup (MWD/MTU)<sup>(b)</sup></del>
<del>2.0</del>	<del>5,400</del>	<del>6,000</del>
<del>2.5</del>	<del>13,800</del>	<del>18,800</del>
<del>3.0</del>	<del>22,100</del>	<del>28,600</del>
<del>3.5</del>	<del>30,000</del>	<del>37,300</del>
<del>4.0</del>	<del>36,900</del>	<del>44,600</del>
<del>4.5</del>	<del>42,700</del>	<del>52,500</del>
<del>5.0</del>	<del>48,700</del>	<del>Note (e)</del>

- ~~(a) Fuel that does not meet the minimum assembly average burnup at a given initial enrichment is classified as Type 1 fuel. Fuel that meets the minimum assembly average burnup at a given initial enrichment is classified as Type 2 fuel.~~
- ~~(b) Linear interpolation between enrichment levels to determine minimum burnup requirements is permitted.~~
- ~~(c) Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation or where it can be shown that the insertion did not exceed 8 inches below the top of the active fuel.~~
- ~~(d) Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation and where the insertion was more than 8 inches below the top of the active fuel. This configuration also applies to fuel assemblies that have contained a Hafnium Flux Suppressor.~~
- ~~(e) Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel.~~
- ~~(f) Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the maximum allowed initial average enrichment.~~
- ~~(g) Rounding to one decimal place to determine initial enrichment is not permitted.~~

Table 3.1.2-2

~~NON FUEL HARDWARE~~<sup>(a)</sup> ~~Post Irradiation Cooling Times and Allowable Average Burnup~~

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs <sup>(b-d)</sup>	TPDs <sup>(b)(e)</sup>	RCCAs	Hafnium Flux Suppressors
≥ 6	≤ 20000	N/A	≤ 630000	≤ 20000
≥ 7	-	≤ 20000	-	-
≥ 8	≤ 30000	-	-	≤ 30000
≥ 9	≤ 40000	≤ 30000	-	-
≥ 10	≤ 50000	≤ 40000	-	-
≥ 11	≤ 60000	≤ 45000	-	-
≥ 12	-	≤ 50000	-	-
≥ 13	-	≤ 60000	-	-
≥ 14	-	-	-	-
≥ 15	-	≤ 90000	-	-
≥ 16	-	≤ 630000	-	-
≥ 20	-	-	-	-
Allowed Quantity and Location	Up to twelve (12) per transfer in any location	Up to twelve (12) per transfer in any location	Up to four (4) per transfer in Cells 1, 2, 3, and/or 4	Up to four (4) per transfer in Cells 1, 2, 3, and/or 4

- (a) ~~NON FUEL HARDWARE burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post irradiation. NSAs are not authorized for loading in the STC.~~
- (b) ~~Linear interpolation between points is only permitted for BPRAs, WABAs, and TPDs, with the exception that interpolation is not permitted for TPDs with burnups greater than 90 GWd/MTU and cooling times greater than 15 years.~~
- (c) ~~N/A means not authorized for loading at this cooling time.~~
- (d) ~~Burnup and Cooling time limits in this column are only applicable to Loading Patterns 1-6 in Table 3.1.2-3. For Loading Patterns 7-12 in Table 3.1.2-3, the burnup and cooling time limits for a BPRAs are the same as those for the fuel assembly they are located in.~~

Table 3.1.2-3 (Sheet 1 of 2)

Allowable STC Loading Configurations

Configuration <sup>(e)</sup>	Cells 1, 2, 3, 4 <sup>(a)(b)</sup>	Cells 5, 6, 7, 8, 9, 10, 11, 12 <sup>(a)(b)</sup>
1	<del>Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235</del>	<del>Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 25 years Initial Enrichment ≥ 2.3 wt% U-235</del>
2	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235</del>	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235</del>
3	<del>Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235</del>	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235</del>
4	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.6 wt% U-235</del>	<del>Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235</del>
5	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.4 wt% U-235</del>	<del>Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235</del>
6	<del>Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235</del>	<del>Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 2.3 wt% U-235</del>

Table 3.1.2.3 (Sheet 2 of 2)

Allowable STC Loading Configurations

Configuration <sup>(c)</sup>	Cells 1, 2, 3, 4 <sup>(a)(b)</sup>	Cells 5, 6, 7, 8, 9, 10, 11, 12 <sup>(a)(b)</sup>
7	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
8	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 15 years Initial Enrichment ≥ 3.4 wt% U-235
9	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 11 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
10	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 15 years Initial Enrichment ≥ 3.4 wt% U-235
11	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 6 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.2 wt% U-235
12	Burnup ≤ 60,000 MWD/MTU Cooling time ≥ 9 years Initial Enrichment ≥ 4.2 wt% U-235	Burnup ≤ 50,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.6 wt% U-235

(a) Initial enrichment is the assembly average enrichment. Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the minimum allowed initial average enrichment.

(b) Rounding to one decimal place to determine initial enrichment is permitted.

(c) Fuel with five middle Inconel spacers are limited to cells 1, 2, 3, and 4 for all loading configurations except loading configuration 6 which allows fuel with Inconel spacers in all cells.

~~3.1 INTER UNIT FUEL TRANSFER~~

~~3.1.3 Shielded Transfer Canister (STC) Initial Water Level~~

~~LCO 3.1.3            The established water level in the STC shall be 9.0+0.5/ 1.5 inches below the bottom of the STC lid.~~

~~APPLICABILITY:    Prior to TRANSFER OPERATIONS when the STC is in the HI TRAC and the STC lid has been installed.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A.    STC water level not within limit.</del>	<p style="text-align: center;"><del>————— NOTE —————</del></p> <p style="text-align: center;"><del>Water used for level restoration must meet the boron concentration requirement of LCO 3.1.1.</del></p> <p style="text-align: center;"><del>—————</del></p> <p><del>A.1    Initiate action to restore STC water level.</del></p>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.1.3.1    Verify the initial STC water level is within limit by verifying the following during STC water level establishment:</del>	<del>Once prior to TRANSFER OPERATIONS.</del>
<del>    a.    steam is emitted from the STC drain tube; and</del>	
<del>    b.    the volume of water removed is ≥ 35.4 gallons and ≤ 47.9 gallons.</del>	



~~ACTIONS (continued)~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>B. Required Action A.1.3 indicates a fuel misload.</del>	<del>B.1.1 Return the STC to the spent fuel pool and remove the STC lid.</del>  <del>AND</del>	12 hours
	<del>B.1.2 Return any misloaded fuel to the IP3 spent fuel pit in accordance with IP3 Appendix A Technical Specification LCO 3.7.16.</del>	24 hours
<del>G. Required Action A.1.3 does not indicate a fuel misload.</del>	<del>C.1 Develop and initiate corrective actions necessary to return the STC to compliance with LCO 3.1.3 and LCO 3.1.4.</del>	24 hours

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.1.4.1</del> <p style="text-align: center;"><del>NOTE</del></p> <del>Pressure measurements shall be taken once upon establishing required water level AND hourly thereafter for 24 hours. Pressure may initially drop during pressure stabilization.</del>  <del>Verify by direct measurement that the rate of STC cavity pressure rise is within limit.</del>	<del>Once prior to TRANSFER OPERATIONS.</del>
<del>SR 3.1.4.2</del> <p><del>Verify that an ASME code compliant pressure relief valve or rupture disc and two channels of pressure instrumentation with a range of at least 0.1 psia to 15 psia and calibrated to within 1% accuracy within the past 12 months are installed on the STC.</del></p>	<del>During performance of SR 3.1.4.1.</del>

~~3.1 INTER UNIT FUEL TRANSFER~~

~~3.1.5 Shielded Transfer Canister (STC) Unloading~~

~~NOTE~~

- ~~1. Only IP3 spent fuel assemblies are permitted to be in the STC.~~
- ~~2. Once each IP3 spent fuel assembly removed from the STC has been placed in an IP2 spent fuel rack location and disconnected from the spent fuel pit bridge crane, it may not be returned to the STC.~~

~~LCO 3.1.5 IP3 spent fuel assemblies transferred to IP2 via the STC must be either in an approved IP2 spent fuel pit storage rack location per IP2 Appendix A Technical Specification LCO 3.7.13, in their authorized STC fuel basket cell, or be in transit between these two locations.~~

~~APPLICABILITY: Whenever the STC is in the Unit 2 spent fuel pit.~~

~~ACTIONS~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. One or more fuel assemblies not in the required location.</del>	<del>A.1 Initiate action to restore compliance with LCO 3.1.5</del>	<del>Immediately</del>

~~SURVEILLANCE REQUIREMENTS~~

<del>SURVEILLANCE</del>	<del>FREQUENCY</del>
<del>SR 3.1.5.1 Verify by administrative means that a fuel assembly returned to the STC has been re-loaded into the same STC cell from which it was removed.</del>	<del>Once, after each re-loaded fuel assembly is returned to the STC.</del>

## 4.0 DESIGN FEATURES

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### 4.1 Inter Unit Fuel Transfer

#### 4.1.1 Fuel Assemblies

Fuel assemblies selected for inter unit transfer of fuel shall meet the fuel characteristics specified in Table 4.1.1.1.

#### 4.1.2 Criticality

4.1.2.1 The Shielded Transfer Canister (STC) is designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;
- b.  $k_{\text{eff}} \leq 0.95$  if fully flooded with unborated water;
- c. A nominal 9.218 inch center to center distance between fuel assemblies placed in the STC basket;
- d. Basket cell ID: 8.79 in. (nominal);
- e. Basket cell wall thickness: 0.28 in. (nominal);
- f.  $B_4C$  in the Metamic neutron absorber:  $\geq 31.5$  wt.% and  $\leq 33.0$  wt.%;
- g. The  $B_4C$  in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%;
- h. Metamic panel thickness:  $\geq 0.102$  in.;
- i. The size and location of the neutron absorber panels shall be in accordance with drawing 6015, revision 6, which can be found in the Licensing Report (Holtec International Report HI-2094289).

#### 4.1.2.2 Drainage

The STC is designed and shall be maintained to prevent inadvertent draining.

#### 4.1.2.3 Capacity

The STC is designed and shall be maintained with a capacity of no more than 12 fuel assemblies.

(continued)

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## 4.0 DESIGN FEATURES (continued)

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### 4.1.3 Codes and Standards

~~The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2004 Edition, is the governing Code for the STC, as clarified below, except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by these sections, provided a written reconciliation of the later edition against the 2004 Edition, is performed. Table 4.1.3-1 lists approved alternatives to the ASME Code for the design of the STC.~~

### 4.1.4 Geometric Arrangements and Process Variables

~~The following are geometric arrangements and process variables that require a one time verification as part of each inter unit fuel transfer operation:~~

- ~~1. LOADING OPERATIONS, TRANSFER OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures  $\geq 0^{\circ}\text{F}$ .~~
- ~~2. LOADING OPERATIONS shall only be conducted when the spent fuel pit water temperature and the fuel handling building ambient temperatures are both  $\leq 100^{\circ}\text{F}$ .~~
- ~~3. LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel assemblies.~~
- ~~4. LOADING OPERATIONS shall only be conducted when the irradiated fuel assemblies in the IP3 spent fuel pit have been subcritical for at least 90 days.~~
- ~~5. TRANSFER OPERATIONS shall only be conducted when the outside air temperature is  $\leq 100^{\circ}\text{F}$ .~~
- ~~6. TRANSFER OPERATIONS shall only be conducted when the STC trunnions are offset from the HI TRAC trunnions in the azimuthal direction by at least 30 degrees.~~
- ~~7. TRANSFER OPERATIONS shall only be conducted after STC seal leak tests have demonstrated no detected leakage when tested to a sensitivity of  $1 \times 10^{-3}$  ref  $\text{cm}^3/\text{s}$  in accordance with the "pre shipment" test requirements of ANSI N14.5.~~
- ~~8. Prior to installing the HI TRAC lid the HI TRAC water level shall be verified by two separate inspections to be within  $+0/ -1$  inch of the top of the STC lid.~~

(continued)

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4.0 DESIGN FEATURES (continued)

- ~~9. TRANSFER OPERATIONS shall only be conducted after the combined leak rate through the HI TRAC top lid and vent port cover seals are confirmed to be water tight using an acceptable leak test from ANSI N14.5 and the pool lid seal is verified to be water tight by visual inspection.~~
- ~~10. TRANSFER OPERATIONS shall not occur with a TRANSPORTER that contains > 50 gallons of diesel fuel.~~

~~Table 4.1.1-1~~

~~Fuel Assembly Characteristics~~

<del>Fuel Assembly Class</del>	<del>15x15<sup>(a)</sup></del>
<del>No. of Fuel Rod Locations</del>	<del>204</del>
<del>Gladding Type</del>	<del>ZR</del>
<del>Guide/Instrument Tube Type</del>	<del>ZR</del>
<del>Design Initial U (kg/assembly)</del>	<del>≤ 473</del>
<del>Fuel Rod Clad O.D. (in)</del>	<del>≥ 0. 422</del>
<del>Fuel Rod Clad I.D. (in)</del>	<del>≤ 0. 3734</del>
<del>Fuel Pellet Diameter (in)</del>	<del>≤ 0. 3659</del>
<del>Fuel Rod Pitch (in)</del>	<del>≤ 0.563</del>
<del>Active Fuel Length (in)</del>	<del>≤ 144</del>
<del>Fuel Assembly Length (in)</del>	<del>≤ 160</del>
<del>Fuel Assembly Width (in)</del>	<del>≤ 8.54</del>
<del>No. of Guide and/or Instrument Tubes</del>	<del>21</del>
<del>Guide/Instrument Tube Thickness (in)</del>	<del>≥ 0. 017</del>
<del>Axial Blanket Enrichment (wt % U-235)<sup>(b)</sup></del>	<del>≤ 3.2</del>
<del>Axial Blanket Length (in)<sup>(b)</sup></del>	<del>≥ 6</del>

~~(a) All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within the 15x15 class.~~

~~(b) Applicable only if axial blankets are present.~~

(continued)

4.0 DESIGN FEATURES (continued)

Table 4.1.3.1 (page 1 of 2)

List of ASME Code Alternatives for the STC

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
STC Confinement Boundary	ND-1000	Statement of requirements for Code stamping of components.	Cask confinement boundary is designed, and will be fabricated in accordance with ASME Code, Section III, Subsection ND to the maximum practical extent, but Code stamping is not required.
STC Confinement Boundary	ND-2000	Requires materials to be supplied by ASME-approved material supplier.	Holtec-approved suppliers will supply materials with CMTRs per ND-2000.
STC and STC basket assembly	ND-3100 NG-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The Licensing Report, serving as the Design Specification, establishes the service conditions and load combinations for fuel transfer.
STC Confinement Boundary	ND-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of cask vessel is as a radionuclide confinement boundary under normal and hypothetical accident conditions. Cask is designed to withstand maximum internal pressure and maximum accident temperatures.
STC Confinement Boundary	ND-8000	States requirement for name, stamping and reports per NCA-8000	STC to be marked and identified in accordance with drawing 6013 <sup>(a)</sup> . Code stamping is not required. QA data package prepared in accordance with Holtec's approved QA program.

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 2 of 2)

List of ASME Code Alternatives for the STC

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
STC Basket Assembly	NG-4420	NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	<p>Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal STC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1 foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the STC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis.</p> <p>From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).</p>
STC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	STC basket to be marked and identified in accordance with drawing 6015 <sup>(a)</sup> . No Code stamping is required. The STC basket data package is to be in conformance with Holtec's QA program.

(a) Holtec International Report HI-2094289

## 5.0 PROGRAMS

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The following programs shall be established, implemented and maintained.

### 5.1 Transport Evaluation Program

- a. For lifting of the loaded STC or loaded HI TRAC using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.
- b. This program is not applicable when the loaded HI TRAC is in the fuel building or is being handled by equipment providing support from underneath (e.g., on air pads).
- c. The loaded HI TRAC may be lifted to any height necessary during TRANSFER OPERATIONS provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.
  1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.
  2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.
  3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.
- d. The lift height of the loaded HI TRAC above the transport route surface or other supporting surface shall be limited to 6 inches, except as provided in Specification 5.1.e.

### 5.2 Metamic Coupon Sampling Program

A coupon surveillance program shall be implemented to maintain surveillance of the Metamic neutron absorber material under the radiation, chemical, and thermal environment of the STC.

The surveillance program will be implemented to monitor the performance of Metamic by installing a minimum of four bare coupons near the maximum gamma flux elevation (mid height) at no less than four circumferential downcomer areas around the STC fuel basket. At any time during its use the STC must have a minimum of one coupon installed in each quadrant. Metamic coupons used for testing must have been installed during the entire fuel loading history of the STC.

The following specifications apply:

- (i) Coupon size will be nominally 4" x 6". Each coupon will be marked with a unique identification number.

(continued)

5.0 PROGRAMS (continued)

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- (ii) ~~Pre-characterization testing: Before installation, each coupon will be measured and weighed. The measurements shall be taken at locations pre-specified in the test program. Each coupon shall be tested by neutron attenuation before installation in the STC. The weight, length, width, thickness, and results of the neutron attenuation testing shall be documented and retained.~~
- (iii) ~~Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained.~~
- (iv) ~~The coupons shall be examined for any indication of swelling, delamination, edge degradation, or general corrosion. The results of the examination shall be documented and retained.~~
- (v) ~~The coupons shall be tested by neutron attenuation and the results compared with the pre-characterization testing data. The results of the testing shall be documented and retained. Results are acceptable if the measured value is within +/- 2.5% of the value measured for the same coupon at manufacturing.~~
- (vi) ~~The coupons shall be returned to their locations in the STC unless anomalous material behavior is found. If the results indicate anomalous material behavior, evaluation and corrective actions shall be pursued.~~

5.3 Technical Specifications (TS) Bases Control Program

~~This program provides a means for processing changes to the Bases of these Technical Specifications.~~

- a. ~~Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.~~
- b. ~~Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:~~
  - 1. ~~a change in the TS incorporated in the license; or~~
  - 2. ~~a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.~~
- c. ~~The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.~~
- d. ~~Proposed changes that do not meet the criteria of Specification 5.3.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).~~

(continued)

5.0 PROGRAMS (continued)

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5.4 Radiation Protection Program

- 5.4.1 ~~The radiation protection program shall appropriately address STC loading and unloading conditions, including transfer of the loaded TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.~~
- 5.4.2 ~~Total (neutron plus gamma) measured dose rates shall not exceed the following:~~
- a. ~~1400 mrem/hr on the top of the STC (with lid in place).~~
  - b. ~~5 mrem/hr on the side of the TRANSFER CASK~~
- 5.4.3 ~~The STC and TRANSFER CASK surface neutron and gamma dose rates shall be measured as described in Section 5.4.6 for comparison against the limits established in Section 5.4.2.~~
- 5.4.4 ~~If the measured surface dose rates exceed the limits established in Section 5.4.2, then:~~
- a. ~~Administratively verify that the correct contents were loaded in the correct fuel basket cell locations.~~
  - b. ~~Perform a written evaluation to determine whether TRANSFER OPERATIONS can proceed without exceeding the dose limits of 10 CFR 72.104 or 10 CFR 20.1301.~~
- 5.4.5 ~~If the verification and evaluation performed pursuant to Section 5.4.4 show that the fuel is loaded correctly and the dose rates from the STC and TRANSFER CASK will not cause the dose limits of 10 CFR 72.104 or 10 CFR 20.1301 to be exceeded, TRANSFER OPERATIONS may occur. Otherwise, TRANSFER OPERATIONS shall not occur until appropriate corrective action is taken to ensure the dose limits are not exceeded.~~
- 5.4.6 ~~STC and TRANSFER CASK surface dose rates shall be measured at approximately the following locations:~~
- a. ~~The dose rate measurement shall be taken at the approximate center of the STC top lid. Two (2) additional measurements shall be taken on the STC lid approximately 180 degrees apart and 12 to 18 inches from the center of the lid, avoiding the areas around the inlet and outlet ports. The measurements must be taken when the STC is in the HI TRAC after the steam space is established and prior to HI TRAC lid installation.~~

(continued)

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~~5.0 PROGRAMS (continued)~~

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- ~~b. A minimum of four (4) dose rate measurements shall be taken on the side of the TRANSFER CASK approximately at the cask mid-height plane. The measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.~~

**HDI-IPEC-22-076**

**Enclosure - Attachment 3**

Retyped IP2 and IP3 FLs and Consolidated IP2 and IP3 ISFSI Only TS Pages



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

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC AND

HOLTEC INDIAN POINT 2, LLC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

RENEWED FACILITY LICENSE

Renewed License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) having found that:
  - A. The application for a renewed license filed by Entergy Nuclear Indian Point 2, LLC (ENIP2) and Entergy Nuclear Operations, Inc. (ENO), for Indian Point Nuclear Generating Unit No. 2 at the Indian Point Energy Center (IPEC) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
  - B. Deleted per Amendment No. 294;
  - C. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;
  - D. There is reasonable assurance: (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;
  - E. HDI is technically and financially qualified and Holtec Indian Point 2, LLC (Holtec IP2) is financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;
  - F. Holtec IP2 and HDI have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;
  - G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

- H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed Facility License No. DPR-26, subject to the conditions for the protection of the environment set forth herein, is in accordance with 10 CFR Part 51, Appendix B, of the Commission's regulations and all applicable requirements of said Appendix B have been satisfied;
  - I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31; and
2. Renewed Facility License No. DPR-26 is hereby issued to Holtec IP2 and HDI to read as follows:
- A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 2, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by Holtec IP2 and maintained by HDI. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the Defueled Safety Analysis Report, as supplemented and amended, and the Environmental Report, as amended.
  - B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:
    - (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec IP2 to possess and use, and (b) HDI to possess and use, the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;
    - (2) HDI pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended.

- (3) HDI pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use, at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for reactor instrumentation and are used in the calibration of radiation monitoring equipment, and as fission detectors in amounts as required;
  - (4) HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;
  - (5) HDI pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Deleted per Amendment No. 294.
  - (2) Technical Specifications  
The Technical Specifications contained in Appendices A and B, as revised through Amendment No. [XXX], are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.
  - (3) Deleted per Amendment No. 294.
- D. (1) Deleted per Amdt. 82, 12-11-82.  
(2) Deleted per Amendment 238.
- E. Deleted per Amdt. 71, dated 8-5-81, effective 5-14-81.
- F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act amendments of 1972.
- G. Deleted per Amendment [###]

H. HDI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans<sup>1</sup> for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The HDI CSP was approved by License Amendment No. 266, as supplemented by changes approved by License Amendment Nos. 279, 284, and 286.

HDI has been granted Commission authorization to use "stand alone preemption authority" under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. HDI shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

3. Deleted

(a) Deleted

(b) Provisional Trust:

(i) The provisional trust agreement must be in a form acceptable to the NRC.

(ii) Investments in the securities or other obligations of Holtec International or its affiliates, subsidiaries, successors, or assigns are and shall be prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are and shall be prohibited.

<sup>1</sup> The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

- (iii) The provisional trust agreement must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The provisional trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.
  - (iv) The provisional trust agreement must provide that the agreement cannot be amended in any material respect, or terminated, without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.
  - (v) The appropriate section of the provisional trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.
  - (vi) Use of assets in the provisional trust, in the first instance, shall be limited to the expenses related to decommissioning IP2 or IP1 as defined by the NRC in its regulations and issuances, and as provided in this license and any amendments thereto.
- (c) Deleted
- 4. Deleted
  - 5. Deleted
  - 6. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

*/RA/*

Ho K. Nieh, Director  
Office of Nuclear Reactor Regulation

Attachments:

- Appendix A – ISFSI Only Technical Specifications
- Appendix B – Environmental Technical Specification Requirements

Date of Issuance: September 17, 2018



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

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC AND

HOLTEC INDIAN POINT 3, LLC

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

RENEWED FACILITY LICENSE

Renewed License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for a renewed license filed by Entergy Nuclear Indian Point 3, LLC (ENIP3) and Entergy Nuclear Operations, Inc. (ENO) for Indian Point Nuclear Generating Unit No. 3 (IP3 at the Indian Point Energy Center (IPEC) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
  - B. The facility will be maintained in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. Holtec Indian Point 3, LLC (Holtec IP3) and HDI are financially and technically qualified to engage in the activities authorized by this amendment;
  - E. Holtec IP3 and HDI have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;
  - F. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;
  - G. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31;

- H. The issuance of this renewed license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and
2. Accordingly, Renewed Facility License No. DPR-64 is hereby issued to Holtec IP3 and HDI to read as follows:
- A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by Holtec IP3 and maintained by HDI. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Defueled Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended.
  - B. Subject to the conditions and requirements incorporated herein, the Commission licenses:
    - (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec IP3 to possess and use, and (b) HDI to possess and use the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;
    - (2) HDI pursuant to the Act and 10 CFR Part 70, to possess, at any time, special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;
    - (3) HDI pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;

- (4) HDI pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus or components;
  - (5) HDI pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:
- (1) Deleted per Amendment No. 270
  - (2) Technical Specifications
- The Technical Specifications contained in Appendices A and B, as revised through Amendment No. [XXX], are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.
- D. (DELETED)
- E. (DELETED)
- F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.
- G. HDI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and to the authority of 10 CFR 50.90 and CFR 50.54(p). The combined set of plans<sup>1</sup> for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0," and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

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<sup>1</sup> The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The HDI CSP was approved by License Amendment No. 243, as supplemented by changes approved by License Amendment Nos. 254, 260, and 263.

HDI has been granted Commission authorization to use “stand alone preemption authority” under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. HDI shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

3. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Ho K. Nieh, Director  
Office of Nuclear Reactor Regulation

Attachments:

Appendix A – ISFSI Only Technical Specifications

Appendix B - Environmental Technical Specification Requirements

Date of Issuance: September 17, 2018

**APPENDIX A**  
**TO**  
**RENEWED FACILITY LICENSES DPR-26 and DPR-64**

**FOR**

**HOLTEC INDIAN POINT 2, LLC**  
**HOLTEC INDIAN POINT 3, LLC**  
**HOLTEC DECOMMISSIONING INTERNATIONAL, INC.**  
**INDIAN POINT NUCLEAR GENERATING PLANT UNIT NOS. 2 AND 3**  
**DOCKET NOS. 50-247 and 50-287**

**ISFSI ONLY TECHNICAL SPECIFICATIONS**

**IP2 Amendment No. ###**  
**IP3 Amendment No. ###**

**Renewed Facility License Nos. DPR-26 and DPR-64  
Appendix A – ISFSI Only Technical Specifications**

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4.2	Deleted
4.3	Fuel Storage
5.0	<b>ADMINISTRATIVE CONTROLS</b>
5.7	High Radiation Area

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## 4.0 DESIGN FEATURES

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### 4.1 Site Location

IP2 and IP3 are located on the East bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York. The site is approximately 24 miles north of the New York City boundary line. The nearest city is Peekskill which is 2.5 miles northeast of Indian Point.

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### 4.2 Deleted

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### 4.3 Fuel Storage

Spent fuel shall not be stored in the IP2 and IP3 Spent Fuel Pits.

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

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As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

#### 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
- b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following facility radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
  1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
  2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
  3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

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5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
  - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
  - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
  1. All such door and gate keys shall be maintained under the administrative control of the lead licensee representative on shift, radiation protection manager, or his or her designee.
  2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

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5.7 High Radiation Area

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5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following facility radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
  - 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
  - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
  - 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
    - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
    - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

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5.7 High Radiation Area

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5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.
- f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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Enclosure - Attachment 4

Regulatory Commitments

This table identifies actions discussed in this letter for which HDI commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

Commitment	Type (Check One)		Scheduled Completion Date (If Required)
	One-Time Action	Continuing Compliance	
Administrative controls from IP2 PDTs 5.1.1, 5.2.1, 5.3.1, 5.4.1 (excluding 5.4.1.(a)), 5.5.1 (as modified in LAR), 5.5.3 (as modified in LAR), 5.5.10 (as modified in LAR), and 5.6 (as modified in LAR) will be relocated to the IP2 DSAR	X		On implementation of the approved amendment
Administrative controls from IP3 PDTs 5.1.1, 5.2.1, 5.3.1, 5.4.1 (excluding 5.4.1.(a)), 5.5.1 (as modified in LAR), 5.5.4 (as modified in LAR), 5.5.11 (as modified in LAR), and 5.6 (as modified in LAR) will be relocated to the IP2 DSAR	X		On implementation of the approved amendment