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

December 22, 2017

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 – ISSUANCE OF AMENDMENTS RE: AMENDMENT OF INTER-UNIT TRANSFER OF SPENT FUEL (CAC NOS. MF8991 AND MF8992; EPID L-2016-LLA-0039)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 287 to Facility Operating License No. DPR-26 for Indian Point Nuclear Generating Unit No. 2 (IP2) and Amendment No. 264 to Facility Operating License No. DPR-64 for Indian Point Nuclear Generating Unit No. 3 (IP3). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 14, 2016, as supplemented by letters dated April 19, 2017; August 16, 2017; and October 2, 2017.

The amendments revise the Appendix A TS Limiting Condition for Operation (LCO) 3.7.13, "Spent Fuel Pit Storage," for IP2, and Appendix C TS LCO 3.1.2, "Shielded Transfer Canister (STC) Loading," for IP2 and IP3, and increase the population of IP3 fuel assemblies eligible for transfer via the STC to the IP2 spent fuel pool.

A copy of our related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. Amendment No. 287 to DPR-26
2. Amendment No. 264 to DPR-64
3. Safety Evaluation

cc w/Enclosures: Distribution via Listserv

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

ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 287
License No. DPR-26

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated December 14, 2016, as supplemented by letters dated April 19, 2017; August 16, 2017; and October 2, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate 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 amendment 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. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 287, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: December 22, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 287

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following page of the license with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3	3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.7.13-1	3.7.13-1

Replace the following pages of the Appendix C Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.1.2-1	3.1.2-1
3.1.2-5	3.1.2-5
3.1.2-6	3.1.2-6
---	3.1.2-7

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

(4) ENO 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. Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 220 09-06-01

C. This amended 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) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal. Amdt. 241 10-27-04

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 287, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details an Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pit Storage

LCO 3.7.13

IP2 fuel assemblies stored in the Spent Fuel Pit shall be classified in accordance with Figure 3.7.13-1, Figure 3.7.13-2, Figure 3.7.13-3, and Figure 3.7.13-4, based on initial enrichment, burnup, cooling time and number of Integral Fuel Burnable Absorbers (IFBA) rods; and,

Fuel assembly storage location within the Spent Fuel Pit shall be restricted to Regions identified in Figure 3.7.13-5 as follows:

a. Fuel assemblies that satisfy requirements of Figure 3.7.13-1 may be stored in any location in Region 2-1, Region 2-2, Region 1-2 or Region 1-1;

b. Fuel assemblies that satisfy requirements of Figure 3.7.13-2 may be stored in any location in Region 2-2, Region 1-2 or Region 1-1;

c. Fuel assemblies that satisfy requirements of Figure 3.7.13-3 may be stored in any location in Region 1-2, Region 1-1, or in locations designated as "peripheral" cells in Region 2-2; and

d. Fuel assemblies that satisfy requirements of Figure 3.7.13-4 may be stored:

   1) In any location in Region 1-2, or

   2) In a checkerboard loading configuration (1 out of every two cells with every other cell vacant) in Region 1-1; or

   3) In locations designated as "peripheral" cells in Region 2-2.

IP3 fuel assemblies shall be stored in Region 1-2 of the Spent Fuel Pit. Only assemblies with initial enrichment ≤ 4.4 w/o U^{235} and discharged prior to IP3 Cycle 12 shall be stored in the Spent Fuel Pit. IP3 fuel assemblies V43 and V48 are not approved for storage in the Spent Fuel Pit.

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

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 ≤ 1.2 kW (any cell);
3. Total STC Decay heat from all cell locations including NON FUEL HARDWARE ≤ 9.621 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.

- 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 ≤ 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 ≤ 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.

Table 3.1.2-2

NON FUEL HARDWARE[(a)] Post Irradiation Cooling Times and Allowable Average Burnup

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs[(b, d)]	TPDs[(b)(c)]	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 BPRA 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[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
1	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 25 years Initial Enrichment ≥ 2.3 wt% U-235
2	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
3	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
4	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.6 wt% U-235	Burnup ≤40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
5	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
6	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 2.3 wt% U-235

Table 3.1.2-3 (Sheet 2 of 2)

Allowable STC Loading Configurations

Configuration[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
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.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

ENTERGY NUCLEAR INDIAN POINT 3, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 264
License No. DPR-64

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated December 14, 2016, as supplemented by letters dated April 19, 2017; August 16, 2017; and October 2, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate 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 amendment 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. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 264, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: December 22, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 264

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following page of the license with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix C Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.1.2-1	3.1.2-1
3.1.2-5	3.1.2-5
3.1.2-6	3.1.2-6
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(4) ENO 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. — Amdt. 203 11/27/00

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. — Amdt. 203 11/27/00

C. This amended 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) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 264, are hereby incorporated in the License. ENO shall operate the facility in accordance with the Technical Specifications.

(3) (DELETED) — Amdt. 205 2-27-01

(4) (DELETED) — Amdt. 205 2-27-01

D. (DELETED) — Amdt. 46 2-16-83

E. (DELETED) — Amdt. 37 5-14-81

F. This amended license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter of May 2, 1975, to Consolidated Edison Company of New York, Inc., granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

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 ≤ 1.2 kW (any cell);
3. Total STC Decay heat from all cell locations including NON FUEL HARDWARE ≤9.621 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.

- 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 ≤ 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 ≤ 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.

Table 3.1.2-2

NON FUEL HARDWARE[a] Post Irradiation Cooling Times and Allowable Average Burnup

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs[b, d]	TPDs[b][c]	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 BPRA 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[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
1	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 25 years Initial Enrichment ≥ 2.3 wt% U-235
2	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
3	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
4	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.6 wt% U-235	Burnup ≤40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
5	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
6	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 2.3 wt% U-235

Table 3.1.2-3 (Sheet 2 of 2)

Allowable STC Loading Configurations

Configuration[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 287 TO FACILITY OPERATING LICENSE NO. DPR-26

AND AMENDMENT NO. 264 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

1.0 INTRODUCTION

By letter dated December 14, 2016 (Reference 1), as supplemented by letters dated April 19, 2017 (Reference 2); August 16, 2017 (Reference 3); and October 2, 2017 (Reference 4), Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a license amendment request (LAR) for changes to Indian Point Nuclear Generating (Indian Point) Unit Nos. 2 and 3 (IP2 and IP3) Technical Specifications (TSs).

The proposed changes would revise the Appendix A TS Limiting Condition for Operation (LCO) 3.7.13, "Spent Fuel Pit Storage," for IP2, and Appendix C TS LCO 3.1.2, "Shielded Transfer Canister (STC) Loading," for IP2 and IP3. These LCOs ensure that the fuel to be loaded into the STC meets the design basis for the STC and has an acceptable rack location in the IP2 spent fuel pool (SFP) before the STC is loaded with fuel. The proposed changes would increase the population of IP3 fuel assemblies that would be eligible for transfer via the STC to the IP2 SFP. Note that, at Indian Point, the SFP may also be referred to as the spent fuel pit.

The supplemental letters dated August 16, 2017, and October 2, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 19, 2017 (82 FR 27885).

2.0 BACKGROUND

By letter dated July 13, 2012 (Reference 5 or hereafter referred as the "2012 license amendments"), the NRC issued Amendment No. 268 to Facility Operating License No. DPR-26 for IP2 and Amendment No. 246 to Facility Operating License No. DPR-64 for IP3. These amendments authorized the inter-unit transfer of spent fuel from the IP3 SFP to the IP2 SFP. Fuel assemblies to be transferred are selected at IP3 based on restrictions for loading in the STC and for storage in the IP2 SFP. These restrictions on discharge cycle, initial enrichment, burnup, and cooling time provide a significant limitation on the population of assemblies that can be transferred to IP2 and subsequently be loaded into the HI-STORM dry storage system.

Specifically, the amendments modified the IP2 TS 3.7.13, "Spent Fuel Pit Storage," of Appendix A to the IP2 Operating License to restrict storage of IP3 fuel to a specific region of the IP2 spent fuel pit; the IP3 TS 3.7.15, "Spent Fuel Pit Boron Concentration," of Appendix A to the IP3 Operating License to require a specific spent fuel pit boron concentration during inter-unit fuel transfer; and added Appendix C, "Inter-Unit Fuel Transfer Technical Specifications," to the Operating Licenses for both IP2 and IP3.

Appendix C to the IP2 and IP3 Operating Licenses included the following description of the inter-unit fuel transfer system:

> 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.
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> 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.
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> 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 ¾ inch steel, 2 ¾ inch lead and ¾ 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.
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> 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 ¾ inch steel, 2-7/8 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.

The twelve fuel storage locations in the STC consist of four inner cells and eight peripheral cells. The four inner cells are arranged in a square at the center of the STC, and these cells are designated as Cells 1 through 4 in IP2 and IP3 Appendix C, TS LCO 3.1.2, Figure 3.1.2-1. The peripheral cells consist of two cells adjacent to each side of the square formed by the four inner cells. The peripheral cells are designated as Cells 5 through 12 in IP2 and IP3 Appendix C, TS LCO 3.1.2, Figure 3.1.2-1. The substantial internal STC area unoccupied by cells allows for circulation of coolant.

3.0 REGULATORY EVALUATION

NRC regulations in Title 10 of the *Code of Federal Regulations* Section 50.90, "Application for amendment of license, construction permit, or early site permit," specifies requirements for amendments to nuclear power plant operating licenses and states that applications for amendments shall follow, as far as applicable, the form prescribed for original applications. In licensing the STC, the staff referenced 10 CFR 50.34, "Contents of applications; technical information," which states that the application shall include the principal design criteria for the facility and the relationship of the design bases of the new structures, systems, and components (SSCs) to the design criteria. Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants. The following general design criteria (GDC) are applicable to the design of a spent fuel transfer system:

- GDC 1, "Quality standards and records," specifies, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2, "Design bases for protection against natural phenomena," specifies, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4, "Environmental and dynamic effects design bases," specifies, in part, that SSCs important to safety shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures, and from events and conditions outside the nuclear power unit.

- GDC 60, "Control of releases of radioactive materials to the environment," specifies, in part, that the nuclear power unit design shall include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

- GDC 61, "Fuel storage and handling and radioactivity control, " specifies, in part, that fuel storage and handling systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety; (2) with suitable shielding for radiation protection; (3) with appropriate containment, confinement, and filtering systems; (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

- GDC 62, "Prevention of criticality in fuel storage and handling," specifies that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

- GDC 63, "Monitoring fuel and waste storage," specifies that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

The NRC staff identified the following regulatory requirements and guidance documents as being applicable in the review of the proposed amendments:

- 10 CFR 50.36, "Technical specifications," specifies that each operating license issued by the Commission will contain TSs that are derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34.

- 10 CFR 50.68, "Criticality accident requirements," states in part, that if the licensee does not credit soluble boron in its criticality analysis, the k-effective (k_{eff}) of the SFP storage racks must not exceed 0.95 at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. The k_{eff} is defined as the effective neutron multiplication factor.

- Paragraph 50.68(b)(1) of 10 CFR Part 50 requires:

  > Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

- Paragraph 50.68(b)(4) of 10 CFR Part 50 requires:

  > If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if

flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," provides guidance regarding the specific acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68 and GDC 62.

- NUREG-0800, Section 9.1.2, "New and Spent Fuel Storage," provides guidance regarding the specific acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68.

NUREG-0800, Section 15.0, "Introduction - Transient and Accident Analyses," provides guidance regarding the spectrum of transient and accident conditions considered in the safety analysis and the associated acceptance criteria. This guidance specifies that the plant transients and accidents selected for analysis should represent a broad spectrum of transients and accidents. The guidance also specifies that if the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, the design of the facility's SSCs should be such that all of the postulated transients and accidents produce about the same level of risk (i.e., no single event should be a risk outlier).

A typical application for a new facility such as a power reactor starts with a preliminary safety analysis report, which is later superseded by a final safety analysis report. Although IP2 and IP3 each have an Updated Final Safety Analysis Report (UFSAR), there is no particular section in the UFSARs for fuel cask designs. The NRC staff has accepted the licensee's proposal to rely on a cask licensing report, HI-2094289, "Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at the Indian Point Energy Center," Revision 9 (Reference 6), as the equivalent of the UFSAR for the newly-designed STC and the transfer of spent fuel from the IP3 SFP to the IP2 SFP. The licensee will reference this document, as updated, in the next revision of the UFSARs for IP2 and IP3. In this safety evaluation (SE), Reference 6 will be referred to as the safety analysis report (SAR).

4.0 TECHNICAL EVALUATION

4.1 Description of Proposed Changes

The licensee provided the following summary of changes to the TSs in Section 2.0, "Proposed Changes," of Attachment 1 to the LAR:

IP2 Appendix A LCO 3.7.13, "Spent Fuel Pit Storage"

Delete restrictions related to fuel enrichment and discharge cycle for IP3 fuel stored in the IP2 spent fuel pit.[1]

[1] In its letter dated April 19, 2017, the licensee revised the LAR, stating that the lower enrichment limit restriction in the proposed TSs would be removed (for further detail, see Section 4.2 (Criticality Safety Evaluation) of this SE).

IP2 and IP3 Appendix C LCO 3.1.2, "Shielded Transfer Canister (STC) Loading"

LCO 3.1.2 a. applies when the STC is loaded only with intact fuel assemblies of lower reactivity (i.e., Type 2 fuel that satisfies the minimum burnup specified for the initial enrichment and operation condition applicable to the assembly specified in Appendix C TS Table 3.1.2-1), and allows all cells to be loaded. The proposed change to LCO 3.1.2 a. increases the allowable decay heat in any cell to 1.2 kilowatts (kW) from 1,105 watts in an inner cell and from 650 watts in a peripheral cell. The proposed change also adds a limit on total STC decay heat from all cell locations, including non-fuel hardware (NFH) of less than or equal to 9.621 kW.

LCO 3.1.2 b. applies when the STC is loaded with one or more intact fuel assemblies of higher reactivity (i.e., Type 1 fuel that does not satisfy the minimum burnup specified for the initial enrichment and operation condition applicable to the assembly specified in Appendix C, TS Table 3.1.2-1), and allows loading of only the peripheral cells. An existing note requires that the four inner cells be empty with a cell blocker installed that prevents insertion of fuel or NFH if one or more Type 1 assemblies are loaded in the STC. The proposed change to LCO 3.1.2 b. increases the allowable decay heat in a peripheral cell to 1.2 kW from 650 watts. Although not explicitly listed in LCO 3.1.2.b, the maximum allowed STC decay heat with the eight peripheral cells at a decay heat of 1.2kW per cell would be 9.6kW.

The proposed change also deletes LCO 3.1.2.c and modifies Appendix C, TS Table 3.1.2-3, "Allowable ST loading Configurations." The LCO 3.1.2.c proposed for deletion specifies that only intact fuel assemblies with initial enrichment between 3.2 and 4.4 weight percent (w/%) U-235 and discharged prior to IP3 Cycle 12 may be placed in the STC basket. The deletion of LCO 3.1.2.c and the proposed modification to Table 3.1.2-3 increase STC loading flexibility by expanding the range of fuel assembly parameters for burnup, cooling time, and initial enrichment that may be loaded in the STC.

4.2 Criticality Safety Evaluation

4.2.1 Background

The current IP2 and IP3 TSs limit the IP3 fuel that can be moved to the IP2 SFP to fuel that has a uranium-235 (U^{235}) w/% enrichment between 4.4 and 3.2 and from operations earlier than IP3 Cycle 12. The restriction on IP3 fuel discharged prior to Cycle 12 carries with it an implicit minimum burnup of the fuel and minimum cooling time. These TS restrictions provided margin that the NRC staff relied upon in the initial licensing of the STC and storage of the IP3 spent nuclear fuel (SNF) in the IP2 SFP. As initially proposed in the LAR dated December 14, 2016, the IP2 and IP3 TSs would have been revised in a way that would have permitted the transfer of fresh unpoisoned fuel with a maximum 5.0 w/% enrichment of U^{235} from IP3 to IP2 SFP. This would have removed margin the NRC staff had relied on in its previous review of the STC without providing sufficient justification or an alternate basis for approval. In response to the NRC staff's letter requesting supplemental information dated April 11, 2017 (Reference 7), Entergy submitted its response dated April 19, 2017 (Reference 2). The April 19, 2017, letter revised the LAR in that only the lower enrichment limit in the current TSs would be removed. Additionally, the April 19, 2017, letter provided information on the remaining limited population of fuel assemblies still residing in the IP3 SFP that would meet the stipulation that they be from operations earlier than IP3 Cycle 12. In the August 16, 2017, letter (Reference 3), the licensee provided the revised TSs to implement the revised scope from the April 19, 2017, letter.

Additionally, the STC nuclear criticality safety analysis was revised to address an issue from the STC's initial licensing action. The STC has three approved loading configurations. Two are versions with a fuel assembly in all twelve locations, provided they meet the loading requirements in the TSs. The other is for fresh fuel in the eight outside locations with the four interior locations empty. In the initial licensing application, the licensee did not calculate the reactivity effect of the manufacturing tolerances for the fresh fuel loading configuration, but rather, the licensee used information from the burned fuel loading configurations. The NRC credited the limitation in enrichment and cycle loading requirements to approve the initial licensing of the STC. In this LAR, the licensee's analysis has been revised to address the NRC's use of the enrichment and cycle loading requirements to cover that aspect.

4.2.2 Regulatory Evaluation

The NRC staff issued an internal memorandum on August 19, 1998 (Reference 9), containing guidance for performing the review of SFP nuclear criticality safety analysis. This memorandum is known colloquially as the "Kopp Memo," after the author (Laurence Kopp). While the Kopp memorandum does not specify a methodology, it does provide some guidance on the more salient aspects of a nuclear criticality safety analysis, including computer code validation. The guidance is germane to both borated and unborated boiling-water reactors and pressurized-water reactors.

On September 29, 2011, the NRC staff issued Interim Staff Guidance (ISG) DSS-ISG-2010-1, Revision 0 (Reference 8). The purpose of the ISG is to provide updated review guidance to the NRC staff to address the increased complexity of recent SFP license applications. The NRC staff used DSS-ISG-2010-1 for the review of the current application. Also, the review was performed consistent with Section 9.1.1 of NUREG-0800.

4.2.3 NRC Staff Technical Evaluation

Storage of IP3 SNF in IP2 SFP

The current proposal will keep the upper enrichment limit and cycle usage restrictions on the fuel being transferred from the IP3 SFP to the IP2 SFP, which limits the population of IP3 fuel that can be transferred. Additionally, the restriction to only store the IP3 fuel in IP2 SFP Region 1-2 will remain.

The current IP2 SFP TS for Region 1-2 allows for storage of fresh unpoisoned fuel up to 4.5 w/% U^{235}. In its April 19, 2017, letter, the licensee affirmed that the upper enrichment limit and cycle usage restrictions on the fuel being transferred from the IP3 SFP to the IP2 SFP provide approximately 16 percent delta-k margin to the IP2 SFP analysis of record for Region 1-2.

IP2 SFP Region 1-2 relies upon the Boraflex neutron absorbing material to maintain subcriticality requirements. Boraflex is known to degrade in an SFP environment. In its April 19, 2017, letter, the licensee affirmed that the condition of the Boraflex in IP2 SFP Region 1-2 continues to be bounded by its current analysis of record.

While the fuel transferred from IP3 is being restricted to storage in IP2 SFP Region 1-2, a fuel misloading event could result in the fuel being placed in one of the other IP2 SFP regions. In its April 19, 2017, letter the licensee affirmed that the fuel authorized to be moved from IP3 to IP2, if misloaded into IP2 SFP Region 2-2, would meet the regulatory requirement for k-effective with approximately 7 percent delta-k margin if no credit for Boraflex was taken, but with full credit for the IP2 SFP TS requirement of 2,000 parts per million of soluble boron.

The licensee did not explicitly affirm the margin for the other two regions in the IP2 SFP, but its estimate of the margin to the regulatory limit can be extended to Regions 1-1 and 2-1. Region 2-1 is physically identical to Region 2-2. The difference in the regions is that Region 2-1 does not credit Boraflex, while Region 2-2 does. Since the licensee's estimation of the margin in Region 2-2 did not include credit for Boraflex, the margin would be the same for Region 2-1. Region 1-1 is a flux trap design where the pitch between the fuel assemblies is larger than that in Regions 2-1 and 2-2. Region 1-1 does not credit Boraflex, but with the flux trap between the fuel assemblies, the margin will be larger than in the Region 2 racks.

Loading IP3 Fuel into the STC

The licensee's analysis of record for the STC, Holtec Report HI-2094289 (the SAR) was revised to address the NRC staff's crediting the limitation in enrichment and cycle loading requirements without explicitly calculating the reactivity effect of the manufacturing tolerances for the fresh fuel loading configuration. Instead of explicitly calculating the reactivity effect of the manufacturing tolerances for the fresh fuel loading configuration, the analysis was revised to take credit for the IP3 TS limitation that prohibits the STC being in the IP3 SFP any time there is fresh fuel in the IP3 SFP. While the restriction against having fresh fuel in the IP3 SFP while loading the STC should ensure that all of the fuel has some burnup, there is no requirement for any particular amount of burnup that goes along with that restriction. So while the restriction would provide margin relative to the 5.0 w/% U^{235} modeled in the analysis, the amount of margin is indeterminate – it could be small or large, which makes it a poor choice when trying to offset a potential nonconservatism in the analysis. However, in the SE addressing the initial licensing of the Indian Point Energy Center (IPEC) STC, the NRC staff cited both the restrictions on the IP3 fuel that can be moved and the approximate 1,000 percent millirho (pcm) of margin in the unborated estimation of k_{eff} of the STC Configuration 2. While the exact impact of explicitly calculating the reactivity effect of the manufacturing tolerances is unknown, it will be significantly less than 1,000 pcm. Therefore, the NRC staff accepts the STC Configuration 2 criticality safety analysis without consideration of the restrictions that prevent loading of fresh fuel into the STC.

A mechanical change that can occur to fuel assemblies as they are irradiated is spacer grid expansion. This type of physical change was not previously included in the analysis of physical changes. As part of its letter requesting supplemental information, the NRC staff asked the licensee to consider the effect of fuel assembly spacer grid expansion. In its April 19, 2017, letter, the licensee estimated that fuel assembly spacer grid expansion should be no more than 1,100 pcm. Additionally, in the April 19, 2017, letter, the licensee referred to Section 4.7.9 of the SAR that estimated the margin in the analysis due to a conservative modeling of the actinides and fission products in the IP3 SNF. A less conservative modeling of the actinides and fission products, but still consistent with practices for modeling SNF in SFP criticality analysis, would yield sufficient margin to counter-balance the reactivity increase associated with fuel assembly spacer grid expansion. Therefore, the NRC staff considers the fuel assembly spacer grid expansion to be adequately covered in the SAR loading SNF into the STC.

The NRC staff used the licensee's estimated margin with the restrictions on the IP3 fuel that can be moved to the IP2 SFP. Based on those estimations, there is sufficient margin to accommodate the reactivity increase associated with fuel assembly spacer grid expansion. Therefore, the NRC staff considers the fuel assembly spacer grid expansion to be adequately covered in these amendments.

4.2.4 NRC Staff Finding

The licensee submitted its LAR to increase the IP3 fuel inventory that could be moved via that STC to the IP2 SFP. In response to the NRC staff's letter requesting supplemental information, the licensee revised the LAR to leave in place some of the restrictions but still increase the amount of IP3 SNF that can be moved to the IP2 SFP. With those restrictions in place and the information provided by the licensee, the NRC staff finds reasonable assurance that the licensee will comply with the applicable regulatory requirements. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

4.3 Evaluation of Neutron Absorbing Material

4.3.1 Background

This specific portion of the NRC staff's review is limited to the Metamic neutron absorbing material (NAM) in the STC, and the condition of the Boraflex NAM in the IP2 SFP where the fuel from the STC will be placed.

In its LAR dated December 14, 2016, the licensee did not propose any changes to TS 5.2, "Metamic Coupon Sampling Program." Section 8.5.3.5 of the non-proprietary version of the SAR also did not include any changes to the Metamic Coupon Sampling Program that were previously reviewed and approved by the NRC in Reference 5. The NRC staff confirmed that the Metamic Coupon Sampling Program has not been changed since the previous NRC approval. Therefore, the staff has determined that the Metamic Coupon Sampling Program remains acceptable and will continue to identify potential degradation of the material that could impair its neutron attenuation capability.

In the supplemental letter dated April 19, 2017, the licensee discussed its analysis that supports the transfer of spent fuel from IP3 to IP2. The licensee also stated that this analysis is based on the current IP2 criticality analysis of record. In this submittal, the licensee stated that 50 percent of the original Boron-10 (B-10) areal density for the Boraflex NAM is credited for criticality control in the IP2 criticality analysis of record. The licensee stated that preliminary test data from the April 2017 BADGER testing of Boraflex in Region 1-2 of the IP2 SFP indicated that the Boraflex degradation is still bounded by the IP2 SFP criticality analysis of record. By supplemental letter dated August 16, 2017, the licensee provided additional information as further support for finding reasonable assurance that the condition of the Boraflex NAM is still adequate for the fuel being moved via the STC.

4.3.2 NRC Staff Technical Evaluation

The NRC staff reviewed the supplement dated August 16, 2017, to determine if the condition of the Boraflex NAM is still adequate, given past degradation, and whether the Boraflex NAM will continue to be appropriately monitored so that the material remains in adequate condition. The

licensee provided comparisons of the 2017 BADGER test data and the assumptions used in the IP2 SFP criticality analysis of record for local degradation, average degradation, and gap size for the Boraflex NAM. In addition, the licensee provided a comparison between the 2017 BADGER results and RACKLIFE predictions. The NRC staff finds reasonable assurance that the 2017 BADGER test data shows the condition of the Boraflex NAM is adequate for the fuel being moved by the STC into Region 1-2 of the IP2 SFP. In addition, the staff finds reasonable assurance that the condition of the Boraflex NAM in Region 1-2 of the IP2 SFP will continue to be adequate due to the current Boraflex monitoring program, as well as the RACKLIFE predictions.

4.3.3 NRC Staff Finding

The staff finds that the coupon surveillance program for the Metamic NAM used in the STC remains acceptable, given that it has not changed since the prior NRC review and approval. In addition, the staff finds reasonable assurance that the condition of the Boraflex NAM used in Region 1-2 of the IPEC IP2 SFP is still adequate for the fuel being moved via the STC. The staff also finds that the monitoring program, in conjunction with BADGER testing and RACKLIFE predictions will continue to provide reasonable assurance that the condition of the Boraflex NAM will remain adequate. Based on the acceptability of the Boraflex and Metamic monitoring programs, and the condition of the Boraflex NAM, the staff finds that the licensee meets the applicable portions of 10 CFR 50.68, GDC 61, and GDC 62 with respect to the NAMs used in the STC and for the fuel being moved into Region 1-2 of the IPEC IP2 SFP.

4.4 Thermal-Hydraulic Evaluation

4.4.1 Thermal Safety

The SAR stated that the 2004 Edition of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) would govern the fabrication, testing, and inspection of the STC. The SAR stated that the material procurement, design, fabrication, and inspection of the STC basket would be performed in accordance with ASME Section III, Subsection NG (2004 Edition). The SAR also stated that the pressure boundary of the STC meets the stress limits of the ASME Code, Section III, Class 3, Subsection ND, with large margins, but the STC is not a code stamped vessel. The SAR listed the applicable design pressure and temperature for the STC, in Table 3.2.1 of the SAR. According to the ASME Code, Section ND-7000, pressure vessels are required to have overpressure protection; however, no overpressure protection is provided in the STC. The function of the STC is to retain the radioactive contents under normal, off-normal, and accident conditions. The STC is designed to withstand a maximum internal pressure considering maximum accident temperatures. To ensure the retention of radioactive contents, the SAR determined that a pressure relief valve would not be necessary once the proper initial conditions had been established within the STC and verified by test.

As specified by GDC 61, fuel storage and handling systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety; (2) with suitable shielding for radiation protection; (3) with appropriate containment, confinement, and filtering systems; (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and (5) to prevent significant reduction in fuel storage coolant inventory under accident

conditions. The design of the fuel transfer system during transport consists of a sealed STC inside a sealed HI-TRAC. The STC contains a heat source (i.e., the stored fuel) and a heat transfer medium (i.e., water in thermal equilibrium with a steam bubble). The licensee included the provisions to establish the steam bubble in the design of the STC to more effectively control the pressure increase that could result from reduced cooling of the contained fuel during accident conditions. Within the HI-TRAC pressure boundary, the STC rejects heat to the annular water volume within the HI-TRAC, and an air space mitigates the pressure rise that results from the thermal expansion of the STC and the annular water volume.

For the fuel transfer system, the containment, heat removal, and retention of coolant inventory functions are interrelated. As stated above, the STC pressure boundary and the HI-TRAC pressure boundary provide robust containment barriers for internal pressures up to their respective design pressures. The initial conditions established within each pressure boundary (i.e., the heat generation rate of the fuel loaded in the STC, the volume of water, and the volume of the steam (STC) or air (HI-TRAC) bubble), and the heat removal rate determine whether the internal pressure of the STC and HI-TRAC remains below the respective component design pressure. Based on the pressure within the STC and HI-TRAC remaining within the respective design pressures under postulated accident conditions that affect the heat removal rate, the staff finds reasonable assurance that the design is adequate to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

To address the performance of the fuel transfer system, the licensee evaluated several postulated accident scenarios. The licensee provided the SAR in non-proprietary and proprietary versions as Enclosure 2 and Enclosure 3, respectively (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16355A068 and ML16355A069), to the LAR. The SAR presented analyses of the following conditions that could challenge the adequacy of the fuel transfer system heat removal mechanisms:

- Normal onsite transfer of IP3 fuel
- Loss of HI-TRAC water jacket
- External fire (rupture of transporter gas tank)
- Loss of HI-TRAC annulus water
- Fuel misloading
- Hypothetical tipover accident
- Crane malfunction

The licensee presented the results of these thermal analyses in Section 5 of the SAR. The initial NRC licensing of the fuel transfer process was based on evaluation of Revision 5 to the Holtec International Licensing Report HI-2094289 (the SAR). The licensee indicated that the following changes to the thermal analyses provided in Section 5 of the SAR were implemented through Revision 8:

- Revision 7 revised Table 5.0.1, "Bounding Shielded Transfer Cask Thermal Load," to provide more flexibility in loading configurations.
- Revision 7 added Section 5.3.5 to support the revised allowable heat load distributions presented in Table 5.0.1.
- Revision 8 updated Section 5.3.5, "Sensitivity Study of Heat Load Distribution," to provide justification for the heat loads adopted in thermal evaluations.

- Revision 8 modified Section 5.4.4, "Fuel Misload Accident," to clarify heat load distribution adopted under fuel misload scenario.
- Revision 8 updated Section 5.4.5, "Hypothetical Tipover Accident," to incorporate the evaluations performed for the bounding loading scenario. The SAR included revised Table 5.4.9, "Hypothetical Tipover Accident Temperatures," and Table 5.4.10, "Hypothetical Tipover Accident Pressures," reflecting the analysis results for the bounding heat load scenario outlined in Section 5.4.5.
- Revisions 7 and 8 included editorial changes and clarifications throughout Chapter 5.

4.4.1.1 NRC Staff Technical Evaluation

The staff reviewed the changes to the analyses described through Revisions 7 and 8 to the SAR and evaluated the changes relative to the proposed TS changes. The change to Table 5.0.1 reflected the proposed changes to IP2 and IP3 Appendix C TS 3.1.2 by adopting a maximum allowable decay heat per storage location value of 1.2 kW and a maximum allowable STC decay heat value of 9.621 kW. The two heat load distributions considered for sensitivity evaluations in Section 5.3.5 reflect the extremes of heat distributions allowed by proposed IP2 and IP3 Appendix C TSs 3.1.2.a and 3.1.2.b. Specifically, one heat distribution scenario considered the four inner locations at the maximum allowed decay heat value and the remaining locations adjusted to comply with the total STC decay heat value of 9.621 kW (e.g., the peripheral locations contained fuel with average decay heat values of about 600 watts), which is consistent with proposed Appendix C TS 3.1.2.a. The second heat distribution scenario considered that all peripheral locations contained fuel with the maximum allowed decay heat value and the four inner locations were empty (e.g., total STC decay heat value of 9.6 kW), which is consistent with proposed Appendix C TS 3.1.2.b. In Section 5.4.4, the SAR was clarified to reflect that the fuel misload accident was evaluated for a hypothetical misload event defined as a loading where location is loaded with fuel generating twice the heat load used for the normal loading configuration. Section 5.3.3 stated that the distribution used for the normal loading configuration was 1105.2 watts decay heat in the four interior cells and 650 watts in the eight peripheral cells (e.g., a total STC decay heat loading of 9.62 kW). The update to Section 5.4.5 stated that the decay heat distribution in the hypothetical tipover accident used decay heat values of 1.2 kW for the two locations partially uncovered when the STC rests in a horizontal orientation and the other locations close to the water surface to maximize the surface water temperature. This heat distribution provides a bounding evaluation of STC temperature and pressure because the maximum STC pressure equals the saturation pressure for the maximum water surface temperature. Therefore, the staff concluded that the values proposed as operating limits in IP2 and IP3 Appendix C TS 3.1.2 were consistent with the values used in the supporting analyses provided in Revision 8 of the SAR.

The staff evaluated the sensitivity of the STC temperature and pressure under accident conditions to the change in heat distribution permitted by the proposed IP2 and IP3 Appendix C TS 3.1.2. The sensitivity analysis provided in Section 5.3.5 of the SAR indicated that the STC internal pressure was insensitive to changes in heat distribution with the STC in a vertical orientation. As stated above, the scenarios evaluated represented the limiting decay heat configurations permitted under the proposed revision to Appendix C TS 3.1.2. The total STC decay heat loading remained the same in these scenarios as in the normal loading configuration evaluated for licensing of the fuel transfer system. The analysis determined that the resulting steady-state STC temperature and pressure for these two scenarios would be essentially the same as the normal loading configuration. This is an expected result because the total STC

decay heat generation rate is the same and the STC in its design configuration provides for effective heat transfer through submergence of the fuel in water and the large open channels for downward flow outside of the fuel locations. Therefore, although the distribution of fuel is important in other contexts such as criticality and shielding, the staff found the thermal response of the STC to be dependent on the total decay heat when in a vertical orientation, rather than the distribution of the decay heat.

The staff reviewed the change to Section 3.4.4 of Revision 8 to the SAR addressing STC misloading events. The staff verified that the change clarified the decay heat loading used in the analysis, but did not change the analysis inputs or results. Therefore, the change to the description of the STC misloading event analysis is acceptable.

The staff also reviewed the Revision 8 update to Section 5.4.5, "Hypothetical Tipover Accident." The proposed change to Appendix C TS 3.1.2 would allow for higher decay heat fuel assemblies to be located in positions that may become uncovered in the event of a hypothetical tipover accident. The analysis results presented in Tables 5.4.9, "Hypothetical Tipover Accident Temperatures," and 5.4.10, "Hypothetical Tipover Accident Pressures," indicate a substantial increase in the STC temperatures and cavity pressure, and the calculated STC cavity pressure exceeded the STC design accident pressure of 90 psig used for original licensing of the transfer system, which was presented in Table 3.2.1, "Internal Pressure and Temperatures," in Revision 5 of the SAR. The change in decay heat distribution during the hypothetical tipover accident resulted in an increase in the calculated STC internal pressure from 83.6 psig in Revision 5 of the SAR to 148.9 psig in Revision 8 to the SAR. To accommodate the higher STC accident pressure, the SAR presented a reevaluation of the STC structure for a design accident pressure of 165 psig. The NRC staff's evaluation of the revised STC design accident pressure is presented in Section 4.5 (Structural Evaluation) of this SE; as stated there, the staff found the increased STC design accident pressure acceptable.

Safety functions of the STC include providing a cooling water retention boundary and providing for adequate heat rejection. These safety functions satisfy the following relevant criteria of GDC 61: a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal and prevention of a significant reduction in fuel storage coolant inventory under accident conditions. Since the structural integrity of the STC under accident conditions assures retention of the cooling water and provides for adequate heat removal to prevent further temperature increases, the maintenance of peak STC internal pressure below the STC accident design pressure demonstrates that these relevant criteria of GDC 61 are satisfied. Therefore, the thermal design of the STC is acceptable for the proposed decay heat distribution.

4.4.1.2 NRC Staff Finding

The licensee proposed TS 3.1.2 to ensure that an acceptable inventory of fuel and other activated material would be loaded into the STC to maintain heat generation rates within analyzed values. The analyses of the various accident conditions determined that heat generation rates consistent with the prescribed limits (i.e., no more than 1.2 kW per cell for the four interior cells and a total STC heat generation rate of no more than 9.621 kW) would ensure that STC design pressure would remain below design pressure for all analyzed conditions. Therefore, consistent with the requirements of 10 CFR 50.36(b), the proposed TS limits on fuel loading are acceptable with respect to heat generation rates.

4.4.2 Thermal-Hydraulic Evaluation

The NRC staff reviewed Chapter 5, "Thermal-Hydraulic Evaluation," of Holtec report HI-2094289 (the SAR) and the associated thermal calculation report HI-2084146, "Thermal Hydraulic Analysis of IP3 Shielded Transfer Cask" to determine cask design compliance with the thermal-hydraulic regulatory requirements of 10 CFR Part 71 and 10 CFR Part 50. The staff also audited some of the supporting thermal analysis files to verify that the analyses were consistent with the cask design and thermal-hydraulic results provided in the SAR.

To ensure fuel integrity, the thermal design utilizes the temperature limits established in NRC SFST-ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," and in the Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System to ensure cask integrity and vessel pressure limits. The design is also governed by the pressure limits stated in the SAR. The thermal criteria are set forth in Chapter 3 of the SAR, Table 3.1.1, "Temperature Limits Applicable to Inter-Unit Transfer," and Table 3.2.1, "Internal Pressure and Temperature." The maximum permissible heat load is specified in SAR Table 5.0.1.

The following proposed changes to IP2 and IP3 TSs are applicable to the thermal evaluation:

- Revise IP2 Appendix C, TS LCO 3.1.2, "Shielded Transfer Canister (STC) Loading"

  LCO 3.1.2.a.2
  Decay heat including NON FUEL HARDWARE ≤ 1.2 kW (any cell)

  LCO 3.1.2.a.3
  Total STC decay heat from all cell locations including NON FUEL HARDWARE ≤ 9.621 kW

  LCO 3.1.2.b.2
  Decay heat including NON FUEL HARDWARE ≤ 1.2 kW

- Revise IP3 Appendix C, TS LCO 3.1.2, "Shielded Transfer Canister (STC) Loading"

  LCO 3.1.2.a.2
  Decay heat including NON FUEL HARDWARE ≤ 1.2 kW (any cell)

  LCO 3.1.2.a.3
  Total STC decay heat from all cell locations including NON FUEL HARDWARE ≤ 9.621 kW

  LCO 3.1.2.b.2
  Decay heat including NON FUEL HARDWARE ≤ 1.2 kW

The NRC staff previously reviewed in the 2012 license amendments (Reference 5), the thermal design description of the STC and concluded that the description is sufficient to make a determination of the adequacy of the STC thermal design. Therefore, an evaluation of the description of the thermal design is not necessary, and the previous evaluation remains valid for these amendments. In addition, the NRC staff previously reviewed the thermal properties used for the package analyses and determined that they were appropriate for the materials specified.

Therefore, an evaluation of the thermal properties is not necessary, and the previous evaluation remains valid for these amendments. Likewise, the NRC staff previously reviewed the thermal model used for the package analyses and determined that it was adequate for the thermal design. Therefore, an evaluation of the thermal model is not necessary, and the previous evaluation remains valid for these amendments.

4.4.2.1 Thermal Evaluation of Fuel Transfer Operation

Thermal analysis of the STC was performed by the licensee under the bounding heat load scenario defined in SAR Table 5.0.1 wherein all fuel assemblies are assumed to be generating heat at the maximum permissible rate.

4.4.2.2 Maximum Temperatures and Pressures

The fuel transfer scenario assumes maximum permissible fuel heat load, hot ambient temperature (per SAR Table 5.0.1), insolation heating, and steady-state maximum temperatures. The licensee performed a grid sensitivity study to determine the impact on the predicted maximum temperatures. From the mesh sensitivity study, the licensee decided to use mesh 1 (described in the SAR) for this analysis since it predicts maximum temperatures and pressures. The results of the analysis are tabulated in SAR Tables 5.3.1 and 5.3.2. All temperatures are within the design-basis limits provided in SAR Table 3.1.1, and pressures are within the design-basis limits provided in SAR Table 3.2.1.

The staff reviewed the calculated maximum temperatures during typical fuel transfer conditions and with the bare STC (i.e., the STC outside of the HI-TRAC transfer cask) using the design-basis decay heat and determined that the predicted temperatures are below the allowable material limit. The staff also reviewed the licensee's approach for detecting the misload of fuel assemblies with decay heat greater than permitted and found it acceptable.

4.4.2.3 Thermal Evaluation During Accident Conditions

The licensee performed the evaluation of several postulated accidents during transfer operations to demonstrate that the STC will remain within design-basis conditions. The licensee determined that the proposed TS changes have a minor impact in the cask thermal performance during the postulated accidents evaluated in the SAR.

The staff reviewed the licensee's analysis models and assumptions used to evaluate the STC and the HI-TRAC during accident conditions. The staff audited the analysis files and determined that the models were prepared correctly based on the information provided in the SAR in terms of geometry and material properties. The staff also verified that the selected models and assumptions were adequate for the analyzed conditions. The staff reviewed the calculated maximum temperatures and pressures during accident conditions and determined the predicted temperatures and pressures are all below the allowable limits.

4.4.2.4 Review of Thermal Models

The NRC staff reviewed the licensee's thermal models used in the analyses. The staff examined the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The staff further verified that the licensee's

selected code models and assumptions were adequate for the flow and heat transfer characteristics that exist due to the STC and HI-TRAC geometry and analyzed conditions. The engineering drawings were also examined to verify that adequate geometry dimensions were translated to the analysis models. In addition, the material properties presented in the SAR were reviewed to verify that they were appropriately referenced and used. The staff further verified that the licensee performed appropriate sensitivity analysis calculations to obtain mesh-independent results that would provide bounding predictions for all analyzed conditions during normal fuel transfer and accident conditions. The staff also independently performed a number of sensitivity calculations to determine that the results provided in the SAR have been obtained with adequately converged analysis. Finally, through a request for additional information (RAI), the staff received additional information needed to make a safety determination on the adequacy of the fuel transfer thermal design. Based on its review, the staff finds the IPEC STC thermal-hydraulic analysis and conclusions are acceptable, and provide reasonable assurance that the IPEC STC will continue to safely transfer spent nuclear fuel within the IP2 and IP3 STC TS parameters.

4.4.2.5 NRC Staff Finding

Under the proposed amendments, the NRC staff finds that the thermal-hydraulic design of the IP STC continues to remain in compliance with the regulatory requirements of 10 CFR Part 72 and 10 CFR Part 50 and that the applicable design and acceptance criteria as identified in the IP SAR would be satisfied. The evaluation of the thermal-hydraulic design provides reasonable assurance that under the proposed amendments, the cask will continue to allow safe transfer of spent fuel.

4.5 Structural Evaluation

4.5.1 Background

The licensee's initial license amendment request for approval of the STC, dated July 8, 2009 (Reference 10), was a first-of-a-kind application, as it was the first fuel cask that a licensee had sought to be licensed by the NRC under 10 CFR Part 50. The NRC staff evaluated the application using the regulations stated in 10 CFR Part 50; however, the requirements in 10 CFR Part 72 and related regulatory guidance documents (for example, NUREG-1536) were used to inform the staff's review because of the direct relevance to specific SSCs for different portions of the LAR.

The proposed changes to the IP2 and IP3 TSs in the LAR dated December 14, 2016, increase the heat load in the STC. As a result of the thermal analysis of the increased heat load, the licensee increased the design-basis accident pressure from 90 pounds per square inch gauge (psig) to 165 psig.

The objective of the NRC staff's structural review is to verify that the structural performance of the package continues to meet the requirements of 10 CFR Part 50. The structural design of the STC in Revision 8 of the SAR is identical to that of the previously approved revision. As such, the staff limited its review to the load combinations that were affected by the increased internal pressure. Additionally, Chapter 6 of the SAR identifies and describes the structural components of the STC. The objective of the staff's review is to ensure that the STC and the

HI-TRAC transfer cask are capable of withstanding the design, normal, and accident conditions of transporting spent fuel from the IP3 SFP to the IP2 SFP.

4.5.2 Structural Design and Structural Evaluation Details

The HI-TRAC transfer cask is an existing piece of equipment that is part of the HI-STORM 100 Dry Cask Storage System, certified by the NRC under Docket No. 72-1014, and is currently in use at Indian Point. The licensing drawings for the STC and HI-TRAC transfer cask are listed in Section 1.5 of the SAR. The STC is designed to meet ASME Code, Section III, Division 1, Subsection ND, stress limits. The STC is the defined confinement boundary for the spent fuel, as indicated in Table 6.1.6, "List of ASME Code Alternatives for STC," of the SAR. Spent fuel canisters (i.e., the confinement boundary) are normally constructed to ASME Code Subsection NB or NC (reference NUREG-1536, Section 3.4.1). However, in the 2012 license amendments (Reference 5), the NRC staff determined that ASME Code, Section III, Division 1, Subsection ND, is adequate and applicable based on the licensee's code reconciliation between Subsections ND and NC and post-fabrication examination and acceptance testing. As previously mentioned, the HI-TRAC 100 transfer cask, which was certified by the NRC under Docket No. 72-1014, has already been in use for IP2 dry cask storage operations. It is also noted that the HI-TRAC 100D, with the new bolted top lid, will be subject to an internal pressure when it is used to transport the STC. The HI-STORM FSAR does not consider any internal pressure loads on the HI-TRAC 100D since the lid used for dry storage contains a large circular hole in the middle. Therefore, the newly designed top lid, as well as the HI-TRAC inner shell and bottom lid, must be evaluated for the effects of internal pressure. For this purpose, the stress limits of ASME Section III, Subsection ND (pressure vessel code), are used.

Load Case 1: Design Pressure

The STC normal pressure limit is 50 psig, while the design accident pressure limit is 165 psig. The licensee's analysis for the design pressure-induced stresses determined that an appropriate safety factor (against service level A and D conditions) exists for normal and accident conditions. The stresses in the STC baseplate due to the design internal pressure are bounded by the combined effects of internal pressure plus normal handling. A fatigue analysis was not conducted because using Appendix I of the ASME Code, the number of cycles of loading and unloading is conservatively less than 500 loading cycles which translates to transfer of 6,000 fuel assemblies at 12 assemblies in each transfer evolution). For purposes of the fatigue margin assessment, the number of cycles is assumed to be 1,000 (i.e., N = 1000), and the corresponding stress amplitude due to fatigue would not be sufficient to approach conditions for a fatigue failure.

Load Case 2: Normal Operating Pressure plus Temperature

There are no significant thermal stresses in an STC enclosure vessel since the presence of water both inside and outside of the STC minimizes the thermal gradients across the pressure boundary. The design temperature limits of the materials that comprise the STC and HI-TRAC are tabulated in Table 3.1.1 of the SAR. The load case of normal operating pressure plus the effect of temperature[2] is determined to be minimal, and no significant thermal stresses will be generated.

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[2] The STC maximum temperature limits are identified in Chapter 3, Table 3.1.1 of the SAR, which are within the design-basis limits.

Load Case 3: Normal Handling

In Reference 1, the licensee stated that the HI-TRAC 100D lifting trunnions and the surrounding structure are analyzed in Section 3.4.3 of the HI-STORM 100 FSAR (K.A) for a bounding lifted weight of 200,000 pounds (lbs) (as compared to a total weight of 190,000 lbs for the HI-TRAC 100D, including a fully loaded STC). All applicable lift points on the STC and HI-TRAC have been designed to satisfy the requirements of NUREG-0612 and ANSI N14.6.[3] Prior to initial use, the STC will be tested to 300 percent of the maximum design lifting load of 80,000 lbs to demonstrate adequate design and fabrication of the lifting attachment and trunnions per Section 8.4.3 of the SAR. The STC and HI-TRAC lifting devices, including trunnions, will be recertified annually in accordance with ANSI N14.6, per Section 8.5.3.3 of the SAR.

The licensee analyzed the STC lifting trunnions to show they have an adequate safety factor (the safety factor is greater than 6.0) for lifting. The analysis included an appropriate dynamic load factor increase per Crane Manufacturers Association of America specifications.

The licensee analyzed the STC lid, which is part of the load path, by using ANSYS Workbench with an appropriate dynamic load factor. Details of this analysis are provided in Appendix E, "Analysis of STC Lid during Lifting," of Holtec Report HI-2084118 (Reference 11, proprietary report). The licensee provided the detailed STC trunnion and STC closure lid lifting analysis in Appendix A, "Shielded Transfer Canister Structural Calculation Package"; Appendix B, "STC Baseplate and Closure Lid Stress Analysis"; and Appendix C, "Structural Analysis," of Holtec Report HI-2084118 for pool bottom lid and top lid lifting (plus internal pressure).

During the review of the LAR, the NRC staff verified that the analyses show that all stresses that are generated during the normal handling of the STC and HI-TRAC transfer cask have acceptable safety margins with conservatism (using dynamic load factors and following appropriate codes and standards).

Load Case 4: Fuel Assembly Drop

A fuel assembly drop accident analysis was performed by the licensee in order to determine the effect of a 2,000 lb fuel assembly dropping 36 inches onto the STC fuel basket during handling operations in the SFP. The acceptance criterion is to ensure that after the drop accident, the fuel storage array (with the damaged basket) will remain in a subcritical configuration. The analysis performed includes conservatism and is appropriate to determine that the damage is limited to the top of the basket and that the fuel will remain in a subcritical configuration. Details of the analysis are provided in Appendix D of HI-2084118.

Load Case 5: HI-TRAC Vertical Drop Accident

A vertical drop accident analysis was performed by the licensee to ensure that a loaded STC inside the HI-TRAC transfer cask will survive a 6-inch drop when the HI-TRAC is being lifted by

[3] NUREG-0612 is the applicable guidance with specific limits for interfacing lift points in the STC and the HI-TRAC transfer cask; however, stress limits from ANSI N14.6 were conservatively applied (see Section 6.1.2.2 of the SAR).

the vertical cask transporter (VCT).[4] The analysis was done in LS-DYNA. The licensee provided details of the analysis in HI-2094345, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation" (proprietary report). The acceptance criteria provided were determined to be acceptable. The analysis results conclude that the loaded STC and HI-TRAC transfer cask would survive the 6-inch vertical handling accident drop without impairment of their safety functions.

The NRC staff noted that per the Indian Point licenses, Appendix C, Section 5.1.d, the HI-TRAC transfer cask (with a loaded STC inside) is not permitted to be lifted above 6 inches unless certain conditions have been met (e.g., the VCT in use at Indian Point has a lifting device designed in accordance with ANSI N14.6 and has redundant drop protection features once the load is pinned). Therefore, the 6-inch drop is a bounding drop height.

Load Case 6 and 7: Seismic Stability of Loaded VCT and Loaded HI-TRAC

The licensee analyzed a VCT loaded with a HI-TRAC transfer cask for a seismic event equal to the Indian Point site design-basis earthquake (DBE). The approach used to analyze the systems is consistent with what was done in the HI-STORM 100D FSAR to demonstrate stability of the freestanding HI-STORM. Based on the inputs that were used, the analysis concludes that a loaded VCT and a loaded HI-TRAC are not susceptible to tipping during a DBE and will remain stable during operations. The staff notes that a tipping analysis (Load Case 9) was also done and it conservatively concluded that in the event of tipping, a loaded STC and HI-TRAC transfer cask will remain below acceptable stress/deceleration limits.

Load Case 8: Seismic Stability of the STC in the Fuel Pool

In Section 6.2.7 of the SAR, the licensee analyzed the STC sitting on the floor of the SFP with no crane attached for a seismic event equal to the Indian Point DBE. The IP2 and IP3 SFPs are each one large pool with one corner designed for placing the spent fuel transfer cask. In this case, the concern is that the STC could tip or slide into adjacent fuel storage racks during a seismic event, possibly damaging the spent fuel assemblies stored in those racks or damaging any spent fuel assemblies that have been loaded into the STC. The licensee used the criteria for static equilibrium to show that the STC will not slide or tip in the SFP during the DBE.

During the course of its review, the NRC staff noted that the seismic stability analyses of the freestanding equipment for all applicable fuel transfer evolutions have been carried out using the appropriate seismic excitation for the supporting surface on which the equipment is staged. The Zero Period Acceleration of the DBE for IP3 at the ground elevation (54.5 feet above mean sea level) is listed in Table 3.2.2 of the SAR. Inasmuch as the SFP, the FSB truck-bay, and the travel path adjacent to the truck-bay are all located near ground level, the same earthquake is applicable. Therefore, the NRC staff finds this analysis of the STC is acceptable.

[4] The VCT is a high-capacity, tracked vehicle designed specifically for the lifting and handling of spent fuel storage casks. The VCT lifts the HI-TRAC via special lifting devices designed, constructed, and tested in accordance with ANSI N14.6. The HI-TRAC is lifted by its trunnions using hydraulic lifting towers, which are an integral part of the VCT and which have features to prevent a load drop, even under complete hydraulic line failure.

Load Case 9: Non-Mechanistic Tip-over of a Loaded HI-TRAC Cask

The licensee performed a tipover analysis to ensure that a loaded STC inside the HI-TRAC transfer cask will survive a tipover event when travelling from IP3 to IP2. The analysis was done in LS-DYNA. The acceptance criteria (proprietary) were provided and determined to be acceptable. The analysis performed was conservative because of the neglected effect of the water (hydrodynamic damping), and the STC was modeled as a single rigid body, thus providing no energy absorbing capability during the event. The analysis is detailed in Holtec Report HI-2104706, "Non-Mechanistic Tip-over of HI-TRAC 100D" (proprietary report). The analysis concluded that fuel integrity was fulfilled, continued leaktightness of the STC was maintained, the HI-TRAC lid gasket remained compressed and the lid bolt stress did not exceed design limits, there was no loss of shielding due to tipping, and the stresses/decelerations induced in the HI-TRAC transfer cask and STC were below the acceptable limits. Therefore, the non-mechanistic tipover event will not cause the failure of any of the safety features of the HI-TRAC transfer cask and the STC.

Load Cases 1 and 9

As a result of the increased accident pressure (Table 3.2.1), the licensee revised the two governing load case evaluations from Table 3.2.4 of the SAR that included the accident pressure and presented the results in Chapter 6 of the SAR. The design internal pressure (Load Case 1) and the postulated tipover accident of the loaded HI-TRAC transfer cask (Load Case 9) were reevaluated with the higher internal accident pressure in the STC.

For Load Case 1, the licensee computed the bending stress in the lid, the tensile stress in the bolts, and the shear stress in the bolt threads, and compared the result to the allowable stress from the ASME Code, Section III, Division 1, Subsection ND. These results were presented in Section 6.2.1.1 of the SAR. For all three components, the maximum calculated stress was less than the allowable stress, indicating that the components have adequate strength to resist the applied loads.

For Load Case 9, the licensee evaluated the sealing of the STC at the top closure lid joint due to the tipover accident with the increased internal pressure. In Section 6.2.8 of the SAR, the licensee stated that the STC closure joint seals remain compressed, and the stresses in the STC closure bolts remain below the stress limits of ASME Code, Section III, Division 1, Subsection ND, subsequent to the tipover event.

The staff reviewed Section 6 of the SAR, as well as the affected calculations in Holtec proprietary reports HI-2084118 and HI-2104706. In Load Case 1, the calculated stresses in the lid, the bolts, and the bolt threads increased, but are still well below the allowable stresses stipulated in ASME Code, Section III, Division 1, Subsection ND. Because the calculated stresses are less than the allowable stresses, the staff concludes that the structural performance of the STC is adequate for Load Case 1.

In Load Case 9, subsequent to the tipover event, a positive compressive force remains between the STC lid and flange. Based on the configuration of the bolted joint, the gasket remains adequately compressed. Additionally, the tensile stress in the bolts is less than the ASME allowable stress. Because of the above, the staff concludes that the sealing state of the bolted

lid on the STC will not be impaired, and the structural performance of the STC is adequate for Load Case 9.

Section 10.3.1, "Haul Path Inspection and Control"

Section 10.3.1, "Haul Path Inspection and Control," includes ten steps of operating procedures that will be verified by the Region I inspector at the time of VCT movement of the HI-TRAC transfer cask with the STC loaded with fuel assemblies. Therefore, there is no scope of review required by the NRC staff regarding this section.

4.5.3 NRC Staff Finding

In summary, the staff finds that the structural analyses for the STC and the HI-TRAC transfer cask systems for the nine different load cases were appropriate, and the results of those analyses are acceptable. Based on the licensee's structural evaluation and the staff's review of the statements and methodologies employed in the application, the staff concludes that the structural aspects of the proposed design have been adequately described and meet the regulatory requirements of 10 CFR Part 50.

4.6 Shielding and Radiological Evaluation

4.6.1 Shielding and Protection

4.6.1.1 Background

The NRC originally approved wet spent fuel transfer operations at IPEC for IP2 and IP3 in 2012 for a set of six spent fuel loading patterns in an STC (Reference 5). These loading patterns, together with other limits such as decay heat and limits on specifications for non-fuel hardware (NFH), were incorporated into Appendix C, Part II, of each unit's license TSs. These patterns and limits defined the specifications for the spent fuel that could be transferred from the IP3 SFP to the IP2 SFP.

On December 14, 2016, the licensee submitted an application requesting amendments to the licensing conditions to add additional loading patterns and revised fuel qualification parameters for the spent fuel transfer operations. The objective of the shielding and radiation protection review is to verify that the STC design with the proposed new loading patterns and fuel characteristics continues to provide adequate protection to the public and workers against radiation from the operations to transfer the spent fuel, including NFH.

The licensee requests authorization to add six new loading patterns with spent fuel specifications that will increase the population of the spent fuel that may be transferred from the IP3 SFP to the IP2 SFP using the STC. The licensee also requests authorization to revise the specifications of the burnable poison rod assemblies (BPRAs) that may be transferred with the spent fuel in the six new loading patterns.

The staff reviewed the licensee's application to evaluate whether regulatory requirements will be met during normal operating, off-normal (e.g., crane hangup and vertical cask transporter breakdown), and design-basis accident conditions for the proposed STC loading patterns and contents. Since the confinement design remains unchanged, so that there will be no effluent or

radiological release from the STC operations, the staff's review and evaluation only addresses doses from direct radiation. Also, because the shielding design and operation procedures remain the same as previously approved, this review focuses on the radiation source definition for proposed contents and the impacts on the dose rate and dose analyses for compliance with occupational worker and public dose limits and as low as is reasonably achievable (ALARA) requirements. Note that the term "members of the public" used in this evaluation also includes the people at the plant site who are not occupational radiation workers.

The licensee's evaluation results are compared to the requirements and limits in 10 CFR Part 72, as well as 10 CFR Part 20. Given the nature of the LAR, the staff followed the guidance provided in NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility – Final Report," and NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," as appropriate.

4.6.1.2 Radiation Source Definition

Proposed Contents Changes

The proposed six new spent fuel loading patterns include high burnup fuel (i.e., fuel irradiated beyond 45 GWd/MTU. The proposed loading patterns also include the same NFH types that are allowed in the currently approved loading patterns as defined in Table 3.1.2-3 of Appendix C, Part II, of the TSs for the STC. However, the maximum irradiation and minimum cooling time of the BPRAs is changed to be the same as the burnup and cooling time of the fuel assembly with which it is transferred. In addition, the new loading patterns continue to require that assemblies with Inconel grid spacers be loaded only in the inner region of the STC fuel basket. The assemblies in the new load patterns must also meet the specifications in Appendix C, Part II, Table 4.1.1-1, of the TSs, which remain the same with these amendments.

Changes in Cobalt Impurity Levels in Assembly Hardware and Burnable Poison Rod Assembly Steel Components

For evaluating the radiation source terms, the licensee used the same code, method, and assumptions as were used previously, with a few changes. These changes included reduction of the cobalt impurity assumed in the steel of the assembly hardware and the BPRAs. The licensee reduced the cobalt in the assembly hardware components from 1.2 grams to 0.5 grams per kilogram of steel and reduced the cobalt in the BPRA steel components from 1.2 grams to 0.8 grams per kilogram of steel, respectively. The staff evaluated the licensee's justifications for using the lower cobalt impurity values. In addition, the staff also considered the burnup and cooling time limits for the fuel assemblies and the BPRAs to be loaded in the new loading patterns in the STC. The 1.2 gram/kilogram (g/kg) value was previously used because the burnup and cooling time limits for the currently approved six load patterns indicated that assemblies in some of the load patterns and some BPRAs would have been manufactured in the late 1980s or earlier. Prior to the late 1980s, cobalt levels in steel were much higher than the current fuel designs. In the late 1980s, a significant effort was put forth to reduce the cobalt levels in steels used for assembly components. For the proposed new loading patterns, consideration of the burnup and cooling time limits, given the timing of the application, indicates that assemblies and BRPAs at the limits for maximum burnup and minimum cooling time would have been manufactured at a time later than the 1980s. Thus, the cobalt levels in the steel components should be significantly reduced. Based on these considerations, the staff finds use

of the reduced cobalt levels in the assembly and BPRA steel components to be acceptable and consistent with values the staff has seen reported for components fabricated since the late 1980s.

The licensee performed analyses for the additional cooling times to support the proposed change to the BPRA burnup and cooling time limits (i.e., to be the same as the assembly with which the BPRAs are transferred). Since all assemblies in the six new load patterns are burned to greater than 40 GWd/MTU, the licensee calculated the sources in terms of curies of cobalt in the BPRAs at the cooling times as specified in Table 7.2.9 of the SAR for a burnup of 60 GWd/MTU. The minimum analyzed cooling time is 9 years. For items with shorter cooling times, which applies only to the inner assemblies of Load Pattern 11, the analysis used the design-basis curie content of cobalt that was used in the original analysis. The staff notes that even for the actual assembly burnup for the inner assemblies of Load Pattern 11, the cobalt activity of the BPRAs at that burnup will exceed the maximum cobalt activity used in the analyses for BPRAs. Information provided by the licensee in a clarification call (Reference 12) indicates that, for the activity that the burnup and cooling time would yield, the dose rate impact would be no more than 4 percent. The staff notes that even with this increase in dose rates, Load Pattern 11 dose rates are still significantly bounded by the dose rates from Load Patterns 4 and 8. Based on this information, the staff finds that applying the same burnup and cooling time to the BPRAs in the inner region of Load Pattern 11 is acceptable. The licensee modified Table 3.1.2-2 in Appendix C, Part II, of the TSs to include these limits for BPRAs in the new load patterns. The staff reviewed the information regarding this analysis and finds it is consistent with the proposed TS change and that the curie levels are consistent with the reduced cobalt levels assumed for the BPRA steel components.

Reactor Control Component Assemblies Irradiation Configurations

The licensee also included a third irradiation configuration for reactor control component assemblies (RCCAs). The original analyses considered two configurations, a configuration fully withdrawn from the reactor core and a configuration with 10 percent insertion into the core and a corresponding neutron flux distribution under full power operations. This is consistent with activation analyses of RCCAs in the FSAR for the Holtec HI-STORM 100 storage cask system, which Indian Point uses for its spent fuel dry storage operations. The analysis used the appropriate flux factors to determine the source from both the cobalt and the silver-indium-cadmium absorber in the RCCAs. The licensee calculated the source term for a third configuration with 10 percent insertion but with a flux factor that is one fourth of the flux factors previously used for the same RCCA insertion depth. The licensee reasoned that this would cover typical operations with RCCAs.

In response to the NRC staff's RAIs, the licensee further explained that the Indian Point units operate with all control rods fully withdrawn during full power operations and with the majority of them fully withdrawn during all power operations. A small subset may be partially inserted during startup and shutdown (reduced power operations), and these operations are significantly less than 10 percent of the total operating cycle time. The licensee also performed a comparison of dose rates measured on two STC loadings with dose rates calculated for those same loadings, although the calculations neglected both RCCA source terms and their materials, and not just their materials as is done in all of the other calculations in Chapter 7 of the SAR. The comparison of dose rates at the bottom of the STC showed that the calculation over-predicts the bottom dose rates even without including the RCCAs. The staff reviewed the

information regarding RCCA operations presented in the SAR and considered additional operating experience information regarding typical RCCA operations. The staff finds the calculation and use in the shielding analysis of this third configuration is bounding for the Indian Point RCCAs and, therefore, is acceptable. The staff also finds that the dose rate comparison (calculation vs. measurement) for the loaded casks further supports the acceptability of the assumptions and calculated source terms for the third RCCA source term configuration.

High Burnup Fuel

As noted previously, the proposed new spent fuel loading patterns include high burnup fuel. While the currently approved loading patterns also include high burnup fuel, it is limited to only the inner STC basket region for only two loading patterns. High burnup fuel may be loaded in four of the new loading patterns and may be in the inner region or the outer region of the STC basket. In addition, the extent of the burnup is increased to 60 GWd/MTU (assembly average burnup) in one of the proposed loading patterns. As noted in the staff's previous SE (see Section 3.7.2 of Reference 5), because the version of the depletion code the licensee used is validated for only low burnups, future extension of burnups for contents may necessitate evaluation of source term uncertainties associated with high burnup fuel similar to what was done in the HI-STORM 100 FSAR. With the increase in the quantity of and the maximum burnup (in one load case) of high burnup fuel in the new loading patterns, the staff determined that the uncertainties in the source terms should be evaluated.

To address the staff's concerns on the adequacy of the computer code used for calculations of the source terms for high burnup fuel, the licensee discussed the approach taken in the HI-STORM 100 FSAR, which was based on uncertainties in the decay heat calculation. The licensee also pointed out that the calculated decay heat for fuel in the new loading patterns with the proposed burnup, enrichment, and cooling time limits exceeds the STC decay heat limit (defined in Appendix C, Part II, LCO 3.1.2, of the TSs) by greater amounts than the uncertainty value calculated in the HI-STORM 100 FSAR. Thus, the decay heat limit restricts the spent fuel overall such that the decay heat is much less than the calculated decay heat for the STC. The licensee argued that since the STC decay heat limit must also be met, the analysis for an STC completely loaded with spent fuel at the burnup, enrichment, and cooling time limits adequately offsets the uncertainty in the source term calculation for high burnup fuel.

The staff considered the licensee's evaluation of the decay heats for the calculated source terms versus the STC decay heat limit. The staff also reviewed the individual basket cell decay heat limit of 1.2 kW versus the decay heats calculated for each burnup, enrichment, and cooling time combination in the new load patterns as shown in Table 7.1.1 of the SAR. The staff noticed that in terms of an individual assembly, only the fuel with burnup up to 60 GWd/MTU exceeded the individual basket cell limit (by ~6.7 percent). Thus, based on the individual basket cell's decay heat limit, only that fuel would have any margin to compensate for the source term uncertainties. It is true that for a completely loaded STC, the total STC heat limit of 9.621 kW would necessitate that at least some assemblies be loaded at lower decay heats (and thus lower radiation source terms), which is not the case for STCs that only have eight assemblies in the outer region because of criticality requirements. Also, while there is some connection between decay heat and radiation source term, it is not necessarily a simple relationship such that reduction in one by some amount necessarily will result in reduction in the other by the same amount. Therefore, the staff considered how the uncertainty would affect the dose rates for the new load patterns in the STC and the transfer cask (the HI-TRAC). The staff evaluated

this uncertainty along with the other uncertainties and conservatisms that the staff had identified and quantified, as described in the section below titled, "Staff Dose Evaluation." For the STC, the total dose rates (spent fuel plus NFH) for the approved Load Pattern 4 remain bounding; thus, any uncertainty resulting from the calculation of high burnup source terms does not affect the bounding dose rates for the STC. For the HI-TRAC, the new Load Pattern 8 results in bounding total dose rates, at least in some instances. Thus, the staff considered the uncertainty as part of the staff's evaluation of the dose rates and doses for the HI-TRAC.

4.6.1.3 Dose Analyses

Shielding and Source Configuration

The configuration of the shielding is unchanged from the previous amendments that added the wet transfer operations to the IP2 and IP3 licenses. The definition of the proposed new loading patterns includes the source configuration. The changes in these amendments are only to the actual source terms for the spent fuel and NFH that are inserted into the different STC basket cell locations, as defined in the proposed modification to Appendix C, Part II, LCO 3.1.2, and the corresponding tables in the license's TSs. The configuration in terms of locations and the type or number of NFH, and the division of the STC basket into inner and outer regions, is unchanged.

Bounding Dose Rates

Since the currently approved Load Pattern 4 yields bounding dose rates for the STC, the maximum dose rates for the STC remain unchanged from the previous amendments. For the HI-TRAC transfer cask, new Load Pattern 8 results in, at least for some instances, bounding dose rates. Therefore, the licensee had to update the dose rate and dose analyses involving the transfer cask to reflect the new loading patterns. The difference is due to the fact that the neutron contribution is more significant to the transfer cask dose rates than it is in the STC, and the higher burnup limits for the new load patterns result in larger neutron source terms than the currently approved load patterns. Together, these two characteristics result in some of the new load patterns having higher dose rates in the transfer cask than Load Pattern 4, with Load Pattern 8 yielding the bounding dose rates. Accordingly, the licensee updated the SAR evaluations for compliance with the dose limits and ALARA requirements, as necessary, to reflect the changes in bounding dose rates and doses.

Licensee's Dose Analysis

The licensee performed dose analyses to confirm compliance with the dose limits prescribed in 10 CFR Part 20 and 10 CFR Part 72. For some of these limits (e.g., 10 CFR 72.104(a)), contributions from other site operations need to be considered. Thus, the licensee included the dose contributions from other areas of the plants, including waste management and operations, and from the storage of spent fuel at its general license independent spent fuel storage installation (ISFSI). Since the approval of the amendments initially allowing wet transfer operations, the amount of spent fuel in storage at the ISFSI has increased. This resulted in an increased contribution to the cumulative doses, which is reflected in the licensee's new analyses. This latter item is a point for the licensee to be aware of since continued operation of the ISFSI will mean storage of increasing quantities of spent fuel at the ISFSI and an increasing contribution to the doses that must meet regulatory limits. For the ISFSI, this will be evaluated

in accordance with 10 CFR 72.212, "Conditions of general license issued under Section 72.210," in the licensee's evaluation report.

It is important to note that operations for the dry storage of spent fuel and other storage-related facility operations, including the wet transfer of spent fuel from the IP3 SFP to the IP2 SFP, will be controlled by the regulatory dose limits in 10 CFR 72.104(a). Any changes to these operations (e.g., increased number of dry storage casks loaded and stored at the ISFSI pad), will also be controlled by these regulatory limits. Any licensee-initiated changes will require evaluation under 10 CFR 72.48(c) or 10 CFR 50.59 and are subject to NRC inspection.

SAR Tables 7.4.7, 7.4.8, 7.4.14 through 7.4.20, and 7.4.22 summarize the licensee's evaluation of doses for both the public and occupational workers. Table 7.4.22 provides an estimate of occupational doses for a single transfer sequence. The listed tables demonstrate that with the appropriately updated doses for the ISFSI and the transfer cask, the regulatory dose limits relevant to the cases evaluated in the tables will be met. The tables also identify those limits. These evaluations use the same models, computer code, and methods, including assumptions regarding distances, occupancy times, and the number of transfers in a year, as were used in the previous amendments to add wet transfer operations. See the SE (Reference 5) for a discussion of these models, the code, and methods. The evaluations include consideration of off-normal events such as a crane 'hangup' for personnel and public doses and breakdown of the transfer cask transporter. The staff's evaluation of and findings with respect to the licensee's analysis and compliance with limits are discussed in the following section.

NRC Staff's Dose Evaluation

The staff evaluated the public and occupational doses in a similar manner as was done for the previous amendments for the licensee's wet transfer operations of spent fuel. The staff's evaluation included consideration of the various uncertainties, as well as conservatisms that the licensee and staff identified in the licensee's analyses. The staff applied these uncertainties and conservatisms to the estimated doses from both the STC and the transfer cask as appropriate. The staff considered different locations with appropriate occupancy times for evaluation of compliance with the limits in 10 CFR 72.104(a), in addition to the location and occupancy time selected in the licensee's analysis. The staff also considered different combinations of normal conditions with anticipated occurrences (also referred to as off-normal conditions) for the 10 CFR 72.104(a) evaluation. These considerations are described below.

The licensee identified various conservatisms in the analyses. These conservatisms include use of Babcock and Wilcox 15x15 fuel assemblies to determine the fuel's radiation source term (Indian Point uses Westinghouse 15x15 assemblies);[5] use of a single full power operation cycle to achieve the allowable maximum assembly average burnup; use of the Babcock and Wilcox 15x15 assembly axial configuration with the STC in the transfer cask; and, for the new loading patterns, use of the analyzed spent fuel burnup, enrichment and cooling time specifications, resulting in STC heat loads in excess of the allowable limit. These conservatisms, with the exception of the last one for heat loads, were also present in the previous analysis. Based on the information available, the staff performed quantitative estimates for the first two

[5] While the fuel assembly type used at IP3 is Westinghouse 15x15, evaluations have shown that the B&W 15x15 fuel assembly design is bounding when compared to other PWR fuel assembly designs and classes (see Section 7.0 of the SAR).

conservatisms. Since the third conservatism is only applicable to the transfer cask and the last conservatism is based on the discussion of the decay heat limit above, the staff did not account for these two conservatisms, to avoid potentially reducing dose rates and doses.

The licensee also identified various aspects of the analysis method, which the staff considers as introducing uncertainties into the analysis. These include neglect of axial blankets for blanketed fuel; use of the response function, calculation error or uncertainty, uncertainty for rounding up enrichments to the nearest one tenth; and, for high burnup fuel, the uncertainty in determining the source terms with the depletion code the licensee used. Based on the information available, the staff performed quantitative estimates of these uncertainties. The staff also identified from the hardware cobalt levels for all 12 load patterns, some differences in what the SAR described for the approved load patterns versus what the new information indicated was used. The staff also included these differences in the uncertainties evaluated for the analysis.

Based on the available information, the staff, in addition to the controlled area boundary location used by the licensee, identified the locations of the nearest recreational areas, nearest businesses, and the nearest residence. With appropriate residence times for these other locations, the staff estimated the doses from normal and off-normal conditions and determined that the licensee's selection of the Hudson River with a 500 hour per year occupancy was the most limiting case in terms of evaluations against the 10 CFR 72.104(a) and 10 CFR 20.1301(e) limits. These limits are for the combination of normal and off-normal conditions during the year.

Similar to the review done in the 2012 license amendments (Reference 5), the staff evaluated the doses for a crane 'hangup' event at each unit's SFP and for a breakdown of the transfer cask transporter, all occurring in a single year, along with the doses from normal operations for 16 transfers in a year. The staff accounted for the conservatisms and uncertainties identified above. The staff also considered additional time required to complete the transfer operations. The additional time arises due to the inclusion of a 24-hour pressure rise test on the STC that is performed for each transfer. The test is done in the fuel storage building when the STC is in the HI-TRAC. Thus, each transfer lasts more than the 8 hours considered by the licensee. The staff's calculations indicated that the doses may slightly exceed the regulatory limit. However, the staff also considered the additional conservatism in the licensee's analysis that arises from neglect of the structures between the STC and HI-TRAC and the analyzed controlled area boundary (the shore of the Hudson River), which means that the licensee's analyses do not account for the additional shielding those structures provide. Also, with the greater distances between the controlled area boundary and various operations, the dose will also decrease. Finally, the analyses included the contribution from the ISFSI at the shortest distance to the controlled area boundary, which also adds conservatism. Accounting for these conservatisms, the staff finds reasonable assurance that the annual doses from normal conditions and anticipated occurrences will not exceed the regulatory limits for transport operations with the new spent fuel contents.

In its evaluations of doses under accident conditions, the staff included the effects of the uncertainties and conservatisms identified above. The staff also considered the typical occupancy time used in dry storage system analyses of 24 hours a day for the 30 day-duration of the accident. Further, the staff considered doses at a distance from the transfer cask top and bottom in addition to the transfer cask side, which is very conservative. The staff's evaluation indicates that there is significant margin between doses from an accident and the 10 CFR 72.106(b) limit, even with the different assumptions the staff used.

The staff also reviewed and evaluated the licensee's analyses for doses to members of the public who may be on site during the wet transfer operations. The staff notes that the licensee's analysis indicates that the regulatory limits in 10 CFR 20.1301(a) and (b) will be met under normal and off-normal conditions (see Tables 7.4.7, 7.4.8, 7.4.14, and 7.4.15 of the SAR). However, when conditions are combined, as in the staff's analysis of doses for normal and off-normal conditions beyond the controlled area boundary, the staff's evaluation indicated that the annual limit may not be met. However, this is based on a conservative, unrealistic assumption that the same individual is present for each stage of all of the transfers and each off-normal condition (assumed to all occur in the same year) for the entire duration of each of these events. Accordingly, the staff finds reasonable assurance that doses to these persons would not exceed regulatory limits. Furthermore, radiation protection personnel and practices in place would also help to ensure that the limits are not exceeded. Finally, the staff also considered the licensee's estimate of occupational dose for a single transfer sequence. Even accounting for the uncertainties and conservatisms identified above, as well as changes in transfer cask dose rates for the now bounding Load Pattern 8, the staff concludes that its previous findings regarding occupational doses and ALARA (see Section 3.7.4.4 of the staff's SE, Reference 5) continue to be applicable to the transfer operations with the new proposed spent fuel contents. The staff finds the estimated doses expected to be incurred as part of the wet transfer operations to be reasonable and to comply with the requirements of 10 CFR Part 20, including ALARA.

Comparison of Measured Dose Rates vs. Calculated Dose Rates

To further support the proposed amendments, including the acceptability of the changes to the assumptions for cobalt levels in assembly hardware and BPRA steel components, and the source term irradiation of RCCAs, the licensee provided a comparison between the calculated dose rates and the measured dose rates for two loadings of the STC. The dose rates were for two specific loadings and were measured and calculated for the bare STC (i.e., the STC outside of the HI-TRAC transfer cask). The comparison indicates that significant margin remains in the calculated dose rates. The licensee provided additional details for the measurements and details of the calculations in an appendix to the shielding calculation package submitted with the LAR.

The staff evaluated the information regarding the actual measurements and the calculations in the calculation package. In most cases, the staff was able to make findings regarding acceptability of the proposed changes without the need for the comparison, although the licensee's comparison does further support the conclusions of the licensee and the findings of the staff, such as for the changes to the RCCA source term analysis. For these changes, the dose rates on the bottom of the STC are the most useful. Both STC loadings contained RCCAs. It is not clear to the staff based on the information provided by the licensee, however, where on the bottom of the STC the measurement was taken, and the staff's expectation is that it was off-center to some extent. However, the calculations neglected both the RCCA source terms and materials (versus just neglecting the RCCA materials, as is done in all of the other calculations in Chapter 7 of the SAR) and still resulted in dose rates noticeably higher than the measured dose rates. Even adjusting the measured dose rates higher to estimate what the measurement might be if it were centered on the STC bottom, the calculated dose rates still exceed the adjusted measured dose rate.

For the radial dose rates, the staff finds that the information is still less clear. For example, the measurement data do not clearly show where the contents in the STC are relative to the location of the actual mid-height measurement, and there is some uncertainty as to the locations of the measurements with respect to the features of the STC. Further, the data are not sufficient to confirm that the location of the calculated maximum dose rate for STC number 1 is the same as the location of the measured dose rate identified as the maximum. The measured dose rates for both STC loadings are clearly higher than the lowest azimuthal dose rate in the calculations, however, measured dose rates do not appear to match up sufficiently with the different azimuthal peaks in the calculated dose rates. Also, the staff notes that the measured dose rate at 30 cm from the surface does seem to match up better with the calculated peaks at that distance from the STC. Thus, the staff finds the comparison to be only somewhat useful, although it does provide some further indication that the dose rate calculations for the new fuel loads (with the changes to the assumptions identified in the LAR and SAR) are acceptable as a representation of the source term and dose rates for these new fuel loads.

4.6.1.4 NRC Staff Findings

The staff reviewed the shielding and radiation protection evaluation for the wet transfer operations of the proposed new spent fuel loads from the IP3 SFP to the IP2 SFP. The staff reviewed the changes to the analysis methods described in the SAR changes submitted with the LAR. Based on its review of the LAR, the NRC staff finds that:

- The SAR sufficiently describes the proposed new contents and the proposed changes to the analysis.
- The shielding and confinement features continue to be sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 for the proposed new contents.
- The methods described in the SAR for controlling and limiting occupational exposures for the proposed activity to remain within the dose and ALARA requirements of 10 CFR Part 20 continue to be acceptable.

In sum, the NRC staff finds there is reasonable assurance that the wet spent fuel transfer operations with the proposed new content loading patterns will continue to meet the shielding and radiation protection requirements of 10 CFR Part 20, Part 50, and Part 72.

4.6.2 Radiological Safety

4.6.2.1 Background

The proposed changes will expand the population of fuel assemblies that can be transferred from IP3 to IP2. During a fuel transfer, fuel is removed from the IP3 SFP and placed in a specially designed STC. The STC is then placed in a HI-TRAC transfer cask from the HI-STORM 100 dry storage system and transported to the IP2 fuel support building, where the STC is unloaded into the IP2 SFP. The IP3 fuel is then relocated to the IP2 SFP for further transfer to the onsite ISFSI.

This section of the SE provides the NRC staff's evaluation of the radiological safety aspects (i.e., accident consequences and public and occupational radiation safety considerations) of the LAR. The staff performed a limited review of this LAR because the proposed changes were deemed to have minor impacts on the radiological safety of fuel transfer operations, and the licensee did not propose changes to the TSs that specifically apply to the radiation protection aspects of the fuel transfer operation. The NRC staff reviewed the original licensing basis for the fuel transfer operation to verify that the proposed changes do not invalidate the original assumptions of the radiological safety and consequence analyses. The staff also verified the licensee's categorization of the radioactive source term to verify that the proposed changes were adequately modelled and analyzed, and that the applicable TSs continue to be sufficiently protective for the fuel transfer operation. Finally, the NRC staff verified that applicable radiation limits continue to be satisfied and that the licensee would use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve radiation doses that are ALARA.

4.6.2.2 Regulatory Evaluation

As discussed above, the 2012 license amendments (Reference 5) revised the IP2 and IP3 licenses by authorizing the transfer of spent fuel from the IP3 SFP to the IP2 SFP. The NRC staff's evaluation of that LAR considered a combination of regulatory requirements from 10 CFR Part 72 and 10 CFR Part 20. This approach was taken because of the unique circumstances involved with transferring spent fuel from one SFP to another, which renders applicable both sets of regulations. Additionally, the 10 CFR Part 72 requirements were used as acceptance criteria because of the similarity between the IP2/IP3 fuel transfer operation and transfer operations to an ISFSI. Therefore, while the requirements of 10 CFR Part 72 do not specifically apply to the fuel transfer operation via the STC, which is licensed under 10 CFR Part 50, satisfactorily meeting 10 CFR Part 72 was used as an acceptance criterion in the staff's review. The licensee demonstrated compliance with the most restrictive requirements of these regulations where there was more than one limit for the same condition.

The staff applied the same approach in its review as was used for the 2012 license amendments. Additionally, this evaluation relies in part on the understanding that the licensee implements an existing approved radiation protection program pursuant to 10 CFR 20.1101. Specifically, the licensee has implemented a radiation protection program that is commensurate with the scope and extent of its licensed activities and is sufficient to ensure compliance with the provisions of 10 CFR Part 20. Therefore, compliance with certain relevant provisions of 10 CFR Part 20 was not specifically verified by this review because these provisions are required by the current license, were satisfied during prior fuel transfers, and are being overseen and assessed through the NRC's reactor oversight process. For example, how the licensee intends to control access to and post radiologically significant areas has been already verified through other NRC processes, as described above, and thus was excluded from this evaluation.

With regard to all radiation exposures, 10 CFR 20.1101(b) requires, in part, that licensees use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational and pubic exposures that are ALARA.

With regard to doses to individual members of the public, 10 CFR 20.1301 requires, in part, that each licensee shall conduct operations such that the total effective dose equivalent (TEDE)

does not exceed 100 millirem (mrem) (1 millisievert (mSv)) in 1 year and that the dose in any unrestricted area from external sources does not exceed 2 mrem (0.02 mSv) in any 1 hour.

The regulations in 10 CFR 72.104(a) establish radiation limits and other requirements for normal and anticipated occurrences. The annual dose equivalent as a result of planned discharges of radioactive material to the general environment, direct radiation, and any other radiation from fuel cycle operations within the region to any real individual who is located beyond the controlled area must not exceed the following:

- 25 mrem (0.25 mSv) to the whole body;
- 75 mrem (0.75 mSv) to the thyroid; and
- 25 mrem (0.25 mSv) to any other critical organ.

The regulations in 10 CFR 72.104(a) also require that operational restrictions be established to meet ALARA objectives and that operational limits be established to meet the dose equivalent limits listed above. Since, as described later in this evaluation, the canister used to transfer the spent fuel is designed as leaktight, there are no expected planned discharges or "any other radiation from fuel cycle operations"; therefore, the only radiation of concern to occupationally exposed workers and the public is direct radiation.

With regard to radiation limits at or beyond the controlled area boundary (i.e., doses to members of the public), the limits in 10 CFR Part 72 and 10 CFR 20.1301(e) are the most restrictive. The requirements of 10 CFR 20.1301(e) refer to the Environmental Protection Agency's standards in 40 CFR Part 190, which are comparable to the limits in 10 CFR 72.104; therefore, further reference to the limits in 10 CFR 72.104 implicitly includes the requirements of 10 CFR 20.1301(e). However, as it pertains to exposures of members of the public while within the controlled area boundary (i.e., onsite), the limits of 10 CFR 20.1301 are applicable.

With regard to occupational exposures, 10 CFR 20.1201 requires licensees to control occupational doses to individual adults to an annual limit, which is the more limiting of:

- 5 rem (0.05 Sv) total effective dose equivalent;
- 50 rem (0.5 Sv) committed effective dose equivalent to any individual organ or tissue (other than the lens of the eye);
- 15 rem (0.15 Sv) lens dose equivalent; and
- 50 rem (0.5 Sv) shallow dose equivalent to skin of the whole body or of any extremity.

The regulation in 10 CFR 72.106 establishes acceptance criteria for design-basis accident radiological analyses. This regulation states that for each ISFSI or monitored retrievable storage (MRS) site, a controlled area must be established, and any individual located on or beyond the nearest boundary of the controlled area may not receive, from any design-basis accident, the more limiting of:

- 5 rem (0.05 Sv) total effective dose equivalent;
- 50 rem (0.5 Sv) committed effective dose equivalent to any individual organ or tissue (other than the lens of the eye);

- 15 rem (0.15 Sv) lens dose equivalent; and
- 50 rem (0.5 Sv) shallow dose equivalent to skin or any extremity.

Additionally, the minimum distance from the spent fuel, high-level radioactive waste, or reactor-related greater than Class C waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters. These acceptance criteria are generally more restrictive than the traditional criteria used for accident analyses at reactors, set out in in 10 CFR 100.11 and 10 CFR 50.67.

4.6.2.3 NRC Staff Technical Evaluation

The SE associated with Reference 5 provides a detailed evaluation of the shielding design and ALARA considerations for the originally licensed fuel transfer operation. Therefore, this current review focused on how the licensee's proposed changes would impact the assumptions, particularly the source term and shielding performance, of the staff's original analysis and verifies that the existing TS (TS 5.4, "Radiation Protection Program") remain sufficiently protective for the proposed activities and that the applicable radiation limits continue to be satisfied.

The SAR provides the descriptions and results of the licensee's evaluations of the proposed changes. The licensee incorporated Revision 6 of the SAR into its UFSAR following issuance of the 2012 license amendments. It is expected that once this LAR is approved, the licensee will update its UFSAR with Revision 8 of the SAR pursuant to 10 CFR 50.71(e).

Chapter 7 of the SAR describes the licensee's evaluation of the shielding design and ALARA considerations. In general, the licensee performed two sets of calculations – doses at distances from the STC and doses in the vicinity of the STC, to demonstrate compliance with regulatory limits and to assist in ALARA planning, respectively. The staff found this approach acceptable because it is sufficiently conservative and consistent with the previously accepted approach described in the 2012 license amendments.

Radioactive Source Term

The STC loading configurations and associated source terms were selected based on a survey of the current IP3 spent nuclear fuel inventory, which was compared with the bounding design-basis fuel assemblies. The design-basis fuel assemblies were used for HI-TRAC dose rate evaluations, and a modified design-basis fuel assembly was used to obtain realistic dose rates at the bare STC (i.e., the STC outside of a HI-TRAC cask) for ALARA planning.

The principal sources of radiation for the fuel transfer operation are the gamma and neutron radiation typical of spent nuclear fuel and other components typically stored in an SFP (e.g., radiation from the decay of fission products and decay of activated material). The primary source of gamma activity in the non-fuel regions of fuel assemblies is from Co-60, which results when Co-59 is exposed to a neutron flux in an operating reactor. This source term is modeled by assuming a 0.5 g/kg Co-59 impurity level in NFH for assemblies that are cooled less than 20 years or constructed after 1989. For older assemblies, an impurity level of 1.2 g/kg Co-59 is assumed. The licensee derived these values from information provided by its fuel vendor. In the licensee's application for the 2012 license amendment, to simplify calculations, the licensee evaluated assemblies loaded into the periphery of the STC using the 1.2 g/kg Co-59 impurity

value and assemblies loaded into the center of the STC using the 0.5 g/kg Co-59 value. However, the changes currently proposed by the licensee introduce six new loading patterns that result in spent fuel cooling times of 15 years or less; therefore, the new evaluations of periphery and center assemblies apply the 0.5 g/kg Co-59 impurity value.

Other sources of radiation (for example, gamma rays from the decay of material that has been activated by neutron interactions) are modeled through the licensee's application of the Monte Carlo transport code.

The proposed changes to the fuel transfer process result in modifications to inputs used in the radiological analysis of the fuel transfer operation. For example, as noted above, the Co-59 content was modified to accommodate new fuel loading patterns. Additionally, since the licensee has amassed a notable amount of operating experience with fuel transfers, this experience was applied toward refining some conservatisms in the licensee's original radiological analysis of the operation. For example, the assumption that control rods stored in assemblies in the inner region of the STC had been inserted 10 percent into the active region of the core during their entire time in the reactor has been modified to more accurately reflect actual plant operations; however, the new assumption remains adequately conservative. The staff found that the licensee's characterization of the radioactive source term, including modifications to inputs and assumptions, is reasonable because it is conservative and consistent with the previously accepted approach described in the 2012 license amendments.

Shielding Model

The licensee modeled the STC and HI-TRAC transfer cask shielding performance using the Monte Carlo transport code and received results within a 2 percent error for dose rates presented in the SAR. The staff found the licensee's analysis acceptable because it is consistent with the previously accepted analysis that was described in the 2012 license amendments.

Dose to Members of the Public Who are Onsite

The most limiting portion of a normal fuel transport operation occurs when the STC is being moved from the SFP to the HI-TRAC transfer cask (or vice versa). The licensee evaluated the dose to a member of the public who is onsite at the time of the transport operation using this most limiting scenario. The licensee's evaluation demonstrated that the nominal dose rate at 60 meters from the STC is 1.72 mrem/hour (hr) (0.0172 mSv/hr). This dose rate, combined with the direct radiation dose rate from other sources at the site (e.g., temporary low level storage building and ISFSI), results in a dose rate of 1.83 mrem/hr (0.0183 mSv/hr) and a yearly dose of 14.64 mrem (0.1464 mSv). The licensee also evaluated the dose to an onsite member of the public that results from a 4-hour crane hangup during the transfer operation. A crane hangup occurs when the STC is lifted out of the SFP and is then stuck in position. The crane hangup evaluation resulted in similar dose rates when compared to normal operations; however, because of the change in occupancy time from 8 to 12 hours, the resultant yearly dose was 21.96 mrem (0.2196 mSv). The results of these evaluations are reflected in Table 7.4.7 and Table 7.4.8 of the SAR. These values are unchanged from the 2012 license amendments and are within the limits of 2 mrem/hr (0.02 mSv/hr) and 100 mrem/yr (1 mSv/yr) as required by 10 CFR 20.1301.

The licensee evaluated the dose contribution to onsite members of the public at 20 meters from the HI-TRAC transfer cask. For normal conditions, the licensee assumed onsite presence for 128 hours per year, based on 16 fuel transfers per year, taking 8 hours per transfer. For off-normal conditions, the licensee assumed 240 hours per year based on a 30-day duration of the off-normal condition and 8 hours occupancy each day. The licensee demonstrated that the dose rate for normal conditions at 20 meters from the HI-TRAC is 0.111 mrem/hour (0.00111 mSv/hr). This dose rate, combined with the direct radiation dose rate from other sources at the site (e.g., temporary low level storage building and ISFSI), results in a dose rate of 0.22 mrem/hr (0.0022 mSv/hr) and a yearly dose of 28.18 mrem (0.2818 mSv). For the off-normal condition, the licensee's evaluation yielded similar dose rates; however, because of the change in occupancy time from 128 to 368 hours, the yearly dose was 80.96 mrem (0.8096 mSv). The results of these evaluations are reflected in Table 7.4.14 and Table 7.4.15 of the SAR. These values are unchanged from the 2012 license amendments and are within the limits of 2 mrem/hr (0.02 mSv/hr) and 100 mrem/yr (1 mSv/yr) as required by 10 CFR 20.1301.

In general, members of the public are not expected to access the area when fuel transfers are taking place. Licensee radiation protection personnel will post, limit, and control access per licensee procedures to protect against undue risks from exposure to radiation and radioactive materials. Additionally, IP3 and IP2 TS 5.4.2, TS 5.4.3, and TS 5.4.6 require surveys of the STC and HI-TRAC and limit dose rates at particular locations. If the limits are exceeded, the licensee is expected to verify correct STC loading and, as applicable, perform a written evaluation to determine if transfer operations can proceed without exceeding applicable dose limits. The staff found that this approach satisfies the requirement of 10 CFR 72.104(c); specifically, that the licensee establish operational limits to meet applicable dose limits. It is expected that the licensee's written evaluation would include appropriate consideration of the transfers that have already taken place, as well as those that are anticipated to occur within the same year. Transfer operations would proceed only if the evaluation indicates that the applicable limits will not be exceeded.

Dose at the Controlled Area Boundary (Members of the Public that are Offsite)

The SE associated with the 2012 license amendments (Reference 5) included the following discussion regarding the dose contribution from effluent releases:

> The STC system design must consider dose contributions from effluent releases in addition to direct radiation for demonstrating compliance with the regulatory dose limits. The STC was initially designed to meet a certain leak rate limit. Based on this leak rate limit, the applicant was required to address effluent releases and the resulting doses under normal, off-normal and accident conditions. The NRC staff questioned the applicant's method for evaluating effluents and doses due to effluents. The applicant modified the STC design and its leak testing program so that the STC is now "leak-tight" as defined in ANSI N14.5, "Radioactive Materials – Leakage Tests on Packages for Shipment." The details of the staff's review of the confinement system are given in Section 3.8 of this safety evaluation. Based on the findings of that review and the STC being "leak-tight," the staff concludes there will be no effluent releases and therefore no dose contribution from effluents.

The current LAR dose not propose any changes to the STC leaktight criteria. Therefore, the NRC staff found that the previous conclusion (that there will be no effluent releases and, therefore, no dose contribution from effluents), remains valid for the current LAR.
The licensee evaluated the doses at the controlled area boundary using the same approach it had applied in the 2012 license amendments. Specifically, the licensee evaluated normal and off-normal conditions using a distance of 160 meters from the Hudson River to the owner-controlled area boundary, an occupancy time of 8 hours per fuel transfer operation, and an occupancy time of 500 hours to evaluate doses for off-normal conditions and for accident conditions. The NRC staff evaluated and accepted these assumptions as sufficiently conservative in the SE to the 2012 license amendments. The license evaluated doses to individuals at the controlled area boundary for normal, off-normal, and accident conditions. Table 7.4.16 and Table 7.4.17 of the SAR provide the results of the evaluations for normal and off-normal conditions, respectively. These values are unchanged from the values in the 2012 license amendments and are within the limits of 25 mrem (.25 mSv) to the whole body per year as required by 10 CFR 72.104. Tables 7.4.18, 7.4.19, and 7.4.20 of the SAR provide the results of the evaluations for accident conditions. These values are unchanged from the values in the 2012 license amendments and are within the limits of 5 rem (0.05 Sv) TEDE per year as required by 10 CFR 72.106.

Because the leaktight nature of the STC prevents effluent releases for any condition of the fuel transfer operation, including off-normal and accident conditions, the intake of radioactive material is also not expected, and thus, no internal radiation exposures are expected. Therefore, the only limits that apply from 10 CFR 72.104 and 10 CFR 72.106 are limits that address external radiation. For the postulated scenarios, since the whole body limit of 10 CFR 72.104 and the TEDE limit of 10 CFR 72.106 are generally more restrictive than any other limit that applies to external radiation, compliance with 10 CFR 72.104 and 10 CFR 72.106 can be demonstrated by satisfying the whole body and TEDE limits of those regulations, respectively.

Occupational Doses

Table 7.4.22 of the SAR provides estimated person-(roentgen equivalent man (rem) exposures from loading and unloading one STC. The values presented in the current LAR are unchanged from the 2012 license amendments. Based on its confirmatory analyses in the 2012 license amendments (Reference 5), the NRC staff found the licensee's estimate of dose during a fuel transfer operation to be reasonable and to comply with the occupational dose limits of 10 CFR 20.1201. Since the licensee has not changed its estimated dose, and furthermore, since the licensee's operating experience with multiple fuel transfers to date validates its assumptions and processes, the staff finds that the licensee's occupational dose estimates are reasonable and in compliance with 10 CFR 20.1201. Additionally, based on the staff's review of the procedures in Chapter 10 of the SAR and verification that the procedures are consistent with those reviewed in the 2012 SE, the staff finds that ALARA principles have been adequately incorporated into the system design and operations for the proposed activity through adequate implementation of design features, procedures, and engineering controls designed to achieve doses that are ALARA, as required by 10 CFR 20.1101 and 10 CFR 72.104(b).

4.6.2.4 NRC Staff Finding

Based on the staff's review of the evaluations, methodologies, and statements related to radiation protection in the LAR and the staff's review of the SE for the 2012 license amendments, the NRC staff finds that there is reasonable assurance that adequate protection is maintained throughout the proposed fuel transfer operations and that these operations meet the applicable requirements of 10 CFR Part 20 and 10 CFR Part 72.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendments on November 8, 2017. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration (June 19, 2017; (82 FR 27885), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion 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 the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Vitale, Anthony, letter to U.S. Nuclear Regulatory Commission, "Indian Point Nuclear Power Plant Units 2 and 3, Proposed License Amendment Request Regarding the Inter-Unit Transfer of Spent Fuel," dated December 14, 2016 (ADAMS Accession No. ML16355A067).

2. Vitale, Anthony, letter to U.S. Nuclear Regulatory Commission, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Response to Request for Supplemental Information Needed for Acceptance of Requested Licensing Action Regarding 'Amendment of Inter-Unit Transfer of Spent Fuel,'" dated April 19, 2017 (ADAMS Accession No. ML17114A467).

3. Vitale, Anthony, letter to U.S. Nuclear Regulatory Commission, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Response to Request for Additional Information Regarding Inter-Unit Transfer of Spent Fuel," dated August 16, 2017 (ADAMS Accession No. ML17234A402).

4. Vitale, Anthony, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Inter-Unit Transfer of Spent Fuel, Indian Point Nuclear Generating Unit Nos. 2 and 3," dated October 2, 2017 (ADAMS Accession No. ML17289A653).

5. Boska, John P., letter to Entergy Nuclear Operations, Inc. "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Issuance of Amendments Re: Inter-Unit Spent Fuel Transfer," dated July 13, 2012 (ADAMS Accession No. ML121230011).

6. Revised HI-2094289, "Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at the Indian Point Energy Center," Revision 9, dated October 2, 2017 (ADAMS Accession No. ML17289A655, non-proprietary version).

7. Pickett, Douglas V., letter to Entergy Nuclear Operations, Inc. "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Amendment of Inter-Unit Transfer of Spent Fuel," dated April 11, 2017 (ADAMS Accession No. ML17100A128).

8. Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01, Revision 0, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," dated September 29, 2011 (ADAMS Accession No. ML110620086).

9. Kopp, Laurence memorandum to Collins, Timothy, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1988 (ADAMS Accession No. ML003728001).

10. Pollock, J. E., letter to U.S. Nuclear Regulatory Commission, "Indian Point Nuclear Power Plant Units 2 and 3, Application for Unit 2 Operating License Condition Change and Units 2 and 3 Technical Specification Changes to Add Inter-Unit Spent Fuel Transfer Requirements," dated July 8, 2009 (ADAMS Package Accession No. ML091940176).

11. Shielded Transfer Canister Structural Calculation Package Report No. HI-2084118 dated November 9, 2016 (ADAMS Accession No. ML16355A072 (non-public), Holtec proprietary report).

12. E-mail from Mirzai, M. to Guzman, R., Re: Indian Point Inter-Unit Transfer of Spent Fuel LAR – Updated Clarification Response following Teleconference on October 19, 2017, dated October 27, 2017 (ADAMS Accession No. ML17300A174).

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Date: December 22, 2017

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 – ISSUANCE OF AMENDMENTS RE: AMENDMENT OF INTER-UNIT TRANSFER OF SPENT FUEL (CAC NOS. MF8991 AND MF8992; EPID L-2016-LLA-0039) DATED DECEMBER 22, 2017

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