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

September 4, 2019

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 NO. 2 – ISSUANCE OF AMENDMENT NO. 290 RE: STORAGE OF FRESH AND SPENT NUCLEAR FUEL IN THE SPENT FUEL POOL (EPID L-2017-LLA-0408)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 290 to Facility Operating License No. DPR-26 for Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 11, 2017, as supplemented by letter dated June 6, 2019.

The amendment revises TS Limiting Condition for Operation 3.7.13, "Spent Fuel Pit Storage," and TS 4.0, "Design Features," Section 4.3, "Fuel Storage," to resolve a non-conservative TS associated with TS Limiting Condition for Operation 3.7.13 and negate the need for the associated compensatory measures, while taking no credit for the installed Boraflex panels.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's 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 No. 50-247

Enclosures:
1. Amendment No. 290 to DPR-26
2. Safety Evaluation

cc 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. 290
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 11, 2017, as supplemented by letter dated June 6, 2019, 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:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A, B and C, as revised through Amendment No. 290, are hereby incorporated in the renewed 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 120 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 License and
Technical Specifications

Date of Issuance: September 4, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 290

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 pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove Page	Insert Page
3.7.13-1	3.7.13-1
3.7.13-2	3.7.13-2
3.7.13-3	3.7.13-3
3.7.13-4	3.7.13-4
3.7.13-5	3.7.13-5
3.7.13-6	3.7.13-6
3.7.13-7	3.7.13-7
---	3.7.13-8
---	3.7.13-9
---	3.7.13-10
---	3.7.13-11
4.0-1	4.0-1
4.0-2	4.0-2

(3) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use, at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; Amdt. 42 10-17-78

(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 renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) 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. 290, are hereby incorporated in the renewed 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 and 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 categorized in accordance with Table 3.7.13-1 or, if pre-categorized, Table 3.7.13-2.

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

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

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

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

ACTIONS

- NOTE -

Separate Condition entry is allowed for each fuel assembly.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------- **- NOTE -** LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable location.	Immediately

SURVEILLANCE REQUIREMENTS

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

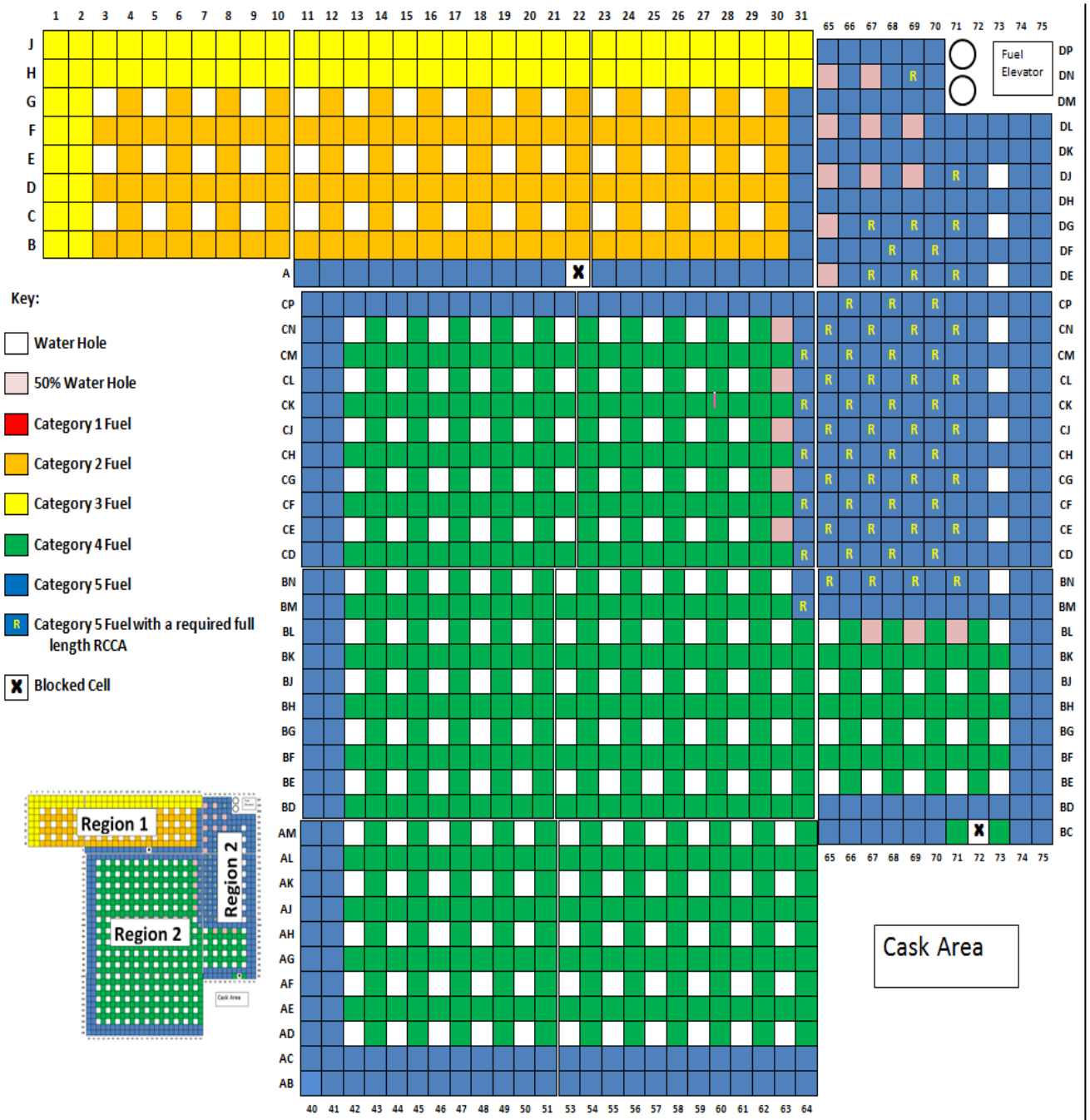

Figure 3.7.13-1 (page 1 of 2)
Allowable Spent Fuel Pit Storage Locations for Category 1 through Category 5 Fuel Assemblies in Regions 1 and 2

-Notes-

1. Fuel assembly Categories are ranked in order of relative reactivity, from Category 1 to 5. Category 1 fuel assemblies have the highest reactivity, and Category 5 fuel assemblies have the lowest.

2. Fuel assembly categorization for assembly IDs after X for IP2 and after AA for IP3 must be performed in accordance with Table 3.7.13-1.

3. Fuel assembly Categories for IP2 assembly IDs A through X are located in Table 3.7.13-2.

4. Fuel assembly Categories for IP3 assembly IDs A through AA are located in Table 3.7.13-3.

5. Fuel assemblies of any higher numbered Category can be stored in any cell location that allows for a lower numbered Category. For example, a Category 5 fuel assembly can be stored in Category 1, 2, 3, 4, and 5 cells. Any cell may be empty.

6. Category 1 fuel assemblies that contain a full length RCCA may be stored in any Category 4, 3, 2, or 1 cell.

7. Category 2, 3 or 4 fuel assemblies that contain a full length RCCA may be stored in any Category 5 cell that does not require an inserted RCCA or in any Category 4, 3, 2, or 1 cell.

8. A Water Hole may contain up to 50% of absorber material by volume in the active fuel area. Stainless steel and Inconel meet the definition of absorber material. There is no restriction for non-actinide material outside of the active fuel area.

9. A 50% Water Hole may contain up to 50% of any non-actinide material by volume in the active fuel area. Zirconium meets the definition of non-actinide material. There is no restriction for non-actinide material outside of the active fuel area.

10. A Blocked Cell has the same requirements as a Water Hole.

11. A checkerboard area consists of every other cell being a Water Hole.

12. An area of Category 1 fuel assemblies may be formed in Region 1. The Category 1 area must be formed by replacing the Region 1 arrangement shown in this figure with an area of Category 1 fuel assemblies in accordance with the following criteria (see examples in Figure 3.7.13-2):

    a) Category 1 fuel assemblies must be face adjacent to at least three Water Holes and not face adjacent to another Category 1 assembly.

    b) Category 2 fuel assemblies must not be face adjacent to more than one Category 1 fuel assembly.

    c) Category 3 and Category 5 locations in Figure 3.7.13-1 may not be moved.

13. A checkerboard area of Category 1 fuel assemblies may be formed in Region 2. All four sides of the checkerboard area must be rows of Water Holes.

14. The edge of Region 2 next to the pool wall or cask loading area can be considered to be a row of Water Holes.

Figure 3.7.13-1 (page 2 of 2)

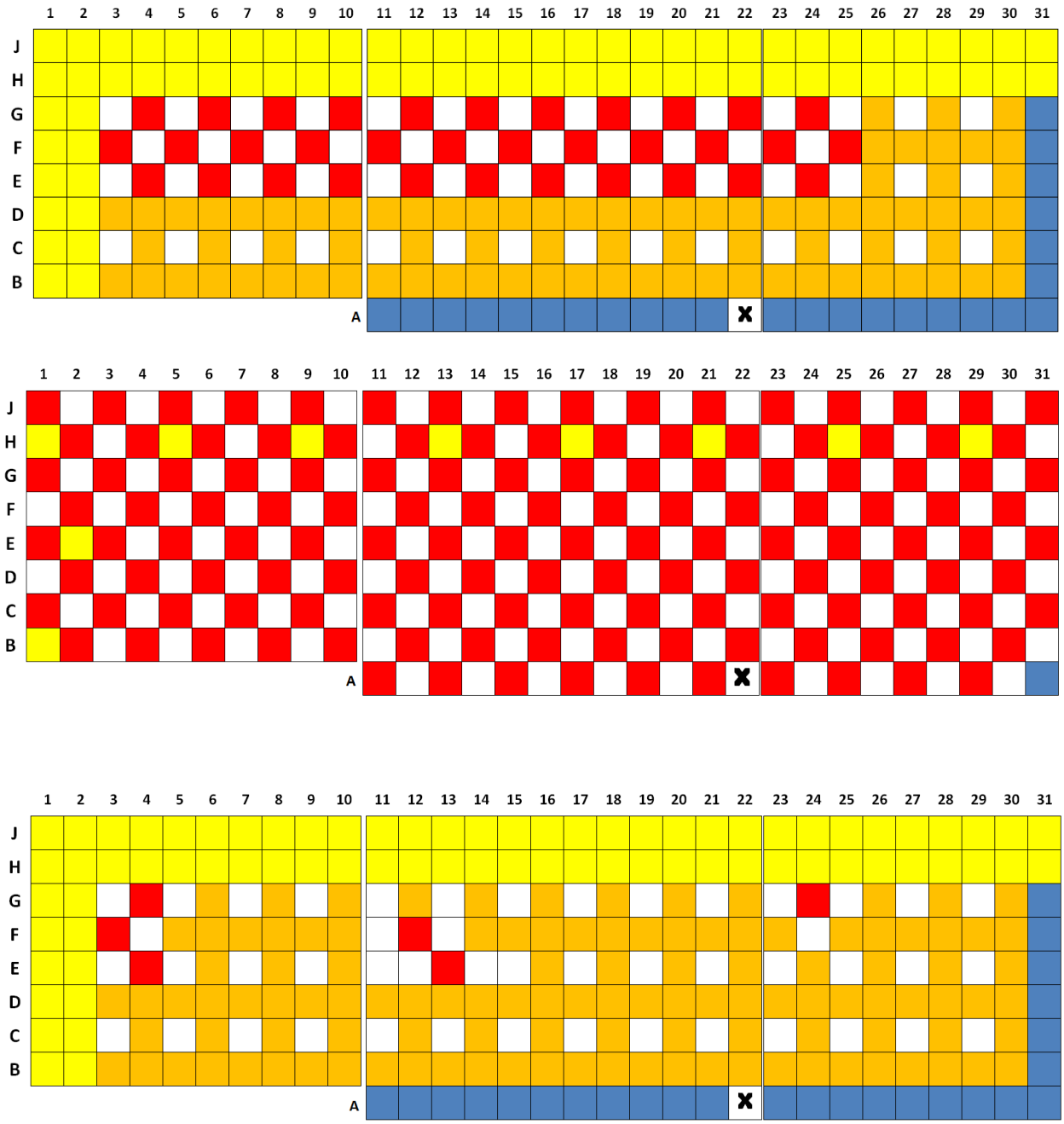

Figure 3.7.13-2
Examples of Allowable Spent Fuel Pit Storage Locations for Category 1 Fuel Assemblies in Region 1

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

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

Where:

E is enrichment in wt% U-235[(e)],

CT is cooling time in years[(f)], and

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

and:

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

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

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

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

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

(e) Fuel assemblies at enrichments less than 4.2 wt% U-235 must use 4.2 wt% U-235 in the Category 4 equation.

(f) Fuel assemblies with cooling times of more than 25 years must use 25 years in the Category 4 equation.

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

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

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

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

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

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

1 FRSB is the Fuel Rod Storage Basket

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

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

4.0 DESIGN FEATURES

4.1 Site Location

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

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone, as defined in 10 CFR 100.3, is 520 meters and 1100 meters, respectively. For the purpose of satisfying 10 CFR Part 20, the “Restricted Area” is the same as the “Exclusion Area” shown in UFSAR, Figure 2.2-2.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy-4 or ZIRLO fuel rods. Fuel shall have a U-235 enrichment of $\leq$ 5.0 weight percent. Limited substitutions of Zircalloy-4, ZIRLO or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver indium cadmium, clad with stainless steel, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent, and poisons, if necessary, to meet the limit for k_{eff},

b. $k_{eff} \leq 0.95$ when flooded with borated water, $k_{eff} < 1.0$ if fully flooded with unborated water, and

c. Each fuel assembly categorized based on initial enrichment, burnup, cooling time, averaged assembly peaking factor, and number of Integral Fuel Burnable Absorbers (IFBA) rods with individual fuel assembly storage location within the spent fuel storage rack restricted as required by Technical Specification 3.7.13.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent, and poisons, if necessary, to meet the limit for k_{eff},

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, and

c. A 20.5 inch center to center distance between fuel assemblies placed in the storage racks to meet the limit for k_{eff}.

4.3.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pit below a nominal elevation of 88 feet, 6 inches.

4.3.3 Capacity

The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 269 fuel assemblies in Region I and 1105 fuel assemblies in Region II.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 290

FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17354A007), as supplemented by letter dated June 6, 2019 (ADAMS Accession No. ML19157A309), Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) for the Indian Point Nuclear Generating Unit No. 2 (Indian Point 2) Technical Specifications (TSs).

The amendment would revise TS Limiting Condition for Operation (LCO) 3.7.13, "Spent Fuel Pit Storage," and TS 4.0, "Design Features," Section 4.3, "Fuel Storage." The proposed changes would resolve a non-conservative Indian Point 2 TS associated with TS LCO 3.7.13 and negate the need for the associated compensatory measures, while taking no credit for the installed Boraflex panels.

Specifically, the amendment would implement the following items associated with fuel storage at Indian Point 2: (1) eliminate reactivity credit for Boraflex panels in current Regions 1B and 2B of the spent fuel pool (SFP); (2) revise allowed storage patterns for fuel assemblies in the SFP to meet effective neutron multiplication factor (K_{eff}) requirements under normal and accident conditions; (3) revise alphanumeric designations of spent fuel regions from Regions 1A, 1B, 2A, and 2B to Regions 1 and 2; (4) allow use of control element assemblies (CEA) in preassigned locations; and (5) require the use of spent fuel rack-cell blocking devices in two preassigned locations.

Additionally, the licensee has decided to permanently shut down Indian Point 2 and 3. The licensee submitted its notification of permanent cessation of power operations for Indian Point 2 and 3 on February 8, 2017 (ADAMS Accession No. ML17044A004). The letter states the licensee will permanently cease operation of Indian Point 2 by April 30, 2020, and Indian Point 3 by April 30, 2021, subject to possible operating extensions until April 30, 2024, and April 30, 2025, respectively, pursuant to an agreement reached by the licensee with the State of New York and other entities under circumstances specified in the agreement. The U.S. Nuclear

Regulatory Commission (NRC, the Commission) staff considered the shutdown of Indian Point 2 and 3 in its review of this LAR.[1]

The supplemental letter dated June 6, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 13, 2018 (83 FR 10916).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance documents that the NRC staff used in the review of the LAR are listed below.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC). GDC 62, "Prevention of criticality in fuel storage and handling," requires that, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

Per 10 CFR 50.68(a), each holder of an operating license shall comply with either 10 CFR 70.24 or the requirements in 10 CFR 50.68(b). The licensee has elected to meet 10 CFR 50.68(b), and accordingly, must comply with the following requirements:

(1) 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.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, 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 unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum

[1] A separate LAR from Entergy dated April 15, 2019 (ADAMS Accession No. ML19105B241), is under NRC staff review, proposing changes to the Indian Point 2 TSs to reflect the permanent cessation of reactor operation and permanent defueling of the reactor (i.e., permanently defueled TSs).

> 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.

The regulations in 10 CFR 50.36, "Technical specifications," contain the requirements for the content of TSs. The regulations in 10 CFR 50.36(b) require TSs to be derived from the analyses and evaluation included in the safety analysis report and amendments thereto. As required by 10 CFR 50.36(c)(4), the TSs will include design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36.

3.0 TECHNICAL EVALUATION

3.1 Background

The proposed amendment requested TS revisions to support removal of criticality analysis credit for Boraflex neutron absorber panels for the Indian Point 2 SFP. The following evaluation presents the results of the NRC staff's review of the nuclear criticality safety (NCS) analysis, which was provided as Enclosures 1 (proprietary) and 2 (non-proprietary) to the licensee's December 11, 2017, application, and updated through the subsequent supplemental letter dated June 6, 2019. The criticality safety basis is thus composed of both the original criticality analysis and the licensee's supplemental letter in response to the NRC staff's request for additional information (RAI).

The licensee's NCS analysis, as supplemented through RAI responses, describes the methodology and analytical models used to show that the SFP storage rack maximum K_{eff} will be less than 1.0 when flooded with unborated water for normal conditions, and less than or equal to 0.95 when flooded with borated water for normal and credible accident conditions at a 95-percent probability, 95-percent confidence level.

The Indian Point 2 SFP storage racks represent a typical pressurized water reactor (PWR) SFP, two-region design with a neutron-absorbing material (NAM) positioned between the fuel assemblies. In this case, the NAM is Boraflex. Region 1 was primarily intended for the storage of fresh or lightly burned fuel. Region 2 was primarily intended for storage of fuel that had completed its service life.

In a previous license amendment to address Boraflex degradation, Indian Point 2 designated parts of Regions 1 and 2 that do not credit Boraflex (Sub-Regions 1A and 2A) and parts that did (Sub-Regions 1B and 2B). This, coupled with a fuel storage approach to primarily place recently irradiated fuel into the areas that did not credit Boraflex, would prolong the useful life of the Boraflex in the areas where it was credited and maximize the useful SFP storage capacity. Eventually, the Boraflex in the areas where it was being credited degraded to a point where continued credit for Boraflex was impractical. This license amendment would remove all Boraflex credit in the Indian Point 2 SFP.

As additional background, the licensee stated the following in its LAR:

> In 1990, SFP storage capacity was increased from 980 fuel assemblies to 1374 fuel assemblies by the installation of high-density racks that reduced the distance

between adjacent fuel assemblies. In order to maintain $k_{95/95}$ of the SFP within the limits of 10 CFR 50.68(b) the following were required:

1) installation of Boraflex absorber panels between spent fuel rack cells; and,
2) restrictions placed on fuel assembly storage location within the SFP based on initial enrichment and burnup.

In 2001, Entergy submitted a license amendment request supported by the CAOR [criticality analysis of record] [ADAMS Accession No. ML012680336] to address the observance of unexpected Boraflex thinning and gaps. The CAOR took varying amounts of credit for Boraflex for reactivity hold-down and a Boraflex monitoring program was initiated to ensure that the amounts of Boraflex credit taken in the CAOR remain valid.

The CAOR paralleled analyses performed by the industry at that time in accordance with an NRC approved methodology. This analysis formed the basis for TS amendment 227 [ADAMS Accession No. ML021230367], issued in 2002, and is the basis for the IP2 [Indian Point 2] fuel portion of LCO 3.7.13 SFP. This analysis demonstrated compliance with 10 CFR 50.68 and TS 4.3.1.1.

In 2009, Indian Point submitted a license amendment request [ADAMS Accession No. ML091940177] to allow the transfer of spent fuel from IP3 to IP2. The NRC approved the license amendment request in 2012 [ADAMS Accession No. ML121230011]. During review of the license amendment request the NRC concluded that IP3 fuel could only be stored in Region 1-2 of the IP2 SFP. This restriction is reflected in LCO 3.7.13 SFP. The NRC also noted that LCO 3.7.13 was non-conservative. The NRC found that the methodology used in the CAOR could not be used to support a current license amendment request and that IN [Information Notice] 2011-03 [ADAMS Accession No. ML103090055] had not been addressed. In response, Entergy evaluated the CAOR and determined that LCO 3.7.13 was non-conservative in SFP Region 2-2 and implemented the necessary compensatory measures to ensure continued compliance with 10 CFR 50.68.

Continued compliance of 10 CFR 50.68 is also dependent on the results of the Boraflex monitoring program. This program determined that the Boraflex in the SFP degrades with time. This determination is consistent with the NRC documented observance of Boraflex degradation [Generic Letters 96-04 and 2016-01 [ADAMS Accession No. ML031110008 and ML16097A169, respectively].

In order to address the non-conservative TS and Boraflex degradation, Entergy submitted a CSA [criticality safety analysis] in 2015 that credited yet to be installed metal-matrix-composite neutron absorber inserts [ADAMS Accession No. ML14329A194]. The intent was to install these inserts concurrent with NRC review and approval of a planned future license amendment request that would have credited the CSA. During review, the NRC staff requested additional information in June 2015 [ADAMS Accession No. ML15148A403] and Entergy issued a response in August 2015 [ADAMS Accession No. ML15261A527]. In November 2015, the NRC staff issued its evaluation of the CSA, concluding that

"The NRC staff finds that the CSA methodology is acceptable for use at IP2" [ADAMS Accession No. ML15292A161]. However, due to the prolonged schedule for the installation of the inserts, Entergy determined that a new approach, that did not credit inserts, was needed.

The new approach, as documented in the proposed CSA (Enclosure 1 of the LAR), does not credit inserts, but instead credits empty cells, RCCAs [rod cluster control assemblies], and neutron leakage along the outer two storage rows of the pool for criticality control. This new approach for the resolution of the non-conservative TS and Boraflex degradation was the subject of an NRC preapplication public meeting held on July 26, 2017 [ADAMS Accession No. ML17200C927]. As noted at the public meeting, this approach is based on the previously submitted CSA methodology [ADAMS Accession No. ML14329A194] which was accepted by the NRC, with several improvements including a depletion analysis with peaking factor and axial blanket credit, validation updates regarding RCCAs and temperature dependence, the inclusion of grid growth and creep biases, eccentric placement bias, and full pool calculations.

3.2 Proposed Change

3.2.1 NCS Analyses and Fuel Storage Requirements

There are several proposed TS changes that either impact NCS analyses or implement changes in fuel storage requirements.

The proposed TSs significantly revise the organization and storage requirements for the SFP. The proposed TS change would remove the subdivision of each region, returning to the original Region 1 and Region 2 designations. While this might seem like a simplification, it is not. The proposed Region 1 will credit burnup, actinide decay, empty cells, blocked cells, wall leakage, and integral burnable absorbers. The proposed Region 2 will credit burnup, actinide decay, empty cells, blocked cells, wall leakage, and control rods. With incorporation of the proposed change, each region will apply its requirements in various combinations, depending on where the cell is located and what classification of fuel assembly is in the adjacent storage cells. Additionally, the change includes alternate storage requirements for contingency situations that could occur adjacent to fuel stored in accordance with the standard requirements.

In its June 6, 2019, letter, the licensee made two modifications to the proposed TSs. One was to add a clarification, and the other was to make TS 4.3.1.1 consistent with the Standard Technical Specifications.

Specifically, the amendment would revise or add the following TSs, TS figures, and TS tables, as provided in Attachment 1 of the Enclosure to the licensee's supplemental letter dated June 6, 2019:

- TS 3.7.13
- TS Figure 3.7.13-1
- TS Figure 3.7.13-2
- TS Table 3.7.13-1
- TS Table 3.7.13-2

- TS Table 3.7.13-3
- TS 4.3

3.3 Method of Review

This safety evaluation involves a review of the licensee's NCS analyses for the Indian Point 2 SFP, which was provided as Enclosures 1 (proprietary) and 2 (non-proprietary) to the licensee's December 11, 2017, application, and updated through the subsequent supplemental letter dated June 6, 2019. The review was performed consistent with Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "Spent Fuel Storage," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The NRC staff also used an internal memorandum dated August 19, 1998, containing guidance for performing the review of SFP NCS analysis (NRC memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (ADAMS Accession No. ML003728001) (hereafter the "Kopp memo"). While the Kopp memo does not specify a methodology, it does provide some guidance on the more salient aspects of an NCS analysis, including computer code validation. The guidance is germane to boiling-water reactors and PWRs for both borated and unborated fuel storage pools. The Kopp memo has been used during NRC staff review of virtually every light-water reactor SFP NCS analysis thereafter, including this LAR analysis.

The NRC staff also used an interim staff guidance document entitled, "'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 2011 (ADAMS Accession No. ML110620086); notice of availability published in the *Federal Register* on October 13, 2011 (76 FR 63676), for review of SFP criticality analyses. The guidance in DSS-ISG-2010-01 is used by the NRC staff to review NCS analyses for the storage of new and spent nuclear fuel as it applies to: (i) future applications for construction and/or operating licenses, and (ii) future applications for license amendments and requests for exemptions from compliance with applicable requirements that are approved after the date of this interim staff guidance.

3.4 SFP NCS Analysis Review

3.4.1 SFP NCS Analysis Method

There is no generic or standard NRC-approved methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the Indian Point 2 SFP are described in the criticality analysis, which was provided as Enclosures 1 (proprietary) and 2 (non-proprietary) to the licensee's December 11, 2017, application, and updated through the subsequent supplemental letter dated June 6, 2019. Some SFP criticality analysis potential non-conservatisms were identified during the review, but as will be discussed below, sufficient margin is built into the analysis methodology to offset the potential non-conservatisms. The methodology is specific to this analysis and is not appropriate for other applications.

3.4.1.1 Computational Methods

The Indian Point 2 NCS analysis considers the decrease in fuel reactivity typically seen in PWRs, as the fuel is depleted during reactor operation. This approach is frequently used in PWR NCS analyses and is sometimes referred to as burnup credit. Burnup credit NCS analysis requires a two-step process. The first step relates to depletion where a computer code simulates the reactor operation to calculate the changes in the fuel composition of the fuel assembly. The second step is a modeling of the depleted fuel assembly in the SFP storage racks and the determination of the system K_{eff}. The validation of the computer codes in each step is a significant portion of the analysis. Since the Indian Point 2 NCS analysis credits fuel burnup, it is necessary for the NRC staff to consider validation of the computer codes and data used to calculate burned fuel compositions, and the computer code and data that utilize the burned fuel compositions to calculate K_{eff} for systems with burned fuel.

For the depletion step, burnup credit NCS analyses typically involve use of a computer code approved by the NRC for the purposes of performing reactor core simulation analyses. Those computer codes have an NRC safety evaluation governing their use, including any necessary limitations and conditions. Additionally, those NRC-approved codes are being used by numerous licensees to perform reactor core analysis, thereby providing a feedback mechanism, should significant differences be observed between reactor core analyses and actual reactor core performance. Indian Point 2 used the T5-DEPL TRITON module depletion sequence from SCALE 6.1.2 to perform its depletion step. Prior to this LAR, the use of T5-DEPL TRITON had been used in an SFP NCS licensing application submitted to the NRC. While that application was eventually approved, there is still no NRC-approved topical report governing its use, nor are there any licensees using it to perform reactor core analysis, thereby providing a feedback mechanism should significant differences be observed between reactor core analyses and actual reactor core performance. Therefore, the NRC staff concluded that the use of the T5-DEPL TRITON depletion sequence from SCALE 6.1.2 to perform the depletion should be justified on an application- specific basis.

3.4.1.1.1 Depletion Computer Code Validation

NET-28091-0003-01, Revision 0, Section 5.6, "Depletion Analysis Model," describes the licensee's analysis to justify the use of the T5-DEPL TRITON depletion sequence in SCALE 6.1.2 as the depletion code of record for this analysis by comparing three sets of TRITON delta k-effective to three sets of CASMO-5 delta k-effective and by listing the differences between them. The NRC staff was concerned that using the comparison of the delta between two state points from one computer code to the delta between two state points from another computer code is inadequate to determine that one code is acceptable for use based on similarity to the other code, as this method of comparison says nothing about how the codes compare at the actual state points. In response to the NRC staff's RAI regarding this matter, the licensee provided information that compared the SFP K_{eff} results of T5-DEPL TRITON depletion sequence in SCALE 6.1.2 and CASMO-5. That comparison indicates that, within the confines of this analysis, using TRITON is conservative. The staff's evaluation regarding the licensee's use of this depletion uncertainty methodology is explained in Section 3.4.3.3.1 below.

3.4.1.1.2 SFP K_{eff} Computer Code Validation

The study used to support validation of K_{eff} calculations using the SCALE 6.1.2 SAS5 sequence was documented in Appendix A of Enclosures 1 (proprietary) and 2 (non-proprietary) to the licensee's December 11, 2017, application, and was updated through the subsequent

supplemental letter dated June 6, 2019. The validation set includes critical configurations from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* and French Haut Taux de Combustion (HTC) critical experiments from NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (ADAMS Accession No. ML082880452). The suite of experiments was similar to those used in NRC-approved license amendments for other facilities. The validation was performed in a manner consistent with NUREG-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (ADAMS Accession No. ML050250061), and included an evaluation for a temperature bias. Therefore, this validation of SCALE 6.1.2 CSAS5 is acceptable.

The licensee modeled the entire SFP to determine K_{eff}. Convergence of Monte Carlo codes can be a concern for very large systems, especially if there are different segments that are loosely coupled neutronically. The NRC staff asked the licensee to provide additional information on the convergence of its whole pool models. In its June 6, 2019, letter, the licensee provided additional information on the convergence of is modeling of the entire SFP. With the supplemental information, the NRC staff finds there is reasonable assurance that the models have sufficiently converged.

3.4.2 SFP and Fuel Storage Racks

3.4.2.1 SFP Water Temperature

NRC guidance provided in the Kopp memo states the NCS analysis should be done at the temperature corresponding to the highest reactivity. The licensee performed a sensitivity analysis for each region of the SFP. Region 1 was determined to have a monotonically increasing reactivity with temperature. The licensee used the maximum design-basis temperature for the SFP for its Region 1 calculations. Region 2 did not have a monotonically increasing reactivity with temperature. Instead, there was a reactivity peak at 70 degrees Celsius (°C) or 158 degrees Fahrenheit (°F). The computer cases used to determine the Region 2 loading patterns were performed at 70 °C.

3.4.2.2 SFP Storage Rack Models

The Indian Point 2 SFP storage racks are constructed of stainless steel boxes with Boraflex positioned on the outside of each face. The NAM is held in place by a thin sheet of stainless steel termed a 'wrapper' or 'sheathing.' The Indian Point 2 NCS uses the terms interchangeably. The wrapper is open at the top and tack welded along the sides and bottom. For Region 1, the boxes are attached to a structural grid that provides some space between each box. The resultant cell is attached to a base plate, forming a Region 1 storage module. The Region 1 design has two sheets of NAM and a water gap between each storage cell. Region 2 has similar boxes attached at the corners forming what is called an "egg crate" with "formed" cells being created between the manufactured cells. The "egg crate" is completed using filler panels to enclose "formed" cells on the periphery. The resultant cell is attached to a base plate, forming a Region 2 storage module. Region 2 has one sheet of NAM and no water gap between the storage cells.

The wrapper and cell wall create a local environment that surrounds the NAM and is open to the SFP water. When a licensee performs an NCS that removes credit for the Boraflex, the volume that is taken up by the Boraflex is modeled as SFP water. While initially a conservative model, eventually the Boraflex will degrade to the point of being ineffective, and this modeling will become the actual condition.

3.4.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The nominal dimensions of cell center-to-center spacing, cell inside dimension, cell walls, wrapper, and Boraflex boxes were used in design-basis calculations. To provide uncertainties associated with rack manufacturing tolerances and uncertainties, the licensee performed sensitivity calculations for each region and for each permitted storage categorization, except for Category 1. For Category 1, the licensee used the Category 3 uncertainties. The NCS posits that there is sufficient margin in the storage of Category 1 fuel to use estimated uncertainties. Table 8.1 of the NCS indicated the allowed storage configuration of Category 1 fuel in Region 1 will have a K_{eff} of 0.8675 using the estimated uncertainties. Based on the margin associated with the allowed storage configuration of Category 1 fuel in Region 1, the NRC staff concludes it is acceptable to use estimated uncertainties for Category 1 fuel. The other uncertainties are those typically used and are considered acceptable.

3.4.3 Fuel Assembly

3.4.3.1 Bounding Fuel Assembly Design

The fuel assemblies used at Indian Point 2 are all 15x15 assemblies manufactured by Westinghouse. The Indian Point 2 SFP NCS did not determine a bounding fuel assembly design. Instead, the analysis divided the fuel assemblies into "batches" with similar characteristics. The "batches" were primarily based on depletion parameters but also included the specific fuel designs associated with those "batches." Most of the "batches" consist of legacy fuel where performing an NCS analysis based on the use of newer and potential future fuel designs and operating parameters could be overly conservative. This approach is acceptable but requires an NCS analysis for each "batch" and a method to identify the appropriate batch for the fuel assemblies. The licensee has appropriately completed its NCS analyses in this manner and has pre-characterized the legacy fuel assemblies. The pre-characterization is captured in a table that is being incorporated into the Indian Point 2 TSs. While the typical practice is to consider more prospects for future fuel assembly designs to avoid re-work and/or re-licensing, the potential for future fuel designs is nil, given that the licensee is shutting down both Indian Point 2 and 3. Therefore, the NRC staff finds this approach acceptable.

3.4.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The Indian Point 2 NCS analysis includes the most significant manufacturing tolerances in a manner consistent with DSS-ISG-2010-01 and neglects the less significant tolerances. DSS-ISG-2010-01 allows independent uncertainties to be statistically combined. The licensee used the Root Sum Squares method to statistically combine the uncertainties. The result of the Root Sum Squares method is dominated by the larger uncertainties. The NRC staff finds that due to the sufficient margin available in the analysis, the licensee's approach is acceptable.

3.4.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on U-235 enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values would also apply to the spent nuclear fuel. Common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. The NRC staff has

previously accepted the practice. The practice is acceptable in this LAR due to the margin available in the analysis. The characterization of spent nuclear fuel is complex. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: depletion uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup. These characteristics are evaluated in the following sections.

3.4.3.3.1 Depletion Uncertainty

With respect to depletion uncertainty, the Kopp memo states, "In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." This method represents the engineering judgment of the memo's author, based on his experience with the ability of approved NRC depletion and reactor operation simulation codes and methods to accurately predict the post-irradiated isotopic concentrations of nuclear fuel. The licensee's NCS used this method to determine the depletion uncertainty. However, as noted in Section 3.4.1.1, the licensee's NCS did not use an approved NRC depletion and reactor operation simulation code. The NRC staff evaluated the licensee's use of the T5-DEPL TRITON depletion sequence from SCALE 6.1.2 to perform its depletion step in Section 3.4.1.1 and finds it acceptable. Because the licensee demonstrated its use of the T5-DEPL TRITON depletion sequence from SCALE 6.1.2 is conservative relative to CASMO-5, which is an approved NRC depletion and reactor operation simulation code and method, the NRC staff considers it acceptable for the licensee to use the Kopp memo depletion uncertainty determination method in the confines of this analysis.

3.4.3.3.2 Axial Apportionment of the Burnup or Axial Burnup Profile

Another important aspect of fuel characterization is the selection of the axial burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to neutron leakage. If a uniform axial burnup profile is assumed, the burnup at the ends is over-predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis" (ADAMS Accession No. ML031110292)," has shown that, at assembly burnups above about 10 to 20 gigawatt-days per metric ton of uranium (GWd/MTU), the use of a uniform axial burnup profile results in an under-prediction of K_{eff}. Generally, the under-prediction becomes larger as burnup increases. This is what is known as the "end effect." Proper selection of the axial burnup profile is necessary to ensure K_{eff} is not under-predicted due to the end effect.

The Indian Point 2 SFP NCS analysis indicated that it used the bounding axial burnup profiles from NUREG/CR-6801. However, the licensee made several changes to how the profiles are typically used, some that are inconsistent with how NUREG/CR-6801 describes the use of the profiles. The NRC staff requested additional information from the licensee to address the apparent discrepancy. The licensee responded with information that makes a node-by-node comparison between the relative burnup that the licensee used in the node and the relative burnup from NUREG/CR-6801 for the same node. In most cases, the licensee's method was conservative. To ensure the licensee's method was appropriate overall, it also amalgamated several nodes for the comparison. The amalgamated nodes showed the licensee's method was either conservative or the difference was insignificant.

NUREG/CR-6801 recommends using at least 18 nodes to model the axial burnup profiles. The NRC staff has accepted the use of fewer nodes, provided the method was conservative. The licensee used fewer than 18 nodes, and its method was not observably conservative. In response to the NRC staff's RAI, the licensee provided information indicating that while its method is not conservative relative to an 18-node model, it is essentially the same as an 18-node model. Therefore, the NRC staff considers the licensee's treatment of axial burnup profiles to be acceptable.

3.4.3.3.3 Radial Burnup Distribution

Due to the neutron flux gradients in the reactor core, assemblies can show a radially tilted burnup distribution (i.e., differences in burnup between portions or quadrants of the cross section of the assembly). The Indian Point 2 analysis did not consider the effect of radial burnup distribution on reactivity.

The impact of radial burnup gradients may be estimated by comparing the distribution of radial burnup tilt information provided in Figure 3-4 of the U.S. Department of Energy document, DOE/RW-0496, "Horizontal Burnup Gradient Datafile for PWR Assemblies," with information on the sensitivity of K_{eff} to radial burnup tilt provided in Section 6.1.2 of NUREG/CR-6800, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs" (ADAMS Accession No. ML031110280). From DOE/RW-0496, the maximum quadrant deviation from assembly average burnup had been observed to be less than 25 percent at low assembly average burnups (burnup < 20 GWd/MTU) and was observed to decrease with burnup, generally being less than 10 percent at burnups above 20 GWd/MTU. Combining these radial tilt bounding estimates with the K_{eff} sensitivity information provided in NUREG/CR-6800, the NRC staff's review of these radial burnup tilts indicate that K_{eff} could increase by as much as 0.002 Δk. Based on the above information, the staff finds that its potential impact is small, and it is conservative to consider this effect as a bias. This reactivity effect is accommodated within the analysis margins.

An exception to this consideration is several fuel assemblies that spent two consecutive cycles on the periphery before being discharged to the SFP. Fuel assemblies on the periphery, while lower in power, also have a much steeper flux gradient than interior fuel assemblies. Fuel assemblies that resided on the periphery for two consecutive cycles could have a more significant radial burnup profile than considered in NUREG/CR-6800. The NRC staff requested additional information from the licensee to address the potential. The licensee responded by stating that those fuel assemblies had been moved between cycles such that the flux gradient was reversed from one cycle to the next. Additionally, the licensee provided an estimate of the maximum effect. The licensee's estimate was similar to that described in NUREG/CR-6800 and was within the conservatism of this analysis. Therefore, the NRC staff finds that the licensee's analysis is conservative regarding the effect of radial burnup distribution on reactivity, and is acceptable.

3.4.3.3.4 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel" (ADAMS Accession No. ML003688150), provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on NCS analysis in storage and transportation casks, the basic principles with respect to the depletion analysis apply generically, since the phenomena occur in the reactor as the fuel is being depleted. The results have some applicability to Indian Point 2

NCS analyses. The basic strategy for this type of analysis is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field, resulting in maximum plutonium-239/241 production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar (i.e., Doppler broadening/spectral hardening of the neutron field resulting in increased plutonium-239/241 production). NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. The largest effect appears to be due to moderator temperature. NUREG/CR-6665 approximates the moderator temperature effect in an infinite lattice of high burnup fuel to be 90 percent mille per degree Kelvin (°K). Thus, a 10 °F change in moderator temperature used in the depletion analysis would result in 0.005 Δk. The effects of each core operating parameter typically have a burnup or time dependency.

The staff guidance in DSS-ISG-2010-01, Revision 0, references NUREG/CR-6665 and recommends the use of bounding parameters for the depletion analysis; however, the Indian Point 2 NCS analysis did not use bounding parameters. Rather, Indian Point 2 used the aforementioned "batches." While the licensee's analysis used bounding parameters for each "batch," it also applied a fuel assembly's lifetime "averaged assembly peaking factor" to those parameters when performing the analysis to determine the pre-characterization of the legacy fuel. This was a unique non-bounding approach that was not fully justified initially in the LAR. As indicated in the licensee's June 6, 2019, letter (RAI 5), the licensee found that using a fuel assembly's lifetime "averaged assembly peaking factor" could be non-conservative if there were higher assembly peaking factors applied for the associated fuel assemblies that were discharged from the core to the SFP. The licensee, however, stated the non-conservatism would be small and affirmed the current SFP inventory had margin in the form of excess burnup. Based on its review of the licensee's RAI response, the staff finds that the licensee has appropriately assessed the current inventory in the SFP, and all assemblies with increasing peaking factors have sufficient margin to account for any resulting non-conservatism associated with applying the fuel assembly's lifetime "averaged assembly peaking factor." Therefore, the NRC staff finds the licensee's use of a fuel assembly's lifetime "averaged assembly peaking factor" acceptable for this application. The staff also notes that with the licensee planning to permanently shut down Indian Point 2 and 3, the potential for the licensee to operate with higher assembly peaking factors than assumed in its analysis is minimal.

3.4.3.3.4.1 Core Moderator and Core Fuel Temperatures

The licensee used the fuel assembly's lifetime "averaged assembly peaking factor" to determine the core moderator and core fuel temperatures in its depletion calculation. Using the fuel assembly's lifetime "averaged assembly peaking factor" on the core moderator temperature and the core fuel temperature makes the higher peaking factors for assemblies discharged to the SFP pool non-conservative. As indicated in the preceding section, the licensee confirmed that the current SFP inventory had margin in the form of excess burnup, and the analysis had sufficient margin to account for the non-conservatism. The staff considers this margin acceptable to accommodate the non-conservatism applicable to the current fuel inventory. Therefore, the NRC staff finds use of the core moderator and core fuel temperatures acceptable for this application.

3.4.3.3.4.2 Core Soluble Boron Concentration

The common practice in the depletion portion of SFP NCS analyses has been to use a constant soluble boron concentration rather than a time-dependent soluble boron letdown curve. This

significantly simplifies the modeling. The use of a constant cycle average soluble boron for the depletion modeling is believed to be conservative, based primarily on the paper by J. C. Wagner, "Impact of Soluble Boron Modeling for PWR Burnup Credit Criticality Safety Analyses," *Transactions of the American Nuclear Society*, 89, pp. 120 (2003). The work in the Wagner paper models three identical complete cycles and compares the effect of modeling a constant cycle-average soluble boron with use of a linear time-dependent soluble boron letdown curve. While the simple modeling is adequate to demonstrate the concept, it does show the methodology is non-conservative for mid-cycle shutdowns and/or incomplete cycles. The work also does not address the use of constant soluble boron based on other methods than cycle average.

The licensee's analysis cites the above-mentioned paper by J. C. Wagner to justify its use of a constant soluble boron concentration rather than a time-dependent soluble boron letdown curve for its depletion calculations. However, the licensee is not using the constant soluble boron concentration in a manner consistent with the reference. Instead, the licensee is using a fuel assembly's lifetime burnup averaged soluble boron. Based on similar concerns discussed in Section 3.4.3.3.4 above, the NRC staff requested the licensee to justify the use of a constant average soluble boron concentration in its depletion calculation in order to address any potential non-conservatisms. In its June 6, 2019, letter, the licensee responded to the NRC staff's RAI (RAI 1) regarding its use of a fuel assembly's lifetime burnup averaged soluble boron. The licensee explicitly addressed legacy fuel, and where necessary, determined the margin between fuel assembly actual burnup and the minimum burnup requirements. The NRC staff finds the licensee's use of a fuel assembly's lifetime burnup averaged soluble boron acceptable for its legacy fuel. For current and future cycles, the licensee confirmed that the cycle average soluble boron will be less than the fuel assembly's lifetime burnup averaged soluble boron that was assumed in the application. Since the fuel assembly's lifetime burnup averaged soluble boron (assumed in the application) exceeds the cycle average soluble boron of every cycle the fuel assembly has experienced, the use of the fuel assembly's lifetime burnup averaged soluble boron is considered acceptable by the NRC staff.

3.4.3.3.5 Integral and Fixed Burnable Absorbers

In the past, Indian Point 2 used the Westinghouse Integral Fuel Burnable Absorber and two types of fixed burnable absorbers: Pyrex burnable absorbers and the Westinghouse Wet Annular Burnable Absorber. Indian Point 2 has used the Integral Fuel Burnable Absorber and Wet Annular Burnable Absorber at the same time. The modeling of integral and fixed burnable absorbers was included in the analyses for each "batch." The analysis indicates the modeling was conservative within each "batch."

3.4.3.3.6 Control Element Assembly Usage

If CEAs are present in assemblies for significant amounts of time in the reactor, the associated spectral hardening can increase plutonium generation, leading to higher fuel reactivity for the same burnup. The modeling of CEAs was included in the analysis for each "batch." The analysis indicates the modeling was conservative within each "batch." In its June 6, 2019, letter, the licensee sufficiently responded to the NRC staff's RAI regarding its modeling of CEA usage. With the clarifications provided in its RAI response, the NRC staff considers the licensee's modeling of its CEA usage to be acceptable.

3.4.3.3.7 Credited Nuclides

Table 2.1 of the NCS analysis report provides a list of isotopes used in the Indian Point 2 analysis. Sections 2.1 and 5.8 of the same report discuss changes made to the isotopic inventories generated during the depletion analysis to account for decay of poisons and gaseous and volatile fission products. The credited nuclides are acceptable.

3.4.3.4 Eccentricity of the Fuel Within the Storage Cell

NET-28091-0003-01, Revision 0, Section 7.3, "Eccentricity," describes the analysis the licensee performed to consider the eccentricity of position of fuel assemblies within a storage cell. The base k-effective calculation models all fuel assemblies in the center of their respective storage cell. However, the fuel assemblies can be anywhere within their respective storage cell. The 'eccentricity' portion of the analysis is intended to determine the reactivity effect of the fuel assemblies being in other than the center of their storage cell. The licensee's analysis used an 8x8 storage cell array to calculate the eccentricity effect. The licensee's analysis considered 16 fuel assemblies to be eccentrically located, while maintaining the outer rows of the 8x8 array centrally located. Using periodic boundary conditions on the 8x8 array effectively models an infinite number of these 8x8 arrays. The boundary rows serve to minimize the nuclear interaction between them; therefore, the resultant k-effective is representative of a single 8x8 array. The licensee's analysis made two implicit assumptions: (1) there is only one eccentric configuration that yields the maximum reactivity, and (2) the fuel assembly can only be in one of four distinct locations within a storage cell. With those assumptions, the licensee calculated the probability of occurrence of the worst-case eccentric positioning configuration to be approximately 2.3E-10. The staff noted that the inverse to this value is equivalent to the number of possible combinations of eccentrically located fuel assemblies (or approximately 4.35E9) and requested the licensee (in RAI 12) to provide a justification for the combination used to calculate the reactivity of the eccentricity effect.

In response to the NRC staff's RAI, the licensee provided additional information regarding the eccentric positioning of fuel assemblies in storage cells. The licensee performed several additional cases for eccentric positioned fuel. All the new cases moved the fuel assemblies closer together. The licensee theorizes that moving the fuel assemblies closest to the center of the array would increase reactivity. The additional cases for Region 1 appear to support that theory, as moving the fuel assemblies closer together increases K_{eff}. The licensee did find a slightly higher reactivity than its original analysis found when using a slightly different combination. The cases for Region 2 do not appear to support the same theory, as K_{eff} decreases as the fuel assemblies are moved closer to the center of the array. At its essence, the eccentric position consideration is one of optimum moderation. Therefore, the NRC staff looked at the licensee's SFP temperature analysis. In Section 3.4.2.1 above, the staff noted that Region 1 had monotonically increasing reactivity with increasing temperature. This supports the licensee's theory that moving the fuel assemblies closer together would increase reactivity. For Region 2, the reactivity did not monotonically change with temperature; instead, there is an optimum moderation temperature indicating optimal eccentric positioning configurations for Region 2, which increases reactivity, supporting the licensee's prior mentioned analysis. The Region 2 temperature analysis indicated the optimum moderation effect is small. The NRC staff considers the reactivity effect of an optimum eccentric positioning in Region 2 to be appropriate within the margin of the analysis. Based on the above, the staff finds that the licensee's analysis to consider the eccentricity of position of fuel assemblies within a storage cell is acceptable.

3.4.4 Non-Standard Fuel Configurations/Reconstituted Fuel

The Indian Point 2 SFP has several non-standard fuel configurations: Failed Fuel Containers, Fuel Rod Storage Baskets, and one assembly with one empty lattice location. The licensee evaluated these items and pre-determined their storage requirement. The licensee identified a penalty to be added to any future fuel assemblies with empty lattice locations.

The process of reconstituting a fuel assembly is addressed in Section 9.1, "Normal Operations," in the licensee's NCS analysis. It states that any fuel assembly being reconstituted must be isolated from the rest of the fuel stored in the SFP. The licensee's proposed control of non-standard fuel configurations and reconstituted fuel is acceptable.

3.4.5 Determination of Soluble Boron Requirements

Section 50.68 of 10 CFR requires that the K_{eff} of the Indian Point 2 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. This requirement applies to all normal and abnormal/accident conditions.

The licensee's analysis for the normal static condition determined that 700 parts per million of soluble boron in the SFP was sufficient to meet the K_{eff} requirement, with more than 0.05 ΔK_{eff} margin. This was done using the biases and uncertainties from the unborated cases. Because of the large margin, the NRC staff finds this acceptable.

The licensee considered several potential accident scenarios. The licensee found the multiple misloading of fuel assemblies to be limiting. The licensee considered only one of numerous possible misloading configurations. In response to the NRC staff's RAI, the licensee responded that there are visual cues to the fuel handlers that would allow them to detect errors in their fuel movement instructions. Additionally, since the licensee is shutting down both Indian Point 2 and 3, and there will no longer be fresh fuel on-site, the potential for the misloading of fresh fuel is nil. Therefore, the NRC staff finds the accident treatment acceptable.

3.5 Conclusion

The NRC staff has completed its review of the Indian Point 2 SFP NCS analyses, which are documented in the licensee's LAR and updated through the June 6, 2019, supplement. Based on the above evaluation, the NRC staff concludes that there is reasonable assurance that the Indian Point 2 SFP meets the applicable regulatory requirements in 10 CFR 50.68.

Additionally, the NRC staff determined that the proposed TSs will continue to be based on the analyses and evaluations included in the safety analysis report and amendments thereto in accordance with 10 CFR 50.36(b). The NRC staff also determined that the proposed TSs will continue to include required design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, in accordance with 10 CFR 50.36(c)(4). Therefore, the NRC staff has determined that the proposed TSs are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on July 26, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (83 FR 10916). Accordingly, the amendment meets 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 amendment.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: K. Wood

Date: September 4, 2019

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 290 RE: STORAGE OF FRESH AND SPENT NUCLEAR FUEL IN THE SPENT FUEL POOL (EPID L-2017-LLA-0408) DATED SEPTEMBER 4, 2019

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