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

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July 13, 2012

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: INTER-UNIT SPENT FUEL TRANSFER (TAC NOS. ME1671, ME1672, AND L24299)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 268 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2) and Amendment No. 246 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendments consist of changes to the licenses and Technical Specifications (TSs) in response to your application dated July 8, 2009, as supplemented by letters dated September 28, 2009, October 26, 2009, October 5, 2010, October 28, 2010, July 28, 2011, August 23, 2011, October 28, 2011, December 15, 2011, January 11, 2012, March 2, 2012, April 23, 2012, and May 7, 2012.

The amendments revise the licenses and TSs by authorizing the transfer of spent fuel from the IP3 spent fuel pool to the IP2 spent fuel pool, using a newly-designed shielded transfer canister, for further transfer to the on-site Independent Spent Fuel Storage Installation.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has also prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed.

Document transmitted herewith contains sensitive unclassified information. When separated from enclosure 4, this document is decontrolled.

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

Sincerely,

John P. Boska

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. Amendment No. 268 to DPR-26
2. Amendment No. 246 to DPR-64
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

cc w/o encl 4: 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. 268
License No. DPR-26

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (ENO or the licensee) dated July 8, 2009, as supplemented on September 28, 2009, October 26, 2009, October 5, 2010, October 28, 2010, July 28, 2011, August 23, 2011, October 28, 2011, December 15, 2011, January 11, 2012, March 2, 2012, April 23, 2012, and May 7, 2012, 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. 268, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 13, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 268

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following pages of the License 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	3
5a	5a
8	8

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 areas of change.

Remove Pages	Insert Pages
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

Insert the following pages of the new Appendix C Technical Specifications. The pages are identified by amendment number.

Insert Pages
Title Page, Part I
1
2
Title Page, Part II
i
1.1-1
1.1-2
1.2-1
1.2-2
1.2-3
1.3-1
1.3-2
1.3-3
1.3-4
1.4-1
1.4-2
1.4-3
2.0-1
3.0-1

Insert the following pages of the new Appendix C Technical Specifications. The pages are identified by amendment number.

<u>Insert Pages</u>
3.0-2
3.1.1-1
3.1.2-1
3.1.2-2
3.1.2-3
3.1.2-4
3.1.2-5
3.1.2-6
3.1.3-1
3.1.4-1
3.1.4-2
3.1.5-1
4.0-1
4.0-2
4.0-3
4.0-4
4.0-5
5.0-1
5.0-2
5.0-3
5.0-4

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 and Indian Point Nuclear Generating Unit No. 3 (IP3). 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; 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 10-27-2004

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 268, 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 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.

O. Control Room Envelope Habitability

Upon implementation of Amendment No. 258 adopting TSTF-448, Revision 3 (as supplemented), the determination of control room envelope (CRE) unfiltered air inleakage as required by Technical Specification (TS) Surveillance Requirement (SR) 3.7.10.4, in accordance with TS 5.5.16.c.(i), the assessment of CRE habitability as required by TS 5.5.16.c.(ii), and the measurement of CRE pressure as required by TS 5.5.16.d, shall be considered met. Following implementation:

(a) The first performance of SR 3.7.10.4, in accordance with TS 5.5.16.c.(i), shall be within the next 18 months since the time period since the most recent successful tracer gas test is greater than 6 years.
(b) The first performance of the periodic assessment of CRE habitability, TS 5.5.16.c.(ii), shall be within the next 9 months since the time period since the most recent successful tracer gas test is greater than 3 years.
(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.16.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from January 4, 2007, the date of the most recent successful pressure measurement test.

P. ENO may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. ENO is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by ENO and Entergy Nuclear Indian Point 3, LLC.

3. On the closing date of the transfer of the license, Con Edison shall transfer to ENIP2 all of the accumulated decommissioning trust funds for IP2 and such additional funds to be deposited in the decommissioning trusts for IP2 such that the total amount transferred for Indian Point Nuclear Generating Unit No. 1 (IP1) and IP2 is no less than $430,000,000. Furthermore, ENIP2 shall either (a) establish a provisional trust for decommissioning funding assurance for IP1 and IP2 in an amount no less than $25,000,000 (to be updated as required under applicable NRC regulations, unless otherwise approved by the NRC) or (b) obtain a surety bond for an amount no less than $25,000,000 (to be updated as required under applicable NRC regulations, unless otherwise approved by the NRC). The total decommissioning funding assurance provided for IP2 by the combination of the decommissioning trust and the provisional trust or surety bond at the time of transfer of the licenses shall be at a level no less than the amounts calculated pursuant to, and required under, 10 CFR 50.75. The decommissioning trust, provisional trust, and surety bond shall be subject to or be consistent with the following requirements, as applicable:

6. This amended license is effective as of the date of issuance, and shall expire at midnight September 28, 2013. Amdt. 118 4-21-87

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Roger S. Boyd

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Technical Specification Requirements
Appendix C – Inter-Unit Fuel Transfer Technical Specifications

Date of Issuance: September 28, 1973

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 $\geq$ 3.2 and $\leq$ 4.4 w/o U^{235} and discharged prior to IP3 Cycle 12 shall be stored in the Spent Fuel Pit.

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

ACTIONS

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 classified in accordance with Figure 3.7.13-1, Figure 3.7.13-2, Figure 3.7.13-3, or Figure 3.7.13-4 and meets the requirements for the intended storage location.	Prior to storing the fuel assembly in the Spent Fuel Pit.
	<u>OR</u>	
	Verify by administrative means that the IP3 fuel assembly meets the requirements for the intended storage location.	Prior to storing the fuel assembly in the Spent Fuel Pit.

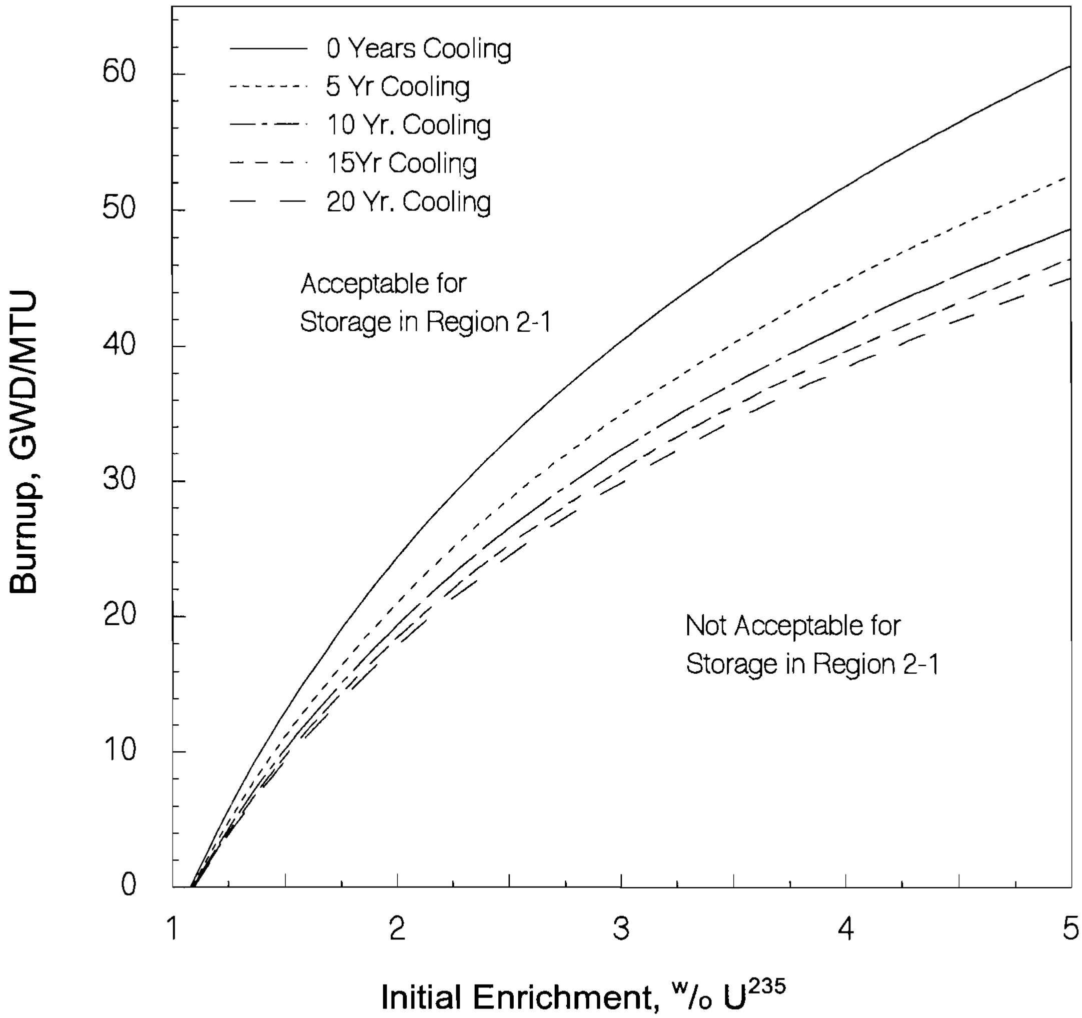

Figure 3.7.13-1
IP2 Fuel Assembly Limiting Burnup and Cooling Time versus Initial Enrichment:
Acceptable for Storage in Any Location in
Region 2-1, Region 2-2, Region 1-2 or Region 1-1

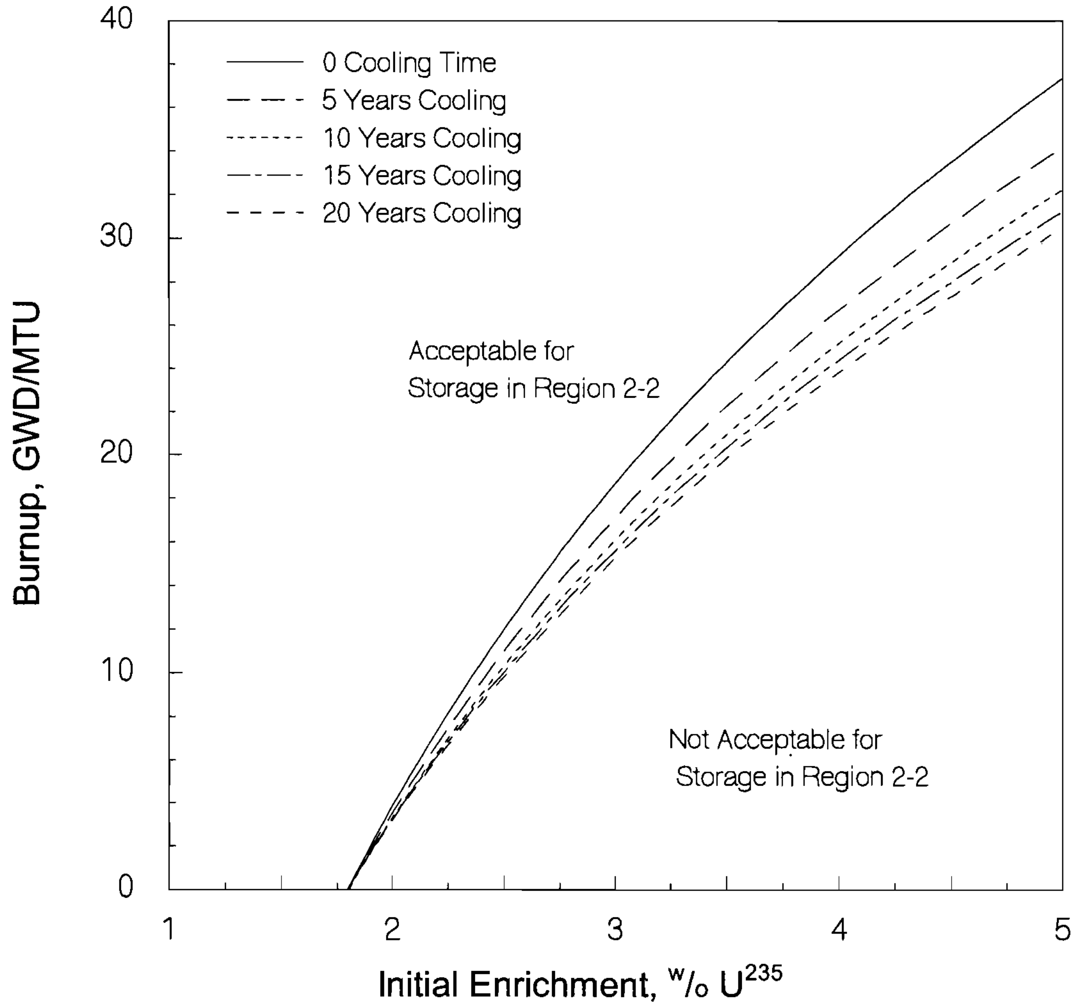

Figure 3.7.13-2
IP2 Fuel Assembly Limiting Burnup and Cooling Time versus Initial Enrichment:
Acceptable for Storage in Any Location in
Region 2-2, Region 1-2 or Region 1-1

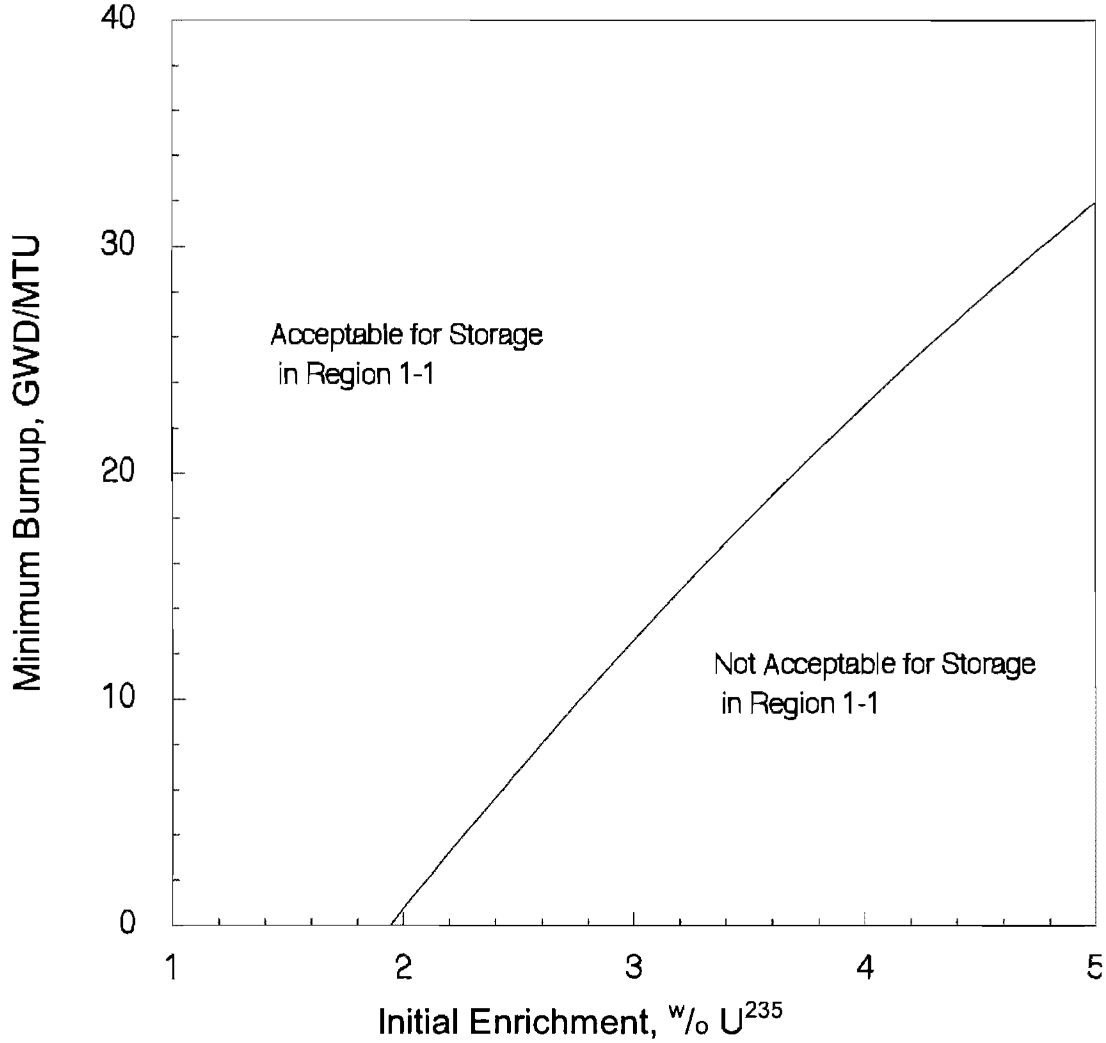

Figure 3.7.13-3
IP2 Fuel Assembly Limiting Burnup versus Initial Enrichment:
Acceptable for Storage in Any Location in Region 1-2, Region 1-1,
or in locations designated as "peripheral" cells in Region 2-2.

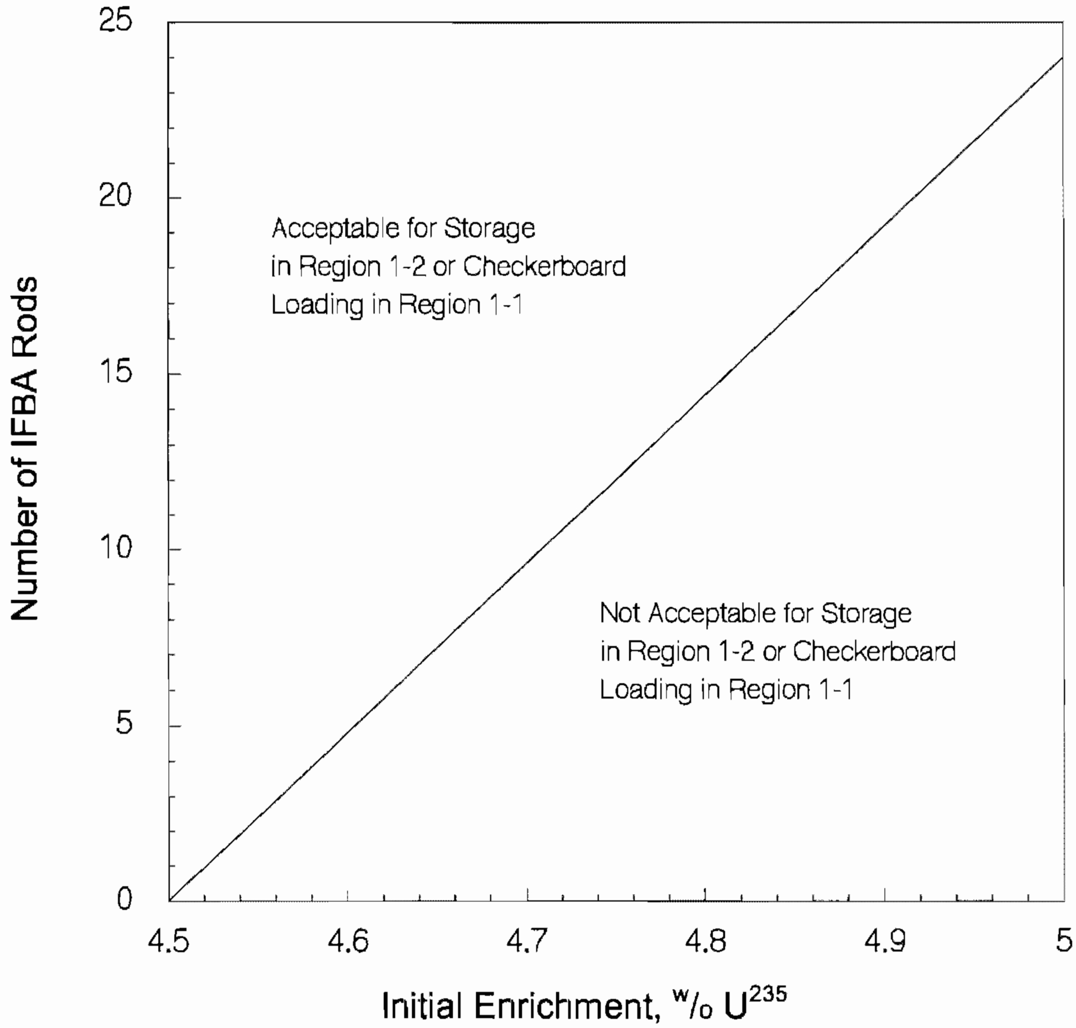

Figure 3.7.13-4
IP2 Fuel Assembly Minimum number of IFBA rods versus Initial Enrichment:

1) Acceptable for Storage in Any Location in Region 1-2, or
2) Acceptable for Storage in a checkerboard loading configuration in Region 1-1, or
3) Acceptable for Storage in locations designated as "peripheral" cells in Region 2-2.

APPENDIX C

TO

FACILITY OPERATING LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 2, LLC (ENIP2)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 2

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM

FACILITY LICENSE NO. DPR-26

DOCKET NO. 50-247

SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM

1.0 DESCRIPTION

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

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

The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base-plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are 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.

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 ⅞ inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI-TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI-TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.

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

2.0 CONDITIONS

2.1 OPERATING PROCEDURES

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

2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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

2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

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

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

APPENDIX C

TO

FACILITY OPERATING LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 2, LLC (ENIP2)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 2

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART II: TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-26

DOCKET NO. 50-247

TABLE OF CONTENTS

1.0 USE AND APPLICATION

1.1 Definitions

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

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

(continued)

1.1 Definitions (continued)

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 VERIFY . . . AND A.2 Restore . . .	

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

(continued)

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

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

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each component.

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

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

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

IMMEDIATE COMPLETION TIME

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

1.0 USE AND APPLICATION

1.4 Frequency

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

(continued)

1.4 Frequency (continued)

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

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

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

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

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

2.0 NOT USED

This section is intentionally left blank

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

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

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an STC.

3.1 INTER-UNIT FUEL TRANSFER

3.1.1 Boron Concentration

LCO 3.1.1 The boron concentration of the water in the Spent Fuel Pit and the STC shall be ≥ 2000 ppm.

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

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

---

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

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 ≤ 650 Watts (cells 5 through 12);
3. Decay heat including NON FUEL HARDWARE ≤ 1105 Watts (cell 1, 2, 3 or 4);
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.

---

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 ≤ 650 Watts;
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 ≥ 3.2 and ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12 shall be placed in the STC basket.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

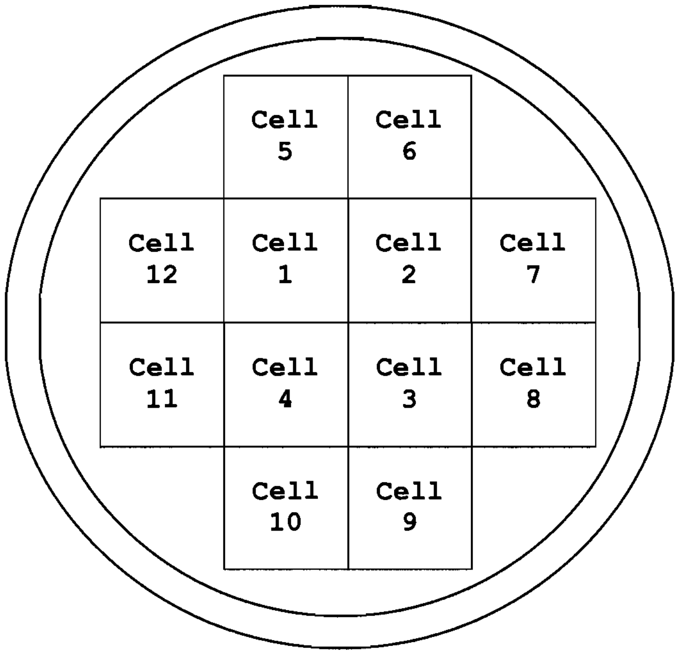

Figure 3.1.2-1
Shielded Transfer Canister Layout
(Top View)

Table 3.1.2-1
Minimum Burnup Requirements at Varying Initial Enrichments[a]

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

(a) Fuel that does not meet the minimum assembly average burnup at a given initial enrichment is classified as Type 1 fuel. Fuel that meets the minimum assembly average burnup at a given initial enrichment is classified as Type 2 fuel.

(b) Linear interpolation between enrichment levels to determine minimum burnup requirements is permitted.

(c) Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation or where it can be shown that the insertion did not exceed 8 inches below the top of the active fuel.

(d) Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation and where the insertion was more than 8 inches below the top of the active fuel. This configuration also applies to fuel assemblies that have contained a Hafnium Flux Suppressor.

(e) Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel.

(f) Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the maximum allowed initial average enrichment.

(g) Rounding to one decimal place to determine initial enrichment is not permitted.

Table 3.1.2-2

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

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs[b]	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.

Table 3.1.2-3

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

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

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

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.3 Shielded Transfer Canister (STC) Initial Water Level

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STC water level not within limit.	-------------------NOTE---------------- Water used for level restoration must meet the boron concentration requirement of LCO 3.1.1. -- A.1 Initiate action to restore STC water level.	Immediately

SURVEILLANCE REQUIREMENTS

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.4 Shielded Transfer Canister (STC) Pressure Rise

LCO 3.1.4 The pressure rise in the STC cavity shall be ≤ 0.2 psi/hr averaged over a rolling 4 hour period.

APPLICABILITY: Over a 24 hour period after successful completion of LCO 3.1.3 and prior to TRANSFER OPERATIONS when the STC is in the HI-TRAC and the STC lid has been installed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Rate of STC cavity pressure rise not within limit.	A.1.1 Establish a vent path on the STC. <u>AND</u> -------------NOTE------------- Water used for recirculation must meet the boron concentration requirement of LCO 3.1.1. ----------------------------------- A.1.2 Begin circulation of borated water in the STC to establish and maintain the STC water exit temperature < 180°F. <u>AND</u> A.1.3 Begin actions to determine the reason for exceeding the pressure rise limit.	Immediately

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.5 Shielded Transfer Canister (STC) Unloading

--NOTE--

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

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

4.0 DESIGN FEATURES

4.1 Inter-Unit Fuel Transfer

4.1.1 Fuel Assemblies

Fuel assemblies selected for inter-unit transfer of fuel shall meet the fuel characteristics specified in Table 4.1.1-1.

4.1.2 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. k_{eff} ≤ 0.95 if fully flooded with unborated water;

c. A nominal 9.218 inch center-to-center distance between fuel assemblies placed in the STC basket;

d. Basket cell ID: 8.79 in. (nominal);

e. Basket cell wall thickness: 0.28 in. (nominal);

f. B_4C in the Metamic neutron absorber: ≥ 31.5 wt.% and ≤ 33.0 wt.%;

g. The B_4C in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%;

h. Metamic panel thickness: ≥ 0.102 in.;

i. The size and location of the neutron absorber panels shall be in accordance with drawing 6015, revision 6, which can be found in the Licensing Report (Holtec International Report HI-2094289).

4.1.2.2 Drainage

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

4.1.2.3 Capacity

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

(continued)

4.0 DESIGN FEATURES (continued)

4.1.3 Codes and Standards

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

4.1.4 Geometric Arrangements and Process Variables

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

1. LOADING OPERATIONS, TRANSFER OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures ≥ 0°F.

2. LOADING OPERATIONS shall only be conducted when the spent fuel pit water temperature and the fuel handling building ambient temperatures are both ≤ 100°F.

3. LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel assemblies.

4. LOADING OPERATIONS shall only be conducted when the irradiated fuel assemblies in the IP3 spent fuel pit have been subcritical for at least 90 days.

5. TRANSFER OPERATIONS shall only be conducted when the outside air temperature is ≤ 100°F.

6. TRANSFER OPERATIONS shall only be conducted when the STC trunnions are offset from the HI-TRAC trunnions in the azimuthal direction by at least 30 degrees.

7. TRANSFER OPERATIONS shall only be conducted after STC seal leak tests have demonstrated no detected leakage when tested to a sensitivity of $1x10^{-3}$ ref-cm^3/s in accordance with the "pre-shipment" test requirements of ANSI N14.5.

8. Prior to installing the HI-TRAC lid the HI-TRAC water level shall be verified by two separate inspections to be within +0/-1 inch of the top of the STC lid.

(continued)

4.0 DESIGN FEATURES (continued)

9. TRANSFER OPERATIONS shall only be conducted after the combined leak rate through the HI-TRAC top lid and vent port cover seals are confirmed to be water tight using an acceptable leak test from ANSI N14.5 and the pool lid seal is verified to be water tight by visual inspection.

10. TRANSFER OPERATIONS shall not occur with a TRANSPORTER that contains > 50 gallons of diesel fuel.

Table 4.1.1-1

Fuel Assembly Characteristics

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

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

(b) Applicable only if axial blankets are present.

(continued)

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 1 of 2)

List of ASME Code Alternatives for the STC

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

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 2 of 2)

List of ASME Code Alternatives for the STC

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

(a) Holtec International Report HI-2094289

5.0 PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Transport Evaluation Program

a. For lifting of the loaded STC or loaded HI-TRAC using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.

b. This program is not applicable when the loaded HI-TRAC is in the fuel building or is being handled by equipment providing support from underneath (e.g., on air pads).

c. The loaded HI-TRAC may be lifted to any height necessary during TRANSFER OPERATIONS provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.

1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.

3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.

d. The lift height of the loaded HI-TRAC above the transport route surface or other supporting surface shall be limited to 6 inches, except as provided in Specification 5.1.c.

5.2 Metamic Coupon Sampling Program

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

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

The following specifications apply:

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

(continued)

5.0 PROGRAMS (continued)

(ii) Pre-characterization testing: Before installation, each coupon will be measured and weighed. The measurements shall be taken at locations pre-specified in the test program. Each coupon shall be tested by neutron attenuation before installation in the STC. The weight, length, width, thickness, and results of the neutron attenuation testing shall be documented and retained.

(iii) Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained.

(iv) The coupons shall be examined for any indication of swelling, delamination, edge degradation, or general corrosion. The results of the examination shall be documented and retained.

(v) The coupons shall be tested by neutron attenuation and the results compared with the pre-characterization testing data. The results of the testing shall be documented and retained. Results are acceptable if the measured value is within +/-2.5% of the value measured for the same coupon at manufacturing.

(vi) The coupons shall be returned to their locations in the STC unless anomalous material behavior is found. If the results indicate anomalous material behavior, evaluation and corrective actions shall be pursued.

5.3 Technical Specifications (TS) Bases Control Program

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

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or
2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that do not meet the criteria of Specification 5.3.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.0 PROGRAMS (continued)

5.4 Radiation Protection Program

5.4.1 The radiation protection program shall appropriately address STC loading and unloading conditions, including transfer of the loaded TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.

5.4.2 Total (neutron plus gamma) measured dose rates shall not exceed the following:

a. 1400 mrem/hr on the top of the STC (with lid in place).

b. 5 mrem/hr on the side of the TRANSFER CASK

5.4.3 The STC and TRANSFER CASK surface neutron and gamma dose rates shall be measured as described in Section 5.4.6 for comparison against the limits established in Section 5.4.2.

5.4.4 If the measured surface dose rates exceed the limits established in Section 5.4.2, then:

a. Administratively verify that the correct contents were loaded in the correct fuel basket cell locations.

b. Perform a written evaluation to determine whether TRANSFER OPERATIONS can proceed without exceeding the dose limits of 10 CFR 72.104 or 10 CFR 20.1301.

5.4.5 If the verification and evaluation performed pursuant to Section 5.4.4 show that the fuel is loaded correctly and the dose rates from the STC and TRANSFER CASK will not cause the dose limits of 10 CFR 72.104 or 10 CFR 20.1301 to be exceeded, TRANSFER OPERATIONS may occur. Otherwise, TRANSFER OPERATIONS shall not occur until appropriate corrective action is taken to ensure the dose limits are not exceeded.

5.4.6 STC and TRANSFER CASK surface dose rates shall be measured at approximately the following locations:

a. The dose rate measurement shall be taken at the approximate center of the STC top lid. Two (2) additional measurements shall be taken on the STC lid approximately 180 degrees apart and 12 to 18 inches from the center of the lid, avoiding the areas around the inlet and outlet ports. The measurements must be taken when the STC is in the HI-TRAC after the steam space is established and prior to HI-TRAC lid installation.

(continued)

5.0 PROGRAMS (continued)

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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. 246
License No. DPR-64

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (ENO or the licensee) dated July 8, 2009, as supplemented on September 28, 2009, October 26, 2009, October 5, 2010, October 28, 2010, July 28, 2011, August 23, 2011, October 28, 2011, December 15, 2011, January 11, 2012, March 2, 2012, April 23, 2012, and May 7, 2012, 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, B, and C, as revised through Amendment No. 246, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 13, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 246

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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

Remove Page	Insert Page
3	3
9	9

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 Page	Insert Page
3.7.15-1	3.7.15-1

Insert the following pages of the new Appendix C Technical Specifications. The pages are identified by amendment number.

Insert Pages
Title Page, Part I
1
2
Title Page, Part II
i
1.1-1
1.1-2
1.2-1
1.2-2
1.2-3
1.3-1
1.3-2
1.3-3
1.3-4
1.4-1
1.4-2
1.4-3
2.0-1
3.0-1
3.0-2
3.1.1-1
3.1.2-1

Insert the following pages of the new Appendix C Technical Specifications. The pages are identified by amendment number.

<u>Insert Pages</u>
3.1.2-2
3.1.2-3
3.1.2-4
3.1.2-5
3.1.2-6
3.1.3-1
3.1.4-1
3.1.4-2
3.1.5-1
4.0-1
4.0-2
4.0-3
4.0-4
4.0-5
5.0-1
5.0-2
5.0-3
5.0-4

(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 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 (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 246 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.

AE. ENO may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. ENO is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by ENO and ENIP3.

3. This amended license is effective at 12:01 a.m., November 21, 2000, and shall expire at midnight December 12, 2015.

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Technical Specification Requirements
Appendix C – Inter-Unit Fuel Transfer Technical Specifications

Date of Issuance: March 8, 1978

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pit Boron Concentration

LCO 3.7.15 The Spent Fuel Pit boron concentration shall be ≥ 1000 ppm.

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit boron concentration not within limit.	-----------NOTE---------- LCO 3.0.3 is not applicable. ----------------------------	
	A.1 Suspend movement of fuel assemblies in the spent fuel pit.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pit boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel pit verification.	Immediately

APPENDIX C

TO

FACILITY OPERATING LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM

1.0 DESCRIPTION

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

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

The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base-plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are 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.

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 ⅞ inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI-TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI-TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.

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

2.0 CONDITIONS

2.1 OPERATING PROCEDURES

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

2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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

2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

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

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

APPENDIX C

TO

FACILITY OPERATING LICENSE

FOR

ENTERGY NUCLEAR INDIAN POINT 3, LLC (ENIP3)

AND

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART II: TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

TABLE OF CONTENTS

1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

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

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

(continued)

1.1 Definitions (continued)

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 VERIFY . . . AND A.2 Restore . . .	

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

(continued)

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

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

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each component.

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

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

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

IMMEDIATE COMPLETION TIME

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

1.0 USE AND APPLICATION

1.4 Frequency

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

(continued)

1.4 Frequency (continued)

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

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

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

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

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

2.0 NOT USED

This section is intentionally left blank

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

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

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an STC.

3.1 INTER-UNIT FUEL TRANSFER

3.1.1 Boron Concentration

LCO 3.1.1 The boron concentration of the water in the Spent Fuel Pit and the STC shall be ≥ 2000 ppm.

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

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 ≤ 650 Watts (cells 5 through 12);
3. Decay heat including NON FUEL HARDWARE ≤ 1105 Watts (cell 1, 2, 3 or 4);
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.

---

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 ≤ 650 Watts;
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 ≥ 3.2 and ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12 shall be placed in the STC basket.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

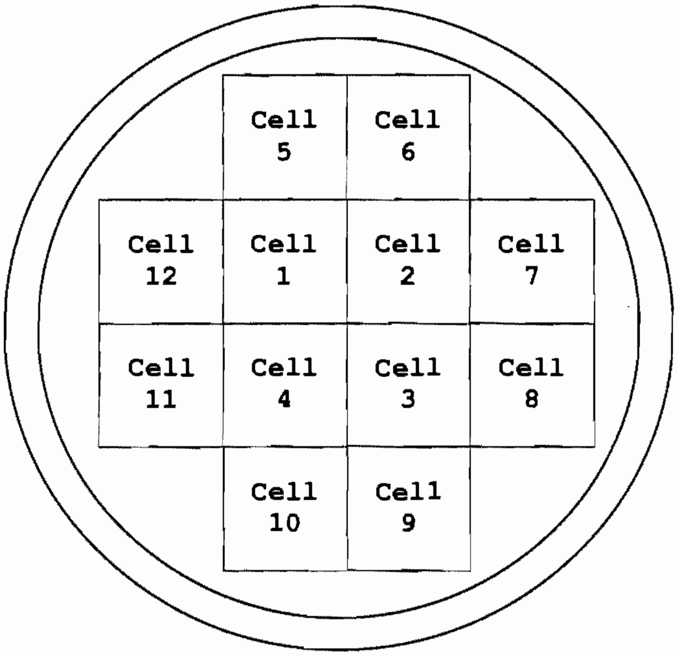

Figure 3.1.2-1
Shielded Transfer Canister Layout
(Top View)

Table 3.1.2-1
Minimum Burnup Requirements at Varying Initial Enrichments[(a)]

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

(a) Fuel that does not meet the minimum assembly average burnup at a given initial enrichment is classified as Type 1 fuel. Fuel that meets the minimum assembly average burnup at a given initial enrichment is classified as Type 2 fuel.

(b) Linear interpolation between enrichment levels to determine minimum burnup requirements is permitted.

(c) Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation or where it can be shown that the insertion did not exceed 8 inches below the top of the active fuel.

(d) Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation and where the insertion was more than 8 inches below the top of the active fuel. This configuration also applies to fuel assemblies that have contained a Hafnium Flux Suppressor.

(e) Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel.

(f) Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the maximum allowed initial average enrichment.

(g) Rounding to one decimal place to determine initial enrichment is not permitted.

Table 3.1.2-2

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

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs[b]	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.

Table 3.1.2-3

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

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

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

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.3 Shielded Transfer Canister (STC) Initial Water Level

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STC water level not within limit.	----------------NOTE-------------- Water used for level restoration must meet the boron concentration requirement of LCO 3.1.1. -- A.1 Initiate action to restore STC water level.	Immediately

SURVEILLANCE REQUIREMENTS

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.4 Shielded Transfer Canister (STC) Pressure Rise

LCO 3.1.4 The pressure rise in the STC cavity shall be ≤ 0.2 psi/hr averaged over a rolling 4 hour period.

APPLICABILITY: Over a 24 hour period after successful completion of LCO 3.1.3 and prior to TRANSFER OPERATIONS when the STC is in the HI-TRAC and the STC lid has been installed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Rate of STC cavity pressure rise not within limit.	A.1.1 Establish a vent path on the STC. AND ------------NOTE------------- Water used for recirculation must meet the boron concentration requirement of LCO 3.1.1. -------------------------------- A.1.2 Begin circulation of borated water in the STC to establish and maintain the STC water exit temperature < 180°F. AND A.1.3 Begin actions to determine the reason for exceeding the pressure rise limit.	Immediately
		(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

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

3.1 INTER-UNIT FUEL TRANSFER

3.1.5 Shielded Transfer Canister (STC) Unloading

-----NOTE-----

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

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

4.0 DESIGN FEATURES

4.1 Inter-Unit Fuel Transfer

4.1.1 Fuel Assemblies

Fuel assemblies selected for inter-unit transfer of fuel shall meet the fuel characteristics specified in Table 4.1.1-1.

4.1.2 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

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

c. A nominal 9.218 inch center-to-center distance between fuel assemblies placed in the STC basket;

d. Basket cell ID: 8.79 in. (nominal);

e. Basket cell wall thickness: 0.28 in. (nominal);

f. B_4C in the Metamic neutron absorber: ≥ 31.5 wt.% and ≤ 33.0 wt.%;

g. The B_4C in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%;

h. Metamic panel thickness: ≥ 0.102 in.;

i. The size and location of the neutron absorber panels shall be in accordance with drawing 6015, revision 6, which can be found in the Licensing Report (Holtec International Report HI-2094289).

4.1.2.2 Drainage

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

4.1.2.3 Capacity

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

(continued)

4.0 DESIGN FEATURES (continued)

4.1.3 Codes and Standards

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

4.1.4 Geometric Arrangements and Process Variables

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

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

(continued)

4.0 DESIGN FEATURES (continued)

9. TRANSFER OPERATIONS shall only be conducted after the combined leak rate through the HI-TRAC top lid and vent port cover seals are confirmed to be water tight using an acceptable leak test from ANSI N14.5 and the pool lid seal is verified to be water tight by visual inspection.

10. TRANSFER OPERATIONS shall not occur with a TRANSPORTER that contains > 50 gallons of diesel fuel.

Table 4.1.1-1

Fuel Assembly Characteristics

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

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

(b) Applicable only if axial blankets are present.

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 1 of 2)

List of ASME Code Alternatives for the STC

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

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 2 of 2)

List of ASME Code Alternatives for the STC

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

(a) Holtec International Report HI-2094289

5.0 PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Transport Evaluation Program

a. For lifting of the loaded STC or loaded HI-TRAC using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.

b. This program is not applicable when the loaded HI-TRAC is in the fuel building or is being handled by equipment providing support from underneath (e.g., on air pads).

c. The loaded HI-TRAC may be lifted to any height necessary during TRANSFER OPERATIONS provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.

   1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

   2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.

   3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.

d. The lift height of the loaded HI-TRAC above the transport route surface or other supporting surface shall be limited to 6 inches, except as provided in Specification 5.1.c.

5.2 Metamic Coupon Sampling Program

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

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

The following specifications apply:

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

(continued)

5.0 PROGRAMS (continued)

(ii) Pre-characterization testing: Before installation, each coupon will be measured and weighed. The measurements shall be taken at locations pre-specified in the test program. Each coupon shall be tested by neutron attenuation before installation in the STC. The weight, length, width, thickness, and results of the neutron attenuation testing shall be documented and retained.

(iii) Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained.

(iv) The coupons shall be examined for any indication of swelling, delamination, edge degradation, or general corrosion. The results of the examination shall be documented and retained.

(v) The coupons shall be tested by neutron attenuation and the results compared with the pre-characterization testing data. The results of the testing shall be documented and retained. Results are acceptable if the measured value is within +/-2.5% of the value measured for the same coupon at manufacturing.

(vi) The coupons shall be returned to their locations in the STC unless anomalous material behavior is found. If the results indicate anomalous material behavior, evaluation and corrective actions shall be pursued.

5.3 Technical Specifications (TS) Bases Control Program

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

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or
2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that do not meet the criteria of Specification 5.3.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.0 PROGRAMS (continued)

5.4 Radiation Protection Program

5.4.1 The radiation protection program shall appropriately address STC loading and unloading conditions, including transfer of the loaded TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.

5.4.2 Total (neutron plus gamma) measured dose rates shall not exceed the following:

a. 1400 mrem/hr on the top of the STC (with lid in place).

b. 5 mrem/hr on the side of the TRANSFER CASK

5.4.3 The STC and TRANSFER CASK surface neutron and gamma dose rates shall be measured as described in Section 5.4.6 for comparison against the limits established in Section 5.4.2.

5.4.4 If the measured surface dose rates exceed the limits established in Section 5.4.2, then:

a. Administratively verify that the correct contents were loaded in the correct fuel basket cell locations.

b. Perform a written evaluation to determine whether TRANSFER OPERATIONS can proceed without exceeding the dose limits of 10 CFR 72.104 or 10 CFR 20.1301.

5.4.5 If the verification and evaluation performed pursuant to Section 5.4.4 show that the fuel is loaded correctly and the dose rates from the STC and TRANSFER CASK will not cause the dose limits of 10 CFR 72.104 or 10 CFR 20.1301 to be exceeded, TRANSFER OPERATIONS may occur. Otherwise, TRANSFER OPERATIONS shall not occur until appropriate corrective action is taken to ensure the dose limits are not exceeded.

5.4.6 STC and TRANSFER CASK surface dose rates shall be measured at approximately the following locations:

a. The dose rate measurement shall be taken at the approximate center of the STC top lid. Two (2) additional measurements shall be taken on the STC lid approximately 180 degrees apart and 12 to 18 inches from the center of the lid, avoiding the areas around the inlet and outlet ports. The measurements must be taken when the STC is in the HI-TRAC after the steam space is established and prior to HI-TRAC lid installation.

(continued)

5.0 PROGRAMS (continued)

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 268 TO FACILITY OPERATING LICENSE NO. DPR-26

AND AMENDMENT NO. 246 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 July 8, 2009, Agencywide Documents Access and Management System (ADAMS) Accession No. ML091940176, as supplemented by letters dated September 28, 2009, ADAMS Accession No. ML092950437; October 26, 2009, ADAMS Accession No. ML093020080; October 5, 2010, ADAMS Accession No. ML102910511; October 28, 2010, ADAMS Accession Nos. ML103080112 and ML103080113; July 28, 2011, ADAMS Accession No. ML11220A079; August 23, 2011, ADAMS Accession Nos. ML11243A174, ML11243A175; and ML11243A220; October 28, 2011, ADAMS Accession No. ML11327A045 and ML11327A046; December 15, 2011, ADAMS Accession No. ML12013A259; January 11, 2012, ADAMS Accession No. ML120400604; March 2, 2012, ADAMS Accession No. ML12074A027; April 23, 2012, ADAMS Accession No. ML12129A457; and May 7, 2012, ADAMS Accession No. ML121370318; Entergy Nuclear Operations, Inc. (Entergy or the applicant) submitted license amendment requests (LARs) for changes to the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) Licenses and Technical Specifications (TSs) to the Nuclear Regulatory Commission (NRC). The supplements provided additional information that clarified the application but did not expand the scope of the application as originally noticed in the *Federal Register* (75 FR 3497). The proposed changes request NRC approval for the transfer of spent fuel from the IP3 spent fuel pool (SFP) to the IP2 SFP using a newly-designed shielded transfer canister, for further transfer to the on-site Independent Spent Fuel Storage Installation (ISFSI), which uses the Holtec HI-STORM 100 dry cask storage system. Note that at Indian Point the SFP may also be referred to as the spent fuel pit.

IP2 and IP3 were designed by the same architect/engineer and were built by the same construction company. Each unit has a separate fuel storage building (FSB) containing the unit's SFP. The storage capacities of the SFPs were maximized by replacing the original fuel storage racks with high-density racks in 1989 and 1990. The IP2 and IP3 FSBs were built with overhead cranes having a lift capacity of 40 tons. The original plan for spent fuel was to periodically transfer spent fuel in a transportation package to a fuel recycle facility. Later, the U.S. government sought to build a geological repository for spent fuel instead of pursuing recycling. When it became evident that neither fuel recycling nor the repository would be operational prior to the SFPs reaching their capacity limits, Entergy implemented an on-site ISFSI in another area of the Indian Point Energy Center (IPEC), which uses a dry cask storage system. This type of dry storage system required the use of a single-failure-proof handling system with a rated capacity of at least 100 tons for transfer of the fuel canister and shield cask in and out of the SFP. Entergy added a single-failure-proof gantry crane with this capacity to the IP2 FSB, by excavating to bedrock and supporting the crane foundation on bedrock. The applicant states that it would be much more difficult to add such a crane to the IP3 FSB for the following reasons: (1) Due to the IP3 FSB configuration the crane would have to be about 23 feet taller than the IP2 gantry crane, which would require a significant increase in the size of the crane structural members in order to withstand design loads, including potential seismic loads. Due to the limited space in the FSB, it may not be possible to fit in such a crane. (2) In the IP3 FSB there are numerous plant equipment interferences that would require significant design and construction effort to relocate. For these reasons, the applicant decided to request approval from the NRC to transfer IP3 spent fuel to the IP2 SFP. From there the applicant is able to move the IP3 spent fuel to the ISFSI.

The applicant considered using a spent fuel cask which was already licensed as a transportation package under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71. The applicant identified one cask which could be lifted by the existing IP3 crane, but it only had the capacity for a single fuel assembly. This would severely limit the rate of fuel transfer and would also increase the total radiation exposure to the workers involved with fuel movement. Entergy instead decided to proceed with designing a new transfer system, which could transfer multiple fuel assemblies in a newly-designed transfer canister. Entergy contracted for the design of a shielded transfer canister (STC), and determined that an STC with a capacity to hold up to 12 fuel assemblies in a borated water environment would be within the 40 ton rated capacity of the IP3 FSB crane. The proposed fuel transfer would involve the movement of a loaded STC a distance of about 300 yards between the IP3 FSB and the IP2 FSB using two major pieces of equipment that have been successfully used on-site to move loaded multipurpose canisters containing 32 spent fuel assemblies for the IP2 dry storage program. That equipment consists of (1) the HI-TRAC 100D Transfer Cask (HI-TRAC), which will be used to contain and shield the STC, and (2) the on-site vertical cask transporter (VCT) which will be used to move the HI-TRAC with the STC inside. Therefore, the STC is the only new piece of major equipment required for the inter-unit fuel transfer. The STC is placed in the IP3 SFP and loaded with spent fuel, then removed from the SFP and placed inside the HI-TRAC. The HI-TRAC is then transported using the VCT to the IP2 FSB, where the STC is placed in the IP2 SFP and the fuel is unloaded into the existing fuel storage racks for eventual transfer to the on-site ISFSI. The transport path is outdoors but is inside the plant security zone (the protected area).

2.0 REGULATORY EVALUATION

The newly-designed STC and the proposed fuel transfer process were submitted to the NRC for approval under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," as amendments to the IP2 and IP3 10 CFR Part 50 licenses. NRC regulations in 10 CFR 50.90 specify requirements for amendments to nuclear power plant operating licenses and state that applications for amendments shall follow the form prescribed for original applications. The requirements of 10 CFR 50.34, "Contents of applications; technical information," specify 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. There are no specific cask criteria in 10 CFR Part 50; as a result, the NRC staff is using evaluation criteria based on 10 CFR Part 50, Appendix A, "General

Design Criteria (GDC) for Nuclear Power Plants"; 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP); standards associated with 10 CFR Part 71 transportation package certification and 10 CFR Part 72 storage cask certification, and other standards. The NRC staff also used applicable portions of Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask"; NUREG-1536, Rev. 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility"; NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities"; and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," for guidance on fuel transfers within the protected area when the criteria in 10 CFR Part 50 did not provide sufficient details.

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 GDC apply to the design of this spent fuel transfer system:

- GDC 1, "Quality standards and records," specifies, in part, that structures, systems, and components 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 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 regulations as being directly applicable to the proposed amendments:

- 10 CFR 50.36, "Technical specifications"
- 10 CFR 50.68, "Criticality accident requirements"
- 10 CFR Part 73, "Physical protection of plants and materials"

In addition, 10 CFR 50.34 specifies that the safety analysis report include an analysis of the performance of SSCs with the objective of assessing the risk to public health and safety resulting from operation of the facility. This analysis shall include determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility. The analysis shall assess the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

Section 15.0, "Introduction - Transient and Accident Analyses," of NUREG-0800, "NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," 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 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 applicant's proposal to rely on a cask licensing report, "Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at the Indian Point Energy Center," HI-2094289, Reference 7, 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 applicant will reference this document in the UFSARs for IP2 and IP3. In this safety evaluation, HI-2094289 will be referred to as the Safety Analysis Report (SAR).

3.0 TECHNICAL EVALUATION

The technical evaluation is divided into the following sections, which also match with the SAR:

1. Introduction
2. Fuel Acceptance Criteria and Engineered Measures for Safety
3. Principal Design Criteria, Applicable Loads, and Service Life
4. Criticality Evaluation
5. Thermal-Hydraulic Evaluation
6. Structural Evaluation
7. Shielding Design and ALARA Considerations
8. Materials Evaluation, Acceptance Tests, and Maintenance Program

9. Economic and Environmental Considerations
10. Operating Procedures

3.1 Introduction

3.1.1 Background

IP2 and IP3 are pressurized-water reactors with a nuclear steam supply system designed by Westinghouse and are co-located in Buchanan, New York. IP2 received its operating license in 1973, and IP3 received its operating license in 1975. The SFPs were initially designed with low-density racks for the spent fuel, and were later upgraded to high density racks. The current SFP storage limit is 1374 fuel assemblies in IP2 and 1345 fuel assemblies in IP3.

In 2007, IP2 began transferring spent fuel from the IP2 SFP into dry cask storage at an ISFSI at the site. The dry cask storage system selected for use was the Holtec HI-STORM 100 System. As noted earlier, this also required the installation of a new gantry crane in the IP2 FSB, due to the limitations of the existing crane. Due to the difficulties noted previously related to the possible installation of a similar crane in the IP3 FSB, the applicant decided to instead request approval from the NRC to transfer the IP3 spent fuel to the IP2 SFP, using a newly-designed STC. From there it will be eventually transferred into dry cask storage. The applicant concluded that the original IP3 FSB 40 ton overhead crane could be upgraded to a 40 ton single-failure-proof crane, and the loading on the building structure will remain within the originally engineered limits. The applicant stated that the upgrade of the IP3 40 ton crane is not part of this license amendment request, and will be implemented pursuant to the provisions of 10 CFR 50.59, "Changes, tests and experiments."

3.1.2 Description of Equipment and Their Safety Functions

3.1.2.1 Shielded Transfer Canister (STC)

In the SAR, the applicant described the STC as a thick-walled cylindrical vessel with a welded base plate and a bolted top lid. The internal cavity space of the STC houses a fuel basket with twelve storage cells for transferring spent nuclear fuel assemblies.

The applicant stated in the SAR that the 2004 Edition of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) would govern the fabrication, testing, and inspection of the STC. The applicant 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 applicant also stated that the pressure boundary of the STC meets the stress limits of ASME Code, Section III, Class 3, Subsection ND with large margins, although the STC is not a code stamped vessel. The applicant listed the applicable design temperature and pressure for the STC in Tables 3.1.1 and 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 applicant 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.

The applicant stated that the loaded STC would be lifted using two special lifting devices: the STC lift lock and the STC lifting device, which consists of two identical assemblies. The application showed that the STC lift lock is attached to the top center of the STC lid using 4 bolts; the applicant stated that the lift lock serves as the attachment point for the overhead cranes at IP3 and IP2. The application showed that a pair of STC lifting devices would be bolted to the top surface of the STC lid at opposite locations on the lid perimeter. The applicant stated that each STC lifting device assembly has a pneumatically controlled lift arm that hangs below the STC lid and connects to an STC lifting trunnion. During a lift of a loaded STC, the weight of the loaded fuel and STC internals rests on the base of the STC. The load from the base of the STC travels through the STC body to the STC lifting trunnions, from the STC lifting trunnions to the STC lifting devices, from the STC lifting devices to the STC lid, from the STC lid to the STC lift lock, and finally from the STC lift lock to the overhead crane. The applicant stated that the special lifting devices (i.e., STC lift lock and STC lifting device) used to lift the STC meet the guidance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Section 5.1.6(1), and American National Standards Institute (ANSI) Standard N14.6-1993 for critical loads. The applicant also stated that the interfacing lift points (i.e., threaded bolt holes and STC lifting trunnions) were designed to meet the guidance of NUREG-0612, Section 5.1.6(3).

3.1.2.2 HI-TRAC 100D Transfer Cask

The applicant stated that the STC would be transported outside the IP2 and IP3 FSBs within the HI-TRAC 100D Transfer Cask (HI-TRAC), which is an existing piece of equipment already used at IP2 to transfer spent fuel from the IP2 SFP into dry storage. The HI-TRAC is certified for use under 10 CFR Part 72 and is used under the plant's 10 CFR Part 50 general license. The applicant could use either that transfer cask or a newly purchased HI-TRAC. The HI-TRAC has a cylindrical shell made of a layer of lead sandwiched between shells of carbon steel for gamma radiation shielding. There is an annulus, called a water jacket, between the cylindrical shell and an outer steel shell that can be filled with water for neutron shielding. The structural integrity is provided by the carbon steel in the cylindrical shell.

The HI-TRAC is designed to:

- Provide maximum shielding to the plant personnel engaged in conducting short-term operations pertaining to inter-unit spent fuel transfer.
- Provide protection to the STC and the contained nuclear fuel during short-term operations against loads resulting from extreme environmental phenomena, such as tornado missiles.
- Serve as the container equipped with the appropriate lifting devices in compliance with NUREG-0612, Section 5.1.6.(3) and ANSI N14.6 to lift, move, and handle the STC, as required, to perform the short-term operations.

The HI-TRAC is part of the HI-STORM 100 Dry Cask Storage System, which was certified for fuel storage applications under NRC Docket 72-1014. The design of the HI-TRAC shell used for this fuel transfer application consists of two major parts: (a) a multi-shell cylindrical cask body, and (b) a multi-plate bottom lid. These components are described in the HI-STORM 100 Final Safety Analysis Report. For the fuel transfer operation the HI-TRAC is fitted with the following additional components: an STC centering assembly inside, a solid top lid, and a bottom missile shield.

The specially designed HI-TRAC 100D solid top lid used for the inter-unit fuel transfer operation has an elastomeric seal to retain the water present in the STC/HI-TRAC annulus space. The applicant stated that the lid would be attached with multiple bolts to provide the necessary bolt pull to maintain joint integrity.

The HI-TRAC bottom lid is also fitted with an elastomeric seal that makes the cask a watertight container. A set of bolts that tap into the machined holes in the bottom lid provide the required physical strength to meet the structural requirements of ANSI N14.6 and to provide the necessary bolt pull to maintain joint integrity. The applicant also designed a bottom missile shield to protect the flanged bottom lid joint from potential tornado missile loads.

The HI-TRAC was originally designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3. However, the original HI-TRAC had no design pressure load because the original top lid design included a large circular opening. The HI-TRAC used for fuel transfer will act as a pressure vessel, with a solid top lid. Therefore, the applicant evaluated the HI-TRAC for the effects of internal pressure using the stress limits of the ASME Code, Section III, Subsection ND. According to the ASME Code, Section ND-7000, pressure vessels are required to have overpressure protection; however the applicant provides no overpressure protection for the HI-TRAC once the lid is bolted down. For the fuel transfer operations, the function of the HI-TRAC includes retention of its contents under normal, off-normal, and accident conditions. The applicant evaluated the capability of the modified HI-TRAC design to withstand the maximum internal pressure associated with postulated accident conditions. Therefore, the applicant proposed operation without a pressure relief valve.

3.1.2.3 Vertical Cask Transporter (VCT)

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. In addition, special locking pins secure the load during movement providing redundant drop protection. A hydraulically-tightened safety strap secures the cask in the VCT and prevents rocking or swaying of the cask during movement. Finally, the VCT is equipped with speed governing features, which limits the travel speed to approximately 0.5 mph and prevents coasting on a loss of power condition, and a braking system with emergency stop which overrides all other controls and brings the VCT to a stop.

The VCT is too large to fit in the FSBs. In order to move the HI-TRAC into and out of the FSBs, additional equipment is required. Specifically, there is a low profile transporter (LPT) installed at the IP2 FSB, which is a platform mounted on steel rollers that ride on a hardened steel surface with steel guiderails. The VCT will place the HI-TRAC on the LPT, and the LPT will move the HI-TRAC into and out of the IP2 FSB. The LPT is also used when spent fuel is transferred from the IP2 SFP to the ISFSI. A platform supported by air pressure, known as an air pad, will be used at the IP3 FSB to move the HI-TRAC into and out of the IP3 FSB before being lifted by the VCT.

3.2 Fuel Acceptance Criteria and Engineered Measures for Safety

3.2.1 Fuel Design

The IP3 fuel is very similar to the IP2 fuel. The fuel assemblies are manufactured by Westinghouse, and have a 15 by 15 array of fuel rods and thimble tubes. The fuel assemblies are comprised of a top and bottom nozzle connected by 21 thimble tubes, with 204 small diameter fuel rods which are held in the fuel assembly by grid assemblies connected to the thimble tubes, resulting in a 15 by 15 array of rods. The fuel rods contain fuel pellets and are about 12 feet tall, while the overall fuel assembly is about 13 feet tall with a square cross section of about 8.5 inches per side. The 21 thimble tubes are empty cylinders. Twenty of the thimble tubes are called guide thimbles, and can be used to hold non-fuel hardware such as control rods, that are inserted from the top of the fuel assembly. The center thimble tube is an empty cylinder called an instrumentation thimble tube, into which an instrument thimble can be inserted from the bottom of the fuel assembly, allowing measurements of neutron flux using a moveable detector inside the instrument thimble.

3.2.2 Fuel Transfer Criteria

In order to be eligible for inter-unit transfer in accordance with this safety evaluation, the fuel must meet the following criteria:

a. The fuel must be intact as defined in TS 1.1, "Definitions."
b. The initial enrichment of the fuel assembly must meet the requirements of TS 3.1.2, "Shielded Transfer Canister (STC) Loading."
c. The maximum burnup of the fuel assembly and the minimum cooling time following discharge from the reactor vessel must meet the requirements of TS 3.1.2, "Shielded Transfer Canister (STC) Loading."
d. If the minimum burnup stated in TS 3.1.2, "Shielded Transfer Canister (STC) Loading," is not met, the fuel assembly can only be loaded into the 8 outer cells of the STC, and the 4 inner cells must remain empty.
e. If the fuel assembly contains non-fuel hardware in its thimble tubes, the non-fuel hardware must meet the requirements of TS 3.1.2, "Shielded Transfer Canister (STC) Loading."
f. The fuel assembly must meet the criteria in TS 4.1.1, "Fuel Assemblies."

3.2.3 Safety Measures

Protection of the fuel from a criticality accident is provided by neutron absorption and spatial separation. About 20% of natural boron is boron-10, which absorbs thermal neutrons. Panels with boron are attached to the walls of the fuel cells to reduce reactivity. Water in which boron has been dissolved (borated water) remains in the STC whenever fuel is present in order to reduce reactivity.

Protection against the release of radioactive material to the environment is provided by three independent barriers. The fuel cladding is the first barrier, the pressure-tested STC is the second barrier, and the sealed HI-TRAC is the third barrier. The materials used in the construction of these barriers have been extensively tested and have been approved by the NRC.

Protection against overpressurization is provided by gas-filled expansion volumes in the STC and HI-TRAC. The STC and HI-TRAC are not completely filled with water, and the expansion volume at the top of each cask accommodates the expansion of the water caused by an increase in the water temperature. The applicant has shown by analysis that natural heat transfer to the environment will prevent overheating of the water in the STC and HI-TRAC. The primary source of heat is decay heat from the fuel. Overheating of the STC and HI-TRAC is the only credible overpressurization mechanism. Although the fuel rods are internally pressurized with helium gas, there is no credible accident to cause large scale ruptures of the fuel rods to release the helium.

Protection against handling accidents is provided by the use of special equipment for the handling processes. The equipment used to lift and move the STC has been designed to meet the applicable stress limits with ample margins, in order to preclude a structural malfunction or failure.

3.3 Principal Design Criteria, Applicable Loads, and Service Life

3.3.1 Design Basis Loads

The STC design pressure is 50 pounds per square inch gauge (psig) and the accident pressure is 90 psig. There are no overpressure relief valves. Since the STC contains borated water, the temperature limits for the fuel are restricted by the saturation temperatures of water corresponding to the design pressures. For example, the saturation temperature of water which would produce a pressure of 90 psig is about 331 °F. Therefore, the fuel cladding temperature must remain below this temperature in order to prevent overpressurizing the STC. The NRC's limit for fuel cladding temperatures during short-term fuel transfer operations for dry cask storage is 752 °F (see NRC SFST-ISG-11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel"), so maintaining the water temperature low enough to prevent overpressurization will also maintain the fuel cladding temperature low enough to prevent overheating of the fuel rods. There is no credible accident that results in the loss of water from the STC.

The HI-TRAC design pressure is 30 psig and the accident pressure is 50 psig. There are no overpressure relief valves. The saturation temperature of water which would produce a pressure of 50 psig is about 298 °F. The applicant's heat transfer analyses show that, given the decay heat limits for the fuel loaded in the STC, these temperature and pressure limits will not be exceeded.

When heavy loads are moved using single-failure-proof handling systems as specified in NUREG-0800, "Standard Review Plan," Section 9.1.5, the NRC does not require the consideration of a drop accident. The cranes in the FSBs used to lift the STC are single-failure-proof cranes. Therefore, there is no requirement to analyze a drop of the STC in either FSB. The HI-TRAC with the STC inside is only lifted by the VCT. During the lift, the lift mechanism is not single-failure proof. The lift height is restricted to six inches, and a drop from this height was analyzed with acceptable results. After the HI-TRAC is lifted by the VCT, locking pins are engaged on the VCT prior to transport. The locking pins make the lift mechanism single-failure proof, so there is no requirement to consider a drop accident during VCT transport.

Stress limits for casks and other safety-related pressurized components are specified in various sections of the ASME Code. The STC and the HI-TRAC are required to meet the applicable stress limits per the ASME Code, Section III, Subsection ND, Class 3.

3.3.2 Accidents Inside 10 CFR Part 50 Structures

The following accidents inside Part 50 structures (the FSB) were considered:

a. Accidental drop of a fuel assembly onto the STC fuel basket.
b. Misloading of an incorrect fuel assembly into the STC fuel cells.
c. Earthquake.

3.3.3 Accidents Outside 10 CFR Part 50 Structures

The following accidents outside Part 50 structures were considered:

a. Accidental drop of the loaded HI-TRAC when it is not held by single-failure-proof devices.
b. Fire, considering the most severe of the possible cases (either a fire fed by the VCT's fuel tank, or a fire due to any combustible material located along the VCT haul path).
c. Lightning strike on the loaded HI-TRAC.
d. Earthquake when the loaded HI-TRAC is on the haul path.
e. Flooding.
f. Other natural phenomena, including high winds, tornado, or tornado missiles.
g. Loss of water in the HI-TRAC water jacket.
h. A non-mechanistic tipover of the loaded HI-TRAC.
i. A large radioactive release from the STC was considered, but the NRC staff determined it was non-credible. Even in the most extreme analysis, which was the tipover, the STC and HI-TRAC closure seals remained intact, and both seals would have to fail to allow a radioactive release.

3.3.4 Off-Normal Conditions

The following off-normal conditions were considered:

a. Crane hang-up with the loaded STC not in the SFP or the HI-TRAC.
b. VCT breakdown, resulting in the loaded HI-TRAC remaining outside the FSB for an extended period of time (must demonstrate acceptable temperatures for 30 days, although most analyses were performed until temperatures reached steady state).

3.3.5 Service Life

The applicant stated that the HI-TRAC transfer cask is engineered for 40 years of service life, as discussed in Section 3.4.11 of the HI-STORM 100 FSAR, and that the STC is also designed for 40 years of service life, as discussed in Section 3.3 of the SAR.

3.3.6 Protection Against Natural Phenomena

As specified by GDC-2, SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes and tornados without loss of capability to perform their safety functions. The applicant considered a number of natural phenomena as design-basis events and listed those events in SAR Table 1.1.4, "Accident/Initiating Events and the Resultant Effects." Table 1.1.4 lists earthquakes, flooding, lightning, environmental loadings (extreme wind, tornado, and tornado missile loadings considered in the licensing of the HI-STORM 100), and tornado missile effects not previously considered in the licensing of the HI-STORM 100.

The postulated effect of an earthquake would be tip-over of the HI-TRAC resulting in a reduction in heat transfer and shielding. The applicant determined that the HI-TRAC would be stable under earthquake loadings while connected to the VCT and also when resting on the FSB floor. The applicant completed analyses demonstrating there would be no tip-over of the HI-TRAC at any time during the transfer operation. These analyses are evaluated in Section 3.6 of this safety evaluation. Nevertheless, the effect of a non-mechanistic tip-over has also been analyzed, and the evaluation of that tip-over analysis is presented in Sections 3.5 and 3.6 of this safety evaluation.

The applicant determined that flooding that could affect the HI-TRAC during transfer operations was not credible based on the geography of the site and the location of the fuel transfer route more than 55 feet above the normal Hudson river level. The NRC staff considered the site geography and the location of the fuel transfer route and finds that flooding effects are not credible. Again, the tip-over of the HI-TRAC, which could be a potential result of flooding, has been analyzed with acceptable results.

The applicant considered the effect of a lightning strike on the loaded HI-TRAC in the Entergy HI-STORM 100 Cask System's 10 CFR 72.212 Evaluation Report, Indian Point Energy Center Site Specific Appendix F, and determined that lightning would not impair the safety function of the cask. The NRC staff finds that the configuration of the HI-TRAC for fuel transfer operations would provide adequate protection against the effects of lightning. The applicant concluded that lightning may cause ignition of the VCT fuel, and the applicant performed an analysis of this event. The analysis of this event is evaluated in Section 3.5 of this safety evaluation.

The applicant initially considered the tornado protection provided by the HI-TRAC as adequate. Specifically, the applicant cited the tornado analysis for the HI-TRAC included in the FSAR for the HI-STORM 100 Cask System, which evaluated the HI-TRAC for penetration and deformation of the HI-TRAC shell and lid to demonstrate that the canister inside would not be penetrated and the canister would be retrievable. However, the fuel transfer application specified a new safety function for the HI-TRAC, retention of cooling water, which was not considered in the analysis for the HI-STORM 100 Cask System. Therefore, the NRC staff concluded that the applicant's analysis was inadequate to demonstrate that the HI-TRAC bolted lid connections would retain the annular water volume following a tornado missile impact. The NRC staff informed the applicant of this initial incomplete tornado missile analysis determination by letter dated September 4, 2009.

The applicant addressed the staff's incomplete tornado analysis determination in its response to NRC Review Comment 1.c provided in Attachment 1 to the letter dated September 28, 2009. The applicant stated that a bottom missile shield was designed to protect the lower flanged joint on the HI-TRAC from an incident tornado missile. The applicant stated that the bottom missile shield will be attached to the HI-TRAC for all fuel transfer evolutions. The applicant determined that the bolted connections of the HI-TRAC top lid would not require protection because a tornado missile would not dislodge the top lid and the HI-TRAC would remain vertical following a tornado missile impact. Therefore, in the event the top lid bolts were damaged, the water would remain in the HI-TRAC. The potential degradation of the HI-TRAC confinement boundary by a tornado missile is acceptable because the STC pressure boundary would be unaffected by the tornado missile.

The NRC staff evaluated the effects of natural phenomena considered in the design of the cask transfer system and the protection afforded against those effects. The NRC staff found the scope of the natural phenomena considered in the design acceptable for the specified cask transfer operations. The NRC staff also found that the design afforded appropriate protection against the considered natural phenomena. Therefore, the fuel transfer system satisfies the requirements of GDC-2 with respect to withstanding the effects of natural phenomena such as earthquakes and tornados without loss of the capability of the fuel transfer system to perform its safety functions.

3.3.7 Protection Against Dynamic Effects

As specified by GDC-4, SSCs important to safety shall be appropriately protected against dynamic effects that may result from equipment failures. The proposed spent fuel transfer system is subject to dynamic effects related to equipment failures during handling operations. Specifically, the design of the facility and the fuel transfer system considers the failure of spent nuclear fuel handling equipment that results in dropping of a fuel assembly, the measures necessary to preclude a drop of the STC in or near the spent fuel pool, and the measures necessary to assure safe handling of a loaded HI-TRAC.

The HI-TRAC with the STC inside would initially be moved into the IP3 FSB truck bay by moving the HI-TRAC on an air pad because FSB door clearance limits the overall height of equipment that can be moved through the door. Once in the FSB, the proposed handling of the STC would involve transfer from the HI-TRAC in the FSB truck bay to the IP3 SFP and return of the loaded STC to the HI-TRAC using the upgraded 40-ton FSB overhead crane. The applicant would load the STC using the standard fuel handling equipment described in Chapter 9 of each unit's UFSAR. The loaded STC within the HI-TRAC would be moved out of the IP3 truck bay using the air pad. Once outside the FSB, the VCT would be used to lift and move the HI-TRAC with the loaded STC to outside the IP2 FSB truck bay. There the HI-TRAC with the loaded STC inside would be lowered onto the existing low profile transporter (LPT) and rolled into the IP2 FSB truck bay using the LPT. IP2 uses an LPT to enter the FSB door, while IP3 uses an air pad. Once inside the IP2 FSB, a single-failure-proof crane will be used to transfer the loaded STC to the IP2 SFP and return the unloaded STC to the HI-TRAC.

To control the handling of heavy loads, the applicant proposed a Transport Evaluation Program, Specification 5.1 in the IP2 and IP3 Operating Licenses, Appendix C, "Inter-Unit Fuel Transfer Technical Specifications." Specification 5.1.a states that, for lifting of the loaded STC or loaded HI-TRAC using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, the 10 CFR Part 50 requirements apply. Therefore, the regulatory controls described in the respective unit UFSARs apply to the handling of the HI-TRAC and STC within the FSB using the fuel building crane. Specification 5.1.b states that the Transport Evaluation Program is not applicable when the loaded HI-TRAC is in one of the FSBs or is being handled by equipment providing support from underneath (e.g., on air pads). Specification 5.1.c describes the criteria for design of reliable lifting equipment employed outside structures governed by 10 CFR Part 50 requirements that would permit lifting of the HI-TRAC to any necessary height. The design of the existing VCT satisfies these criteria, and, therefore, the applicant did not postulate an accidental drop from the VCT when the HI-TRAC is fully secured to the VCT through the use of redundant drop prevention features. Except as provided in

Specification 5.1.c, Specification 5.1.d limits the lift height of the loaded HI-TRAC above the transport route surface or other supporting surface to no more than 6 inches.

The applicant evaluated a fuel-handling accident during loading/unloading of the STC within the cask handling areas of the SFPs. Section 9.5 of both the IP2 and IP3 UFSARs stated that the fuel-handling system provides a safe, effective means of transporting and handling fuel. Nevertheless, fuel-handling accidents involving the assumed failure of the handling system resulting in damage to the fuel and nearby structures have been considered in the design of the fuel storage and handling facilities. The applicant concluded that radiological consequences of the fuel handling accident during STC loading would be bounded by the facility fuel handling accident analyses described in Section 14.2.1 of both the IP2 and IP3 UFSARs. Since single-failure-proof cranes are used, it is not necessary to postulate dropping the STC itself. The postulated accident is dropping a spent fuel assembly on top of the STC during the loading or unloading process. The radiological consequences are bounded by the previous fuel assembly drop analyses because the fuel assemblies acceptable for loading in the STC are less radioactive than those handled during refueling, due to the longer cooling time.

The applicant also considered the possible effect on criticality of the fuel from dropping a fuel assembly. The applicant concluded that criticality would be prevented provided the STC neutron absorber panels were not damaged. To demonstrate that structural damage to the STC fuel basket resulting from dropping a fuel assembly on the STC would be acceptable, the applicant provided the results of the structural analysis for criticality prevention in Section 6.2.4 of the SAR. The analytical assumptions included the drop of a spent nuclear fuel assembly and associated handling tool, which have a total weight of 2000 lbs., from a height of 36 inches over the top of the STC fuel basket. Based on the analysis, the applicant concluded that damage to the STC components would be limited to the top portion of the fuel basket above the neutron absorber panels. The NRC staff evaluated the fuel assembly drop analysis in Section 3.6.2.4 of this safety evaluation and found the analysis acceptable. Therefore, the criticality analysis for fuel within the STC would be unaffected by the postulated fuel assembly drop. The applicant has implemented measures to assure a low probability of a fuel-handling accident, and the accident analyses have demonstrated that the consequences of an unlikely fuel-handling accident would be acceptably small. Accordingly, the NRC staff concluded that the fuel transfer system and contained spent nuclear fuel have appropriate protection against criticality due to the effects of equipment failures that could result in dropping a spent nuclear fuel assembly.

The applicant stated that handling of the STC will be conducted in accordance with NUREG-0612 guidelines. IP2 and IP3 have existing heavy load handling programs described in Chapter 9 of each of the Unit's UFSARs that would apply to the STC movement in and out of each unit's SFP. IP2 has an existing single-failure-proof gantry crane with a 110 ton capacity. The applicant stated that the 40 ton capacity IP3 FSB crane would be upgraded to single-failure-proof design consistent with the guidelines of NRC report NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and ASME Standard NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes," and the upgrade would be conducted pursuant to provisions of 10 CFR 50.59. Therefore, the design and operation of the overhead fuel building cranes will be consistent with NUREG-0612 guidelines for single-failure-proof handling systems, but outside the scope of the detailed review in this safety evaluation. As described in Section 3.1.2.1 of this safety evaluation, the STC design includes integral special lifting devices (i.e., STC lift lock and STC lifting device) and integral interfacing lift points (i.e., threaded bolt holes and STC lifting trunnions) designed to meet the guidance of NUREG-0612, Section 5.1.6 for use with single-failure-proof handling systems. Consistent with the proposed Transport Evaluation Program, the

STC handling operations within the FSB will be subject to the heavy load handling program provisions described in each unit's UFSAR. Therefore, the NRC staff finds that STC handling operations will be consistent with the guidance of NUREG-0612 and provide appropriate protection against equipment failures during STC handling.

The applicant considered additional handling accidents for the period when the loaded STC is secured in the HI-TRAC. However, consistent with the provisions of the Transport Evaluation Program, no handling accidents were postulated for the time when the HI-TRAC rests on the air pad for transfer in and out of the IP3 loading bay, when the HI-TRAC rests on the LPT for transfer in and out of the IP2 loading bay, or when the HI-TRAC is secured to the VCT with redundant load drop protection. Under other described conditions during fuel transfer operations, the maximum lift height of the loaded HI-TRAC will be limited to 6 inches, consistent with Specification 5.1.d of the Transport Evaluation Program. The applicant stated that, if necessary, the lift height of the loaded HI-TRAC will be controlled by placing impact limiters underneath the HI-TRAC as it is raised on the VCT to ensure that this limit is not exceeded. Once the locking pins are engaged, attaching the HI-TRAC to the VCT and providing redundant drop protection, the lift height would no longer be limited by Specification 5.1.d. In Chapter 6 of the SAR, the applicant provided an analysis of a 6 inch drop of a loaded HI-TRAC that demonstrated acceptable structural performance. The NRC staff evaluated the applicant's analysis of the postulated 6-inch drop of a loaded HI-TRAC in Section 3.6.2.5 of this safety evaluation and found the analysis acceptable. Therefore, the proposed handling of the loaded HI-TRAC will provide appropriate protection against equipment failures during proposed movement of the loaded HI-TRAC.

3.3.8 Safety of Fuel Storage and Handling

As specified by GDC 61, "Fuel Storage and Handling and Radioactivity Control," 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 applicant 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 at the top of the HI-TRAC mitigates the pressure rise that results from the thermal expansion of the STC and the annular water volume.

The design of the fuel transfer system includes provisions for periodic inspection and testing of components important to safety. The design of the HI-TRAC includes the capability to remove the top and bottom lids of the HI-TRAC for replacement of seals and inspection of other components important to safety. The top lid of the STC is also removable, which allows inspection of the seals as well as supporting planned fuel transfer operations. The top lids of both the STC and the HI-TRAC include sealable vent and drain ports that support planned operations and periodic testing.

The STC and HI-TRAC contain design provisions for shielding. The NRC staff evaluated the suitability of the shielding in Section 3.7 of this safety evaluation.

For the fuel transfer system, the containment, heat removal, and retention of water inventory functions are inter-related. 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. With the pressure boundaries intact, the water inventory is maintained, assisting in the transfer of heat from the fuel assemblies to the outside air. 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 bubble (STC) or air bubble (HI-TRAC)), and the heat removal rate determine whether the internal pressure of the STC and HI-TRAC remain below their respective design pressures. When the pressure within the STC and HI-TRAC remain within the respective design pressures under postulated accident conditions affecting the heat removal rate, the design is adequate to prevent a significant reduction in water inventory under accident conditions. The NRC staff evaluated the heat removal rate in Section 3.5 of this safety evaluation.

The NRC staff reviewed the applicant's selection of accidents that challenge the adequacy of the fuel transfer system heat removal mechanisms. In accordance with 10 CFR 50.90 and 10 CFR 50.34(b)(4), the application for a license to support fuel transfer shall include an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Furthermore, the evaluation shall include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," contains guidance on the evaluation of postulated accidents. This guidance states that the effects of anticipated process disturbances and postulated component failures should be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

To address the performance of the fuel transfer system, the applicant evaluated several postulated accident scenarios. Table 1.2, "Failure Modes and Effects Analysis," and Table 1.3, "Accident/Initiating Events and the Resultant Effects," in Attachment 1 to the supplemental letter dated September 28, 2009, provided an initial assessment of various proposed equipment failures and initiating events. The applicant determined that many of the failure modes presented in Table 1.2 were either:

i. ruled out by defense-in-depth operational measures, or
ii. detected and corrected before the loaded cask leaves the Part 50 structure.

Accordingly, with the exception of an incorrectly loaded fuel assembly, the applicant did not evaluate the consequences of many of the equipment failures and initiating events listed in Table 1.3 assuming they had progressed without action to correct the condition. Instead, the applicant initially credited operational measures to prevent or provide early detection and correction of conditions before parameters go beyond design bounds. In its review, the NRC staff questioned the reliability of such operational measures to prevent or detect and correct certain adverse conditions. As a result, the applicant completed additional analyses of adverse conditions, including the analysis of a non-mechanistic cask tip-over event and a loss of the HI-TRAC annulus water inventory.

In Section 5 of the SAR the applicant presented analyses of the following conditions that could challenge the adequacy of the fuel transfer system heat removal mechanisms:

- Normal on-site transfer of IP3 fuel
- Loss of HI-TRAC water jacket
- External fire (rupture of transporter fuel tank)
- Loss of HI-TRAC annulus water
- Fuel misloading
- Non-mechanistic tipover accident
- Crane malfunction

The NRC staff's evaluation of these analyses is presented in Section 3.5 of this safety evaluation.

The applicant developed processes for establishing the necessary initial conditions within the STC and HI-TRAC, and these processes are described in Section 10 of the SAR. Key initial conditions include the fuel load, the presence of water in the STC, the availability of overpressure protection, the presence of water in the HI-TRAC, the establishment of the correct steam bubble within the sealed STC, and the establishment of the correct air gap within the sealed HI-TRAC. Many of these initial conditions are assumed in the accident analyses, and, therefore, technical specification limiting conditions for operation or surveillance requirements provide assurance these initial conditions have been properly established. These initial conditions and associated accident analyses demonstrating that necessary water inventory is retained within the STC demonstrate that the requirements of GDC 61 related to containment, residual heat removal, and water inventory retention would be satisfied by the proposed design of the fuel transfer system.

The applicant proposed TS 3.1.2 to ensure an acceptable inventory of fuel and non-fuel hardware 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 1105.2 Watts per cell for the four interior cells and no more than 650 Watts per cell in the eight peripheral cells, which results in a total STC heat generation rate of no more than 9.621 kW) would ensure that STC design pressure will remain below design pressure for all analyzed conditions. Therefore, the proposed TS limits on fuel loading are acceptable with respect to heat generation rates.

The initial water inventory of the STC is established prior to placing the STC in the SFP. Moreover, the loading of the STC within the SFP inherently ensures an adequate initial water inventory.

The proposed operations retain overpressure protection through various measures as the configuration of the STC changes. During the handling of the STC, the top lid of the STC performs lifting device and shielding functions, but the lid is not tightly secured in place. Thus, the lid provides a pressure relief path during handling of the STC until the STC is placed in the HI-TRAC. The analysis of the crane malfunction event demonstrates acceptable safety during the STC handling phase of the fuel transfer operation. The STC handling phase is safe because the low heat generation rate ensures a time of over 17 hours to reach saturation conditions within the STC starting from typical spent fuel pool temperatures. Therefore, there is no immediate need for make-up water and there is reasonable assurance that the loaded STC can be returned to the SFP before a significant loss of coolant inventory could occur.

Once the STC is in the HI-TRAC, the applicant has developed operational steps to ensure that temporary pressure relief valves are installed and verified to be connected to the STC internal space before the STC lid is secured. Steps 27 through 30 of Section 10.2.3 of the SAR, "Removal of STC from SFP and placement in HI-TRAC," (including a preceding cautionary note) specify the actions to install pressure relief and indicating devices, verify their connection to the STC internal volume, and secure the STC lid to the STC flange. TS Surveillance Requirement (SR) 3.1.4.2 specifies that the applicant verify the installation of a code-compliant relief valve or rupture disk and two channels of pressure instrumentation during the performance of the STC pressure rise test. This surveillance requirement provides an appropriate verification of overpressure protection separate from the procedural actions to install the overpressure protection device.

After the STC lid is secured, the applicant has additional operational steps to establish a steam space within the STC and begin a pressure rise test. Steps 33 through 40 of Section 10.2.3 of the SAR, "Removal of STC from SFP and placement in HI-TRAC," specify the actions to ensure the STC is filled with borated water and establish an appropriate steam space within the STC internal volume. These steps are subject to Technical Specification controls through TS SR 3.1.3.1, which requires that the applicant verify that steam is emitted from the STC drain connection and that the water displaced from the STC by the steam is an acceptable volume.

Finally, the STC pressure rise test required by TS LCO 3.1.4 provides verification that the fuel assemblies loaded in the STC conform to the analysis assumptions for heat generation, which will prevent STC overpressure. Step 44 of SAR Section 10.2.3, "Removal of STC from SFP and placement in HI-TRAC," specifies performance of the pressure rise test and provides actions to address a condition where the measured pressure increase exceeds the limit. Step 44 actions are consistent with Required Action A of TS 3.1.4, which apply when the rate of STC cavity pressure rise exceeds the TS 3.1.4 limit of 0.2 psi per hour. Pressure rise within the required limit for the 24-hour period provides reasonable assurance that STC loading conditions would prevent development of an overpressure condition in the STC throughout the duration of fuel transfer operations. This conclusion is based on the analysis of the normal transfer condition provided in Section 5.3.4 of the SAR, which is evaluated in Section 3.5 of this safety evaluation. Furthermore, the administrative controls applied to the loading of the STC and establishment of the steam space within the STC provide additional assurance that the STC internal conditions were established consistent with the assumptions of the analysis.

The water inventory in the HI-TRAC annulus forms part of the normal heat transfer path from the loaded STC to the environment during fuel transfer operations. The water inventory in the HI-TRAC water jacket also contributes to the effective transfer of heat from the interior of the HI-TRAC to the environment. The initial inventory of water in the HI-TRAC annulus is established during preparation activities after the STC and its centering assembly have been placed within the HI-TRAC. Step 11 of Section 10.1.3, "Preparation and Setup for Use," of the SAR specifies that personnel fill the annulus between the STC and the HI-TRAC with demineralized water to an elevation just below the top of the STC flange. Step 5 of the same section calls for verification that the HI-TRAC water jacket is filled. Additionally, Step 31 of Section 10.2.3, "Removal of STC from SFP and placement in HI-TRAC," specifies that the annulus between the STC and the HI-TRAC be filled as necessary with demineralized water to within 1 inch of the top of the STC lid.

In Section 5.4.3 of the SAR, the applicant presents the results of an analysis of the simultaneous loss of water from the HI-TRAC water jacket and HI-TRAC annulus. The applicant assumed that the water in the annulus and jacket was replaced with air and the STC had reached maximum steady state temperatures with the maximum permitted internal heat generation rate. Under these conditions, the applicant determined that the peak pressure within the STC would remain well below the STC design pressure and fuel temperature limits would not be exceeded. The NRC staff's evaluation of this and other postulated accident conditions and analyses is provided in Section 3.5 of this safety evaluation. The NRC staff found the loss of the HI-TRAC annulus water inventory to be among the most limiting accident conditions with respect to peak STC pressure with the STC in its normal orientation. Other analyzed accident conditions produced similar results. Since the peak pressure for these accident conditions would remain within the STC design pressure, the NRC staff concluded there is reasonable assurance that the fuel transfer system would maintain appropriate containment and residual heat removal capability under accident conditions, thereby precluding a significant reduction of coolant inventory under accident conditions. Thus, the design of the fuel transfer system complies with the requirements of GDC 61 and is acceptable.

3.3.9 Instrumentation

As specified by GDC 63, "Monitoring Fuel and Waste Storage," 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 fuel transfer system is a relatively low energy system because of the limits on the internal heat generation rate loaded in the STC. The system is also designed in a manner that preserves water inventory necessary for residual heat removal under accident conditions. The installation of permanent instrumentation such as pressure instruments or temperature instruments in the STC and HI-TRAC would necessitate additional penetrations of the pressure boundary. The additional penetrations result in a greater risk of breaching the pressure boundary during postulated accidents. The applicant proposed that there be no permanently installed instrumentation. Temporary pressure instruments are installed in the STC lid during the pressure rise test following the loading of the STC. These instruments are removed following the completion of the test, and cover plates are installed over the instrument connections. With the STC and HI-TRAC lids bolted in place, the only reasonable safety action possible for an overpressure condition would be to remove the HI-TRAC lid and establish cooling water flow in the STC using the connections in the STC lid. Since the pressure rise test provides verification that the heat generation rate is within design limits, and analyses demonstrate adequate heat removal rates from the STC to the environment under those conditions, the NRC staff finds that additional instrumentation is not necessary and that the installation of temporary pressure monitoring equipment under administrative control during the pressure rise testing is adequate to satisfy GDC 63.

3.4 Criticality Evaluation

3.4.1 Background

There are two aspects of this license amendment request that require a nuclear criticality safety (NCS) analysis. The first involves placing the IP3 spent fuel into the STC and considering the criticality safety of the STC in that configuration. The second involves placing the IP3 spent fuel into the IP2 SFP and considering criticality safety in that configuration. 10 CFR 50.68, "Criticality accident requirements," is the primary regulatory requirement that applies to these analyses, along with 10 CFR 50, Appendix A, GDC 62, "Prevention of criticality in fuel storage and handling."

The NCS analysis for the STC was performed for the applicant by Holtec International (Holtec). The Holtec NCS analysis is described in Chapter 4 of the SAR. Revision 5 of the SAR, as modified by the applicant's letter dated March 2, 2012 (Reference 3), was reviewed in detail for this safety evaluation, and the minor changes made in Revision 6 of the SAR were also considered. The SAR provides the methodology for the NCS analysis for the STC. Revision 5 of the SAR was included as an enclosure to the applicant's December 15, 2011, letter, and Revision 6 was included as an enclosure to the applicant's April 23, 2012, letter.

The NCS analysis for placing the IP3 spent fuel into the IP2 SFP is addressed in Section 4.8 of the SAR. The applicant initially concluded that the IP2 and IP3 spent fuel was so similar that no further analyses were needed. The NRC staff had numerous questions regarding that conclusion, which the applicant addressed in its letter dated March 2, 2012. As discussed below, the NRC staff determined that there were enough differences between the IP2 fuel and the IP3 fuel, and between the IP3 fuel and the assumptions used in the IP2 SFP NCS so that the IP3 fuel may only be placed in a designated region of the IP2 SFP where a satisfactory margin to criticality has been demonstrated.

3.4.2 Proposed Change

Currently, there are no IP2 TS or IP3 TS regarding the STC. This amendment will add requirements to both the IP2 TS and the IP3 TS to govern the use of the STC. Those requirements include, among other things, the requirement to maintain subcriticality, with specified margins. The following discussion provides a broad description of the proposed TS.

The STC subcriticality requirements in the TS consist of a requirement for a minimum soluble boron concentration, and descriptions for three different loading configurations. There are two basic loading configurations, one of which has two subdivisions. The first configuration is where every STC cell contains a fuel assembly that meets the stipulated enrichment, burnup, and operating history requirements for loading 12 fuel assemblies. This configuration has two sub-divisions which are based on operating history, and each has its own enrichment and burnup requirements. The second configuration restricts the loading to a maximum of 8 fuel assemblies with at least the 4 center cells empty. This configuration applies for fuel assemblies that do not meet the enrichment, burnup, and operating history requirements for full loading of the STC. There are also proposed Design Feature and Program TS requirements that support the STC.

The LAR seeks to temporarily store IP3 spent fuel in the IP2 SFP, using the IP2 SFP storage requirements. There were initially no proposed changes to the IP2 SFP storage requirements. During the staff's review of the LAR, limitations were established in the IP2 TS to limit which IP3 spent fuel can be stored in the IP2 SFP, and which regions of the IP2 SFP the IP3 fuel can be stored in. Under these limitations, only spent fuel from IP3 cycles 1 through 11 will be allowed to be moved to the IP2 SFP and this fuel will only be allowed to be stored in IP2 SFP Region 1-2.

3.4.3 STC Nuclear Criticality Safety Evaluation

3.4.3.1 Methodology

There is no generic methodology for performing NCS analyses for spent fuel. Each analysis is specific to the application and the fuel involved in the analysis.

The NRC staff issued an internal memorandum on August 19, 1998 (Reference 4), containing guidance for performing a review of SFP criticality analyses. This memorandum is known colloquially as the 'Kopp Letter,' after the author. While the Kopp Letter does not specify a methodology, it does provide guidance on more salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactors, both borated and unborated. The Kopp Letter has been used as a guideline for virtually every pressurized-water reactor SFP criticality analysis since then, including this evaluation.

The NRC staff has also issued Interim Staff Guidance (ISG) DSS-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools." The draft version of DSS-2010-01 was issued in September 2010; the final version was issued in October 2011 (Reference 5). The staff utilized ISG DSS-2010-01 in this evaluation.

Section 4.2 of the SAR, "General Methodology," provides an overview of the applicant's NCS analysis. Section 4.2.1 contains a broad discussion which had no direct bearing on the actual analysis; therefore, the NRC staff takes no position on that section. The only aspect of that section that may be relevant is the comparison of the STC to the HI-STAR 100 transportation package; however, that discussion lacks sufficient detail to determine whether or not conclusions reached during the HI-STAR 100 licensing action are relevant to approval of the STC. Section 4.2.2 is titled "Details of Methodology." While that section provides some details regarding the methodology, in actuality it only provides an overview. To review the "details" of the methodology, one must read the entire report. As with most spent fuel NCS analyses, the methodology is specific to the analysis and may not be appropriate for other applications.

3.4.3.2 Computer Code Validation

The criticality safety analyses for the Indian Point STC design were performed using computer codes. To be used in criticality safety analyses, the adequacy of the computer codes to accurately predict the effective neutron multiplication factor, k_{eff}, of a system must be established. Any biases, and the bias uncertainties associated with the codes and models used in the analyses, must also be appropriately quantified. ANSI/ANS-8.1, ANSI/ANS-8.17, NUREG-0800, and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," require these computer codes to be benchmarked to ensure the codes' ability to perform the calculation and identify the bias and uncertainties of the codes in predicting the system's k_{eff}. Therefore, the validation of the criticality safety analysis codes is necessary to demonstrate compliance with 10 CFR 50.68 ("Criticality accident requirements").

Code validation is typically achieved by comparing the calculated value of the system parameter to the result of a physical experiment that the computer model simulates. Typically, multiple experiments are selected and modeled. A statistical analysis is performed to determine the average value of the differences between the computed results and the data from the experiments. This difference is the bias of the computer code. The standard deviation of the average difference is the uncertainty associated with the bias. Typically, the bias is used with 2 times the standard deviation. The final value adjusted with bias and bias uncertainty is the value for the parameter of interest with a 95 percent probability, 95 percent confidence level.

The applicant used the burnup credit of 12 actinides and 13 fission products in the criticality safety analysis of the Indian Point inter-unit fuel transfer cask Configuration 1 loading, in which all 12 cells are loaded with fuel assemblies. The criticality safety analysis for a system that takes burnup credit consists of two steps: (1) to determine the spent fuel isotopic composition at a given initial enrichment, burnup, and cooling time, and (2) to determine the neutron multiplication factor of the cask at various conditions with the given fuel load. These two analyses are typically accomplished with two different computer codes because none of the current neutronics codes can perform both tasks. The criticality safety analysis for the Configuration 2 loading, which leaves the four center cells empty, assumes that the fuel contents are unirradiated. This is done to simplify the analysis; the STC has not been approved to move unirradiated fuel, but the analysis using unirradiated fuel bounds the criticality analyses for irradiated fuel.

The applicant used computer codes CASMO-4 for the spent fuel composition analysis and MCNP4a for the criticality safety analyses of the Indian Point fuel transfer cask design. Both computer codes are benchmarked against appropriate physical experiments. The final k_{eff} values of the systems of interest are adjusted with the bias and bias uncertainties associated with the modeling, computational approach, and computer codes. The following sections discuss the details of the code benchmarks performed by the applicant.

3.4.3.2.1 Depletion Code Benchmarking

The CASMO-4 code is a lattice analysis code designed primarily for generating macroscopic cross sections for nuclear reactor neutronics analysis. It also provides the isotopic composition of the fuel with given initial enrichment, depletion history, burnup, and cooling time. With the use of the "J" library, the CASMO-4 code is capable of tracking all of the isotopes that are used in burnup credit analysis.

In accordance with NUREG/CR-6811, "Strategies for Application of Isotopic Uncertainties in Burnup Credit," there are three different methods that can be used to perform code benchmark analysis, i.e., the isotope concentration correction factor method, the direct difference method, and the Monte Carlo Uncertainty Sampling method. The first two methods are commonly used by the industry in burnup credit analyses. In either of the first two methods, the essential assumption is that the isotopic compositions of the spent fuel as measured by destructive radiochemistry assay techniques for fuel samples are accurate and the biases and bias uncertainties are introduced only by the computational method and the computer code(s).

In the isotope concentration correction factor method, the calculated isotopic concentration for a given sample is compared to that of the measured data. A correction factor is obtained by statistical analysis of the ratios of measured/calculated data from all selected samples. The calculated concentrations of the isotopes for which burnup credit are sought are adjusted using the correction factors that were determined and then fed into the criticality safety analysis code to calculate the neutron multiplication factor, k_{eff}, of the system of interest. The biases and bias uncertainties of the isotopic concentrations that were calculated by the fuel depletion analysis code are thus propagated into the final k_{eff} value of the system under evaluation.

In the direct difference method, two separate neutron multiplication factors are calculated for the same system. One calculation uses the measured isotope concentration from each sample and the other calculation uses the calculated isotopic concentration for the same sample with identical depletion history. The criticality models are identical in these two calculations. The difference of the k_{eff} values from these two calculations is determined for each set of chemical assay measurements. A statistical analysis on the Δk_{eff} values obtained from criticality calculations is made to determine the average value of the bias and bias uncertainty associated with the depletion calculation computer code. A regression analysis on the Δk_{eff} values is made to determine the trend of the Δk_{eff} with respect to burnup and enrichment.

The applicant used the direct difference method for benchmarking the CASMO-4 code for major actinides. **[[**

]] The applicant presented its depletion code benchmark analysis for the CASMO-4 code in Appendix A of the Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask," Revision 5. **[[**

]] Table A.2 presents the results of the benchmark analyses.

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]] The bias and bias uncertainty associated with the depletion code are added to the final k_{eff} result for a given burnup and initial enrichment. With this adjustment, the bias and uncertainty associated with the depletion analysis computer code is accounted for in the criticality safety analysis of the cask.

The applicant used the isotope concentration correction factor method for benchmarking the CASMO-4 code for minor actinides and fission products in the Indian Point inter-unit fuel transfer cask design. Table A.3 of Appendix A to the Holtec Report No. HI-2084176, Revision 5, provides a list of all the minor actinides and fission products for which burnup credit is sought. Table A.13 of Appendix A provides a list of samples for which the measured fission products and minor actinides are used in this analysis. The ratio of the measured/calculated isotopic concentrations is taken for each isotope in each chemical assay sample. A statistical analysis is performed on the measured-to-calculated concentration ratio. An average correction factor is obtained for each isotope through the statistical analysis. The results are a set of isotopic concentration correction factors for the fission products and actinides. The standard deviation of the correction factor is the uncertainty of the correction factor.

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]] Use of average correction factors implies that the correction factors are in uniform distribution rather than Normal or off-Normal distribution, therefore this method is more penalizing. The results are hence more conservative.

In addition, the applicant performed a normality test of the measured data to identify non-Gaussian distributions. **[[**

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In order to prevent non-conservative calculation results, the applicant further adjusted the correction factor by using the lower bound values of the average values of the correction factors by either truncating the correction factor or rounding up the correction factor to a lower value for fissile isotopes and to a higher value for absorber isotopes. This is consistent with the correction factor method outlined in NUREG/CR-6811.

The benchmark analysis of the depletion code CASMO-4 demonstrated that the code is capable and adequate for the fuel composition analyses of the spent fuel to be transferred using the Indian Point STC. Biases and bias uncertainties associated with the depletion code have been identified and quantified through the code benchmarking analyses. The NRC staff finds that the results are adequate to meet the guidance in NUREG/CR-6811.

3.4.3.2.2 Criticality Safety Analysis Code Benchmarking

The MCNP4a code is a three-dimensional Monte Carlo method multi-particles transport computer code based on particle transport theory. The ENDF/B-V continuous energy nuclear data libraries distributed with the code were used in this application. The code is capable of modeling complex systems containing various compositions of fissile materials and other components.

The Indian Point STC is a shielded wet spent fuel transfer system. The cask design has two configurations: Configurations 1 and 2. Configuration 1 has fuel assemblies in all twelve cells in the basket assembly. Configuration 2 has eight fuel assemblies in the outer cells with the four center cells empty. The only difference is that the four fuel cells in the center of the basket are physically blocked from being loaded with fuel assemblies, when loaded in the Configuration 2 pattern. Figures 4.5.1 and 4.5.2 of the SAR provide sketches of these two configurations.

The applicant analyzed three loading patterns, namely Configurations 1A, 1B, and Configuration 2. Configuration 1A is a loading pattern for fuel assemblies that meet the burnup-enrichment requirements of fuel assemblies that were ***not*** exposed to control rods or hafnium flux suppressors during irradiation. Configuration 1B is a loading pattern for fuel assemblies that meet the burnup-enrichment requirements of fuel assemblies that were exposed to control rods or hafnium flux suppressors during irradiation. Configuration 2 is for fuel assemblies that do not meet the minimal burnup requirements of either configurations 1A or 1B.

The criticality safety analysis code, MCNP4a, has been benchmarked using the combination of critical experiments with fresh UO_2 fuel and MOX fuel from the International Handbook of

Evaluated Criticality Safety Benchmark Experiments (IHECSBE), the Haut Taux de Combustion (HTC) experiments that were conducted by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), and Commercial Reactor Critical (CRC) experiments. The applicant presented its code validation analysis in the SAR and in Holtec Report No. HI-2084176, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Cask" Revision 5, October 26, 2011, and Holtec Report No. HI-2032973, "Commercial Reactor Critical Benchmarks for Burnup Credit," Revision 5, October 6, 2011. The applicant followed the approach specified in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Methodology" in its code validation analyses. **[[**

]] These experiments cover the appropriate ranges of values of several key parameters to ensure that the safety analysis results are within the area of applicability of the validation suite. The selection of critical experiments for benchmarking analysis was primarily focused on consideration of geometric and material characteristics similar to that of spent fuel storage and transport systems.

The applicant discussed the Area of Applicability (AOA) of these selected experiments. The applicant identified the key parameters for comparison of the selected benchmark experiments to the STC. The key parameters evaluated included uranium enrichment, plutonium content, fuel rod outer diameter and rod pitch, fuel density, soluble boron concentration, neutron poisons, interstitial and reflecting materials, and energy of average lethargy of neutrons causing fission. The applicant also provided a comparison of the ranges of key parameters of the critical experiments and provided the safety analysis models. This comparison adequately demonstrates that the safety analysis models are within the area of applicability of the critical experiments. **[[**

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Configuration 2 permits only eight fuel assemblies to be loaded in the outer cells of the fuel basket with the center four cells physically blocked. When the cask is loaded, the four center cells form a water hole in the configuration with a maximum width of 18 inches. There is no critical benchmark experiment appropriate for this configuration, although the NRC staff notes that there are a few benchmarks with water gaps. To verify that the code is capable of modeling this configuration without significant trending, the applicant performed a trending study of the impact of the center water hole to the k_{eff} of this configuration [[

]] The results show that the center water gap does not have a significant effect on the k_{eff} of the system as shown in Table 4.7.24 of the SAR. [[

]] The result of this trending study demonstrates that the large water gap does not have an adverse impact to system criticality safety.

The statistical methods used for trending analysis and bias and bias uncertainty determination are described in Appendix 4.A of the SAR. These methods are consistent with standards such as ANSI/ANS-8.1 and ANSI/ANS-8.17 and the recommendations of NUREG/CR-6698 and NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," and hence are determined to be appropriate for this application. Trending analysis of key parameters was performed by the applicant and was taken into consideration for the bias and bias uncertainty determination. Table 7.1 of Holtec Report No. HI-2084176, Revision 5, provides a summary for the different components of the biases and bias uncertainties associated with the isotopic composition analysis code and the criticality safety analysis code. Finally, these biases and uncertainties were applied to the calculated results for determination of the spent fuel loading curve. Figure 4.7.1 of the SAR provides the required minimum burnup for fuel assemblies as a function of the initial enrichment for both Configuration 1A and 1B. Since Configuration 2 is conservatively analyzed with fresh fuel, all fuel that meets the maximum enrichment limit automatically qualifies for loading into this configuration as far as criticality safety is concerned.

Based on the information presented in the SAR, the associated supporting documents, and the applicant's responses to the staff's Requests for Additional Information, the NRC staff finds that the applicant has correctly benchmarked the computer codes for the Indian Point STC criticality analyses.

3.4.3.3 <u>STC Fuel Basket</u>

The fuel basket is the rack within the STC that holds the fuel assemblies. Section 4.5.4 of the SAR provides a description of the STC fuel basket and surrounding materials. The fuel basket consists of a rectilinear arrangement of stainless steel plates, forming a total of 12 cells to hold the fuel assemblies. The 12 cells are arranged with four cells forming a face-to-face center array; two additional cells are face-to-face on each side of the center square. There is one Metamic neutron absorber panel attached to each basket cell wall, including the cell walls on the periphery of the basket, with a stainless steel sheathing plate. Half of the Metamic neutron absorber panels are in the cell, the other half are on the other side of the wall in the adjacent cell or on the outside of the periphery wall. The basket is surrounded by the STC structure, which is a cylindrical canister with a steel-lead-steel wall. The fuel basket parameters relevant to the NCS analysis are summarized in Table 4.5.8 of the SAR. The fuel basket models are shown in Figures 4.5.1 and 4.5.2 of the SAR.

3.4.3.3.1 STC Mechanical Uncertainties

The material and configuration of the STC fuel basket and surrounding materials contributes to the reactivity of the fuel; the material does so by providing a fixed neutron absorber, while the configuration does so by controlling the fuel assembly spacing. The NRC staff has provided guidance on how these uncertainties should be treated, in the Kopp Letter.

The applicant's analysis determined that the reactivity and the STC dimensions are dependent. The analysis performed a sensitivity study to determine the most reactive combination for the two 12-fuel-assembly loading configurations due to manufacturing tolerances or placement of the fuel assembly. This study analyzed the reactivity assuming small variations in the cell wall thickness, the distance from the center of one cell to the center of the adjacent cell, and the position of the fuel assembly inside a cell. The analysis then performed a sensitivity study to compare the reactivity of the fuel basket with the most reactive configuration, to the reactivity of the fuel basket with nominal design dimensions. The report states the largest increases for a burnup/enrichment combination were applied as a bias in the final determination of k_{eff}. There appears to be a discrepancy, however, in that for Configuration 1B the Basket Bias in Table 4.7.1 for burnup/enrichment combinations 5.7/2.0 and 27.2/3.0 do not match the maximum values in Table 4.7.9b. Nor do the Basket Bias values in Table 4.7.1 appear to include the stated uncertainty. The NRC staff considers that these errors are small and do not indicate that regulatory requirements would not be met; they are considered in the aggregate in section 3.4.5 of this safety evaluation.

The applicant's analysis used the largest reactivity increase due to the consideration of tolerances for the 12-fuel-assembly loading configurations as a bias for the 8-fuel-assembly loading configuration. No basis was provided for why this is appropriate. With respect to Configuration 2 there is approximately 1000 pcm of margin in the unborated estimation of k_{eff}. Additionally, limits are being placed on the IP3 fuel that can be moved under this license amendment. The fuel assemblies that can be moved are those from IP3 cycles 1 through 11. Those fuel assemblies have a maximum nominal fresh enrichment of 4.4 wt% ^{235}U. The 8-fuel-assembly loading configuration was analyzed assuming fuel with a maximum nominal fresh enrichment of 5.0 wt% ^{235}U. These assumptions provide significant margin, allowing the NRC staff to conclude that a more precise determination of the STC mechanical tolerances is not necessary.

3.4.3.3.2 STC Water Temperature

NRC guidance provided in the Kopp Letter states that the criticality analysis should be done at the temperature corresponding to the highest reactivity. If the STC has a positive moderator temperature coefficient (MTC), the temperature corresponding to the highest reactivity would be the highest allowed operating temperature. The SAR documents several sensitivity studies done to determine the effect of the STC water temperature. The sensitivity studies included unborated and borated conditions up to 1000 ppm of soluble boron. The sensitivity studies indicate that at up to 1000 ppm of soluble boron, full density water with an associated water temperature of 39.2 °F was the limiting condition. The accident analysis determined that 1053 ppm of soluble boron is required to maintain regulatory compliance. The analysis further indicates that at some point above 1000 ppm of soluble boron, full density water with an associated water temperature of 39.2 °F will not be the limiting condition; however, the analysis indicates that this would likely occur at soluble boron concentrations much higher than 1053 ppm. The staff notes that the TSs require a minimum of 2000 ppm soluble boron in the STC, which provides considerable negative reactivity above the 1053 ppm required in the accident analysis.

The SAR states that the design calculations that estimate k_{eff} were performed at 39.2 °F, but a comparison of Tables 4.7.23 and 4.7.2 indicate those calculations may have been performed at 80.6 °F (300 degrees Kelvin). In response to the staff's inquiry, the applicant confirmed that all design calculations that estimate k_{eff} were performed at 39.2 °F. The staff concludes that this is acceptable.

3.4.3.4 Fuel Assembly

3.4.3.4.1 Selection of Bounding Fuel Assembly Design

The basic fuel design used at IP3 has been the Westinghouse 15x15 fuel assembly. Over the years, IP3 has used several different variations of the Westinghouse 15x15. The SAR categorizes the applicant's fuel into three 'types.' The 'types' are defined in Table 4.5.1 of the SAR. Type 1 has variations of the Westinghouse Vantage 15x15 product line. Types 2 and 3 are two variations of the older Westinghouse LOPAR (Low Parasitic) designs where the difference appears to be the fuel pellet diameter. The analysis includes a sensitivity study to determine a limiting fuel assembly type. The results of the sensitivity study are provided in Table 4.7.5 of the SAR, as a delta k_{calc}. In the SAR Type 1 was selected as the limiting design, stating "The results for assembly Types 2 and 3 are statistically identical to or less than the result for assembly Type 1. Further note that the studies on burnable absorbers in Section 4.7.1.2.2 indicate a higher reactivity for fuel with IFBA, which was only used with assembly Type 1. Assembly Type 1 is therefore used as the bounding assembly in all subsequent calculations." However, Table 4.7.5 indicates that Type 2 and/or Type 3 fuel assembly designs would actually be more reactive in some scenarios. The analysis seems to consider the uncertainty associated with the delta k_{calc} comparison to make the conclusion that the fuel types are statistically identical if the delta k_{calc} comparison is less than the uncertainty of the delta k_{calc} comparison. The NRC staff believes it is inappropriate to draw that conclusion, as the uncertainty also indicates that the delta k_{calc} may actually be larger by the amount given by the uncertainty. Therefore, it is not clear that the analysis has correctly identified the limiting fuel assembly for all scenarios. Additionally, while reactivity of fuel assemblies that used integral fuel burnable absorbers (IFBA) is demonstrated to bound fuel assemblies that did not use IFBA, the use of IFBA is not the limiting condition for determining the post-irradiation reactivity of the IP3 fuel assemblies. The actual limiting condition is either (a) fuel assemblies that contained a hafnium flux suppressor during operation, or (b) fuel assemblies that may have had control rods inserted during operation, as demonstrated in Section 4.7.1.2.2.2 of the SAR. Therefore, whether or not the Type1 fuel assembly is the limiting design is dependent upon whether or not there is sufficient analytical margin to the regulatory limits to accommodate other potentially more reactive fuel assembly configurations. Therefore, as discussed later in this section, the NRC staff will apply an appropriate penalty to the final k_{eff} determination in evaluating whether or not the regulatory requirements are met.

The delta k_{calc} comparisons ignore the presence of structural parts of the fuel assemblies such as spacer grids and end fittings. An analysis was performed to evaluate the effect of the spacer grids. That analysis is described in Section 4.7.9.3 of the SAR and the results are provided in Table 4.7.15b of the SAR. From the table it is clear the grid analysis was not performed in the exact storage configurations being requested for the STC. Therefore, definitive conclusions about the effect of the grids cannot be made. However, there are several trends or tendencies that may be deduced. It appears for the depleted fuel, such as is analyzed for the 12-fuel-assembly loading, not modeling the grids appears to be conservative at unborated conditions but at some point the amount of soluble boron in the water would make it non-conservative to not model the grids. This type of behavior is consistent with other analyses the NRC staff has reviewed. The soluble boron concentration where that transition would occur appears to be greater than the soluble boron being credited in the accident analysis. Additionally, there is sufficient margin in the delta between the soluble boron being credited in the accident conditions and that required by the TSs to allow for a slight increase to offset the grids if necessary without challenging the regulatory limit. As discussed in Section 3.4.5 below, this is acceptable.

It appears for the fresh fuel, such as is analyzed for the 8-fuel-assembly loading, there may be a difference as to whether or not the pellet-to-clad gap is filled with water. During manufacturing the gap is filled with pressurized helium gas, but leaks in the fuel clad can allow water to enter the gap. If the gap is empty of water in the depleted fuel assembly, the fresh fuel seems to have the same performance as the depleted fuel. But if the gap is filled with water in a depleted fuel assembly, there may be a non-conservatism by not modeling the grids when the water is unborated. However, this potential non-conservatism appears to be small relative to the margin between the estimated k_{eff} in Table 4.7.2 and the regulatory limit for the 8-fuel-assembly loading. As such it is not of concern in this review. If the water in the gap contains soluble boron, there may be a non-conservatism by not modeling the spacer grids, however this non-conservatism again appears to be small relative to the available margin, and is, therefore, not of concern.

The comparisons discussed above also do not include the size and enrichments of the axial blankets. The analysis models axial blankets 6 inches long with a maximum enrichment of 3.20 weight percent (wt%) ^{235}U. The proposed TS Table 4.1.1-1 for IP2 and IP3 sets the limits on axial blankets that may be loaded into the STC. The staff concludes that the TS limit is sufficient to ensure that any non-conservatism due to axial blankets will be small relative to the available margin.

3.4.3.4.2 Fuel Assembly Mechanical Tolerances

The analysis in the SAR initially did not determine a reactivity uncertainty for fuel assembly manufacturing tolerances. This treatment of the fuel assembly manufacturing tolerances is different than that indicated in the NRC staff's guidance. In response to an NRC request for additional information, consideration of fuel assembly manufacturing tolerances was included in the analysis. The description of how the fuel assembly manufacturing tolerances were included in the analysis is in Section 4.7.5.2 of the SAR. The analysis considered the effect of the manufacturing tolerances on the depletion portion of the analysis as well as in the STC fuel basket. The analysis also determined and applied the uncertainties at each burnup/enrichment combination, rather than using a limiting uncertainty for all burnup/enrichment combinations on the loading curves. The manufacturing tolerances included are listed in Table 4.5.9 of the SAR. The uncertainty for each manufacturing tolerance was statistically combined with the uncertainties for the other manufacturing tolerances, and the result is reported in Tables 4.7.21 and 4.7.22 of the SAR.

The SAR analysis modeled all fuel pellets at a single density, using the nominal density increased by the uncertainty in the density, and as full right circular cylinders. Fuel pellet dishing and chamfering, and annular pellets, which have an outer ring with a hollow core, were not explicitly modeled in the estimation of k_{eff}. As these assumptions increase the amount of fuel present, one might conclude that these assumptions are always conservative. The sensitivity study on axial burnup profiles did model annular pellets. The results of that study are provided in Table 4.7.13 of the SAR. Table 4.7.13 indicates that when the axial blankets are explicitly modeled the annular pellets may actually be more reactive than the full pellets by as much as 0.001 Δk_{eff}. If the STC NCS analysis was modeling axial blankets in the estimation of k_{eff} this would warrant further review. However, since the sensitivity study indicates that for the STC it is conservative to model the fuel as uniformly enriched, which would not have annular pellets, the lack of annular pellet modeling is not an issue for this analysis. In contrast, if annular pellets may actually be more reactive than the full pellets, then potentially dishing and chamfering of the fuel pellet may be more reactive than the full right circular assumption; however, that effect would be much less than the effect of using annular pellets and would not affect the overall estimation of k_{eff}.

The applicant's analysis does not determine a reactivity bias or uncertainty for integral and fixed burnable poisons. The applicant's analysis does model integral burnable poisons during depletion. The manufacturing uncertainty on integral and fixed burnable absorbers could affect the post-irradiation reactivity by affecting the plutonium production in the fuel assembly. Section 4.7.9.5 of the SAR addressed this item. The NRC's review of this issue is provided in Section 3.4.3.7 of this safety evaluation.

3.4.3.5 Spent Fuel Characterization

For the STC criticality analysis, the fuel must be characterized appropriately. Characterization of fresh fuel is relatively straight forward. It is based primarily on ^{235}U 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 carry through to the spent nuclear fuel, and the standard practice has been to treat the uncertainties as unaffected by the depletion. The characterization of spent nuclear fuel is more difficult. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: a burnup uncertainty, the axial apportionment of the burnup, and the core operation that achieved that burnup.

3.4.3.5.1 Burnup Uncertainty

In the Kopp Letter, the NRC staff provided guidance for determining the burnup uncertainty: "A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. 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." Rather than use the guidance in the Kopp Letter, the method used in the SAR is to validate the depletion code by benchmarking, similar to the method used for the HI-STAR 100 transportation package. See Section 3.4.3.2.1 for a discussion of that validation.

3.4.3.5.2 Axial Apportionment or Burnup Profile

Another important aspect of fuel characterization is the selection of the burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped neutron 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 in the center section of the fuel assembly. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff}; generally, the under prediction becomes larger as burnup increases. This is known as the "end effect." Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," provides insights for selecting an appropriate axial burnup profile.

With respect to the axial burnup profile, the SAR did not use the axial burnup profiles from NUREG/CR-6801. The description of how the axial burnup profiles were derived is in Section 4.5.3 of the SAR. Simply put, the analysis took available profiles for Westinghouse 17x17 fuel assembly designs to derive the profiles and then performed a study to demonstrate the suitability of those profiles to IP3 Westinghouse 15x15 fuel assembly designs. The results of that study are discussed in Section 4.7.2.1 and Tables 4.7.13a, 4.7.13b, and 4.7.13c of the SAR. The information indicates the axial burnup profiles derived in the SAR bound those in NUREG/CR-6801 at equivalent burnup levels. Since the profiles used in this analysis result in k_{eff} values greater than the NUREG/CR-6801 axial burnup profiles, use of the profiles from the SAR are acceptable for use in estimating k_{eff} for the STC. It is stated in the SAR, "Note that the Westinghouse 17x17 profiles discussed above are from assemblies without axial blankets. Since assemblies without axial blankets were only used in earlier cycles, they have a much longer cooling time than the 5 years assumed in the design basis criticality analysis, which provides additional but unspecified margin." While the NRC staff concludes that the profiles used are acceptable, the staff does not concur that the cooling time of the fuel assemblies from which the axial profiles were derived provides any additional margin with respect to the profiles themselves.

3.4.3.5.3 Planar Burnup Distribution

Due to the neutron flux gradients in the reactor core, assemblies can show a tilted burnup distribution, i.e. differences in burnup between portions or quadrants of the cross section of the assembly. The SAR analysis considered the effect of planar burnup distribution on reactivity. The analysis concluded there is an increase in reactivity if fuel assemblies with planar burnup distribution are matched together. The increase could be on the order of the remaining analytical margin shown in Table 4.7.1 of the SAR, and therefore needs to be considered in the analysis. The description of how the planar burnup distribution was determined and applied is provided in section 4.7.2.2 of the SAR. The analysis in the SAR conservatively incorporates the effect of planar burnup distribution in determining k_{eff} for the STC.

3.4.3.5.4 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," provides a discussion of the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation systems, the basic principles with respect to the depletion analysis apply generically, since the phenomena occur in the reactor as the fuel is being used. The results have some translation to STC criticality analyses, especially given the similarity between the STC and storage/transportation systems. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron flux, resulting in maximum plutonium 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 flux resulting in maximum plutonium 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 pcm/°K. Thus a 10 °F change in moderator temperature used in the depletion analysis would result in 0.005 Δk_{eff}. The effects of each core operating parameter typically have a burnup or time dependency.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize plutonium production. The SAR analysis used temperatures similar to the maximum at which the plant could be operated as indicated by the applicant's UFSAR Table 3.2-4. The moderator and fuel temperature used is therefore acceptable.

For boron concentration, NUREG/CR-6665 recommends using a conservative cycle average boron concentration. The applicant's analysis used a cycle average soluble boron concentration of 900 ppm throughout the depletion of the fuel assemblies. This seems a little low, as the HI-STAR analysis indicates that the max for Westinghouse 15x15 was 950 ppm and the bounding was 1000 ppm. However, those values are more generic and the applicant has confirmed that 900 ppm is the maximum for IP3. The assumed boron concentration is therefore acceptable.

Specific power and operating history are related. Operating history is essentially just time spent at various specific power levels. Specific power is a second order effect compared to moderator temperature and soluble boron concentration. For actinide and fission product credit in burnup credit analysis, such as was modeled for the STC, NUREG/CR-6665 indicates lower specific powers are conservative. The SAR analysis used a specific power approximately 40% less than that indicated by the applicant's UFSAR Table 3.2-4. The specific power and operating history is therefore acceptable.

3.4.3.5.5 <u>Integral and Fixed Burnable Absorbers</u>

IP3 uses integral and fixed burnable absorbers. IP3 has used fixed burnable absorbers such as Burnable Poison Rod Assemblies (BPRAs) and Wet Annular Burnable Absorbers (WABA) for reactor operation. BPRAs are no longer being used. WABAs are the current and projected fixed burnable poison. IP3 also uses the Westinghouse integral fuel burnable absorbers (IFBA) for reactor operation. IP3 has also used WABA and IFBA in the same fuel assembly for reactor operation. In addition to the fixed burnable poisons, IP3 has also used hafnium flux suppressors. Hafnium flux suppressors are no longer being used. Typically pressurized-water reactors such as IP3 operate with all control rods fully withdrawn. However, the SAR indicates that for some early cycles IP3 operated with control rods inserted during power operation. IP3 is no longer operating with fully inserted control rods, but could operate with control rods inserted in accordance with the rod insertion limits in the TS.

The SAR uses a sensitivity study to compare the reactivity of the effects of combinations of integral and fixed burnable absorbers. That sensitivity study is discussed in Section 4.7.1.2.2 and the results are presented in Tables 4.7.6 and 4.7.7 of the SAR. In the sensitivity study the use of integral and fixed burnable absorbers were modeled as their actual lengths. However, for the estimation of k_{eff} they were modeled as fully covering the active fuel region. Modeling the use of integral and fixed burnable absorbers as fully covering the active fuel region includes a neutron absorber that is not physically present, and modeling it that way in the STC would be non-conservative. However, in the SAR the integral and fixed burnable absorbers are not credited in the estimation of the STC k_{eff}. No fixed absorbers are modeled as being in the STC. Any residual integral burnable absorber is not included in the estimation of STC k_{eff}. The sensitivity study found that the combined use of WABA and IFBA bounded the use of either alone. The sensitivity study also found a significant difference in reactivity between fuel assemblies that had a hafnium insert or control rod inserted at power and the WABA/IFBA combination. The applicant chose to separate these into two different loading configurations, establishing a different enrichment/burnup loading curve for each.

The WABA/IFBA combination enrichment/burnup loading curve is based on the WABA and IFBA descriptions in SAR Tables 4.5.5 and 4.5.1 respectively. An increase in the B^{10} loading would make the WABA/IFBA combination enrichment/burnup loading curve non-conservative.

The hafnium/control rod enrichment/burnup loading curve is based on the hafnium insert and control rod descriptions in SAR Tables 4.5.6 and 4.5.7 respectively. Table 3.1.2-1 of LCO 3.1.2 establishes binding limits for the burnup of fuel assemblies that contained hafnium inserts or were located under an inserted control rod bank. Based on these limits, the NRC review assumes that IP3 will not use hafnium inserts in the future and, therefore, the description of the hafnium inserts is static. Similarly the NRC's review assumes that IP3 will not operate with control rods inserted past the eight inches included in the analysis for an appreciable amount of time in the future. If IP3 violates either of these limits, then the hafnium/control rod enrichment/burnup loading curve is not valid for those fuel assemblies.

The applicant's analysis does not determine a reactivity bias and bias uncertainty for integral and fixed burnable poisons. The applicant's analysis does model integral burnable poisons during depletion. The manufacturing uncertainty on integral and fixed burnable absorbers could affect the post-irradiation reactivity by affecting plutonium production. Section 4.7.9.5 of the SAR addressed this item. The NRC staff's review of this issue is provided in Section 3.4.3.7 of this safety evaluation.

3.4.3.6 <u>Determination of Soluble Boron Requirements</u>

The applicant did not take credit for soluble boron in the STC during normal conditions. Therefore, the regulatory requirement given in 10 CFR 50.68(b)(4) is that the k_{eff} of the STC, 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. By considering the double contingency principle, the 2000 ppm of soluble boron that is required by the STC TS can be credited to ensure the STC k_{eff} does not exceed 0.95 during an accident, provided that the accident and a boron dilution are independent events. Since the NRC has concluded in Section 3.6 of this safety evaluation that a breach of the STC and loss of coolant/moderator will not occur during the postulated accidents, a boron dilution event in a closed and sealed STC need not be projected to occur.

The SAR considered the following accidents: abnormal temperature, dropped, mislocated, and misloaded fuel assemblies, and misalignment between the active fuel region and the neutron absorber.

The temperature sensitivity study in Section 4.7.6 of the SAR indicates that lower temperatures and full water density are more reactive. However, the sensitivity study truncated its analysis at 39.2 °F. While in transit from the IP3 SFP to the IP2 SFP the STC will be outside and subject to local weather, although shielded by the HI-TRAC. Therefore, the STC may actually be subjected to lower temperatures. While the moderator density will not increase with lower temperatures, the effect on nuclear cross-sections has not been evaluated. However, it is not expected that the water in the STC will freeze, due to the decay heat generated by the fuel assemblies and the large mass of the casks. The effect of any slight temperature decrease below 39.2 °F would be bounded by the misloading accident.

A dropped fuel assembly may land in any of several positions: across the top of the STC, into a storage cell already containing a fuel assembly, alongside the STC fuel basket, or into an empty cell. It is not possible for a fuel assembly to lay flat across the top of the STC fuel basket as the STC inside diameter is less than the length of a fuel assembly. The distance between the top of the active fuel region for fuel assemblies loaded into the STC fuel basket and the top of the fuel basket is greater than 12 inches, therefore any fuel assembly laying across the top of the fuel basket would essentially be decoupled (separated) from the fuel in the fuel basket. A fuel assembly dropped on top of an assembly already in a cell will have the active fuel regions separated by the end fittings, and the dropped fuel assembly will be surrounded by borated water; therefore no appreciable increase in reactivity is expected. Any impact on the subject fuel assemblies will have a minimal impact on reactivity. Dropping or mislocating a fuel assembly alongside the STC fuel basket is physically impossible because there is not enough room between the fuel basket and the STC wall. Dropping a fuel assembly in an empty storage cell would be no different than a misloaded fuel assembly. Because a dropped fuel assembly is expected to have only a minimal impact on reactivity, it would be bounded by the misloading accident.

The misloading of fuel assemblies was analyzed for Configurations 1 and 2. The analysis is described in Section 4.7.8.2 and the results are presented in Table 4.7.14 of the SAR. The Configuration 1 analysis considered two cases that are based on the current IP3 SFP inventory. Due to certain storage constraints in the IP2 SFP, the TS being issued with this amendment restrict which fuel assemblies can be transferred. Only fuel assemblies from IP3 cycles 1 through 11 are currently permitted to be transferred. However, this does not limit the scope of the potential misloading analysis as the incorrectly loaded fuel assemblies could be any of the fuel assemblies in the IP3 SFP, not just fuel assemblies from core cycles 1 through 11. Although the applicant considered the current IP3 SFP inventory, no consideration was given to possible increases in fuel assembly reactivity in the future. Based on the past 35 years of operation, the staff finds it unlikely that there will be significant fuel assembly reactivity changes in the future as that would entail changes in fuel assembly design. Nonetheless, the excess soluble boron in the STC provides a substantial safety margin. For example, the applicant addressed a concern about future fuel assemblies that may be discharged from the core before reaching the expected burnup. If the fuel assembly is significantly underburned, it may have a higher reactivity than the analyzed fuel assemblies. For any significantly underburned fuel assemblies permanently stored in the IP3 SFP, the applicant will place a blocking device on the storage cell to prevent inadvertent transfer. The Configuration 2 analysis considered misloading one fresh fuel assembly into an STC cell that is required to be empty. This is conservative as the TS prohibit fresh fuel in the SFP during loading operations, and a fresh fuel assembly at the maximum enrichment (5 weight percent U-235) is more reactive than a spent fuel assembly. The analysis re-evaluated the biases and uncertainties considering the presence of soluble boron in the STC. The analysis determined that Configuration 1B required the largest amount of soluble boron to maintain $k_{eff} \leq 0.95$ in the event of a misloading condition. With respect to the accident conditions the maximum amount of soluble boron required to maintain $k_{eff} \leq 0.95$ is 1053 ppm. The STC TS require at least 2000 ppm of soluble boron, so there is sufficient soluble boron present in the STC to prevent criticality in this situation.

Misalignment between the active fuel region and the neutron absorber plates on the cell walls could occur in the event of a tip-over accident. A tip-over accident, while considered unlikely, was not shown to be a non-credible event. In a tip-over accident the fuel assemblies' position within the storage basket relative to the neutron absorber would change. The SAR describes this analysis in Section 4.7.8.3 with the results presented in Table 4.7.25. The analysis re-evaluated the biases and uncertainties considering the presence of soluble boron and tip-over geometric arrangement in the STC. The analysis assumed soluble boron of 1025 ppm and calculated a k_{eff} below 0.9450 for the three storage configurations.

3.4.3.7 Margin Analysis and Comparison with Remaining Uncertainties

Section 4.7.9 of the SAR evaluates the margin in the analysis. The section includes an evaluation of additional conservatism in the analysis and an evaluation of items that may have been treated non-conservatively. Section 4.7.9.1 includes an evaluation of additional conservatism in the analysis due to the analysis not having modeled all neutron absorbing nuclides that are actually present in depleted fuel. The SAR indicates the margin this provides is on the order of 0.055 Δk to 0.065 Δk for a 5.0 wt% fuel assembly.

The applicant provided the results of five cases to support this determination. The cases varied with the neutron absorbing nuclides considered, the application of number density correction factors, and whether or not biases and bias uncertainties were included. The applicant's determination used the cases at the low k_{eff} end that modeled all known neutron absorbing nuclides with no correction factors and no or limited biases and bias uncertainties while the high k_{eff} end was a case that the modeled the smaller set of benchmarked neutron absorbing nuclides with correction factors applied and a more extensive set of biases and bias uncertainties. Thus, these cases involved more changes than just the modeled neutron absorbing nuclides.

The applicant's use of these cases ignores the effect that correction factors, biases, and bias uncertainties would have on cases involving all known neutron absorbing nuclides if these could be determined for all known neutron absorbing nuclides and not just the benchmarked neutron absorbing nuclides. The use of the correction factors, biases and bias uncertainties for the benchmarked neutron absorbing nuclides is necessary to ensure the actual k_{eff} is less than the regulatory limit at the regulation's prescribed 95 percent confidence and 95 percent probability level. By comparing a case with these components to those without, and then taking the delta as margin, means the estimated k_{eff} no longer meets the 95 percent confidence and 95 percent probability level requirement. Therefore, the staff believes the cases used to determine those values are inappropriate for that purpose.

While the staff agrees that the presence of the additional neutron absorbers that were not modeled provides some conservatism, determining just how much is problematic. There are more appropriate cases in the SAR analysis to make such estimations. The case at the low k_{eff} end would be the one that modeled all known neutron absorbing nuclides with no correction factors and limited biases and bias uncertainties while the high k_{eff} end would be the one that modeled the benchmarked neutron absorbing nuclides with no correction factors and the same limited biases and bias uncertainties. The difference between those two cases is approximately 0.02 Δk. While the staff considers this to be a more appropriate comparison it still ignores the potential for correction factors and fuller application of biases and bias uncertainties to reduce the difference. Since there are insufficient details in the report to fully evaluate the cases performed for the study, the staff believes it is prudent to further reduce the potential conservatism to account for these unknown factors. Even with that reduction, however, it is reasonable to conclude that there is approximately 0.01 Δk of conservatism in the SAR's estimation of k_{eff} for the 5.0 wt% fuel assembly.

Section 4.7.9.1 of the SAR also includes a list of items that are touted as "conservative or bounding assumptions". These items may or may not be conservative depending on the context and the controls in place for spent fuel storage. Any conservatism that may be present is unquantified and therefore unavailable to offset any potential non-conservatism identified in the NRC staff's review. Any change to these items would constitute a change in methodology, and a new justification for the revised assumption would be required.

Section 4.7.9 of the SAR also identified four items that were not included in the estimation of k_{eff}, but that would serve to increase the k_{eff}. Those items are Fuel Geometry Changes, External Reflection, Uncertainty in Inserts, and Control Rod Insertion to the bite position. Uncertainties for each were statistically combined. Each item is described in its own section with the summation in Section 4.7.9.6. These four items are estimated to increase k_{eff} by 0.0050 Δk.

Based on its review, the NRC staff estimates there is 0.0050 Δk margin available to offset any potential non-conservatism identified in its review of the analysis. This will be considered in the aggregate in section 3.4.5 of this safety evaluation.

3.4.4 Storage of IP3 Spent Fuel in the IP2 SFP

The IP3 spent fuel will be stored in the IP2 SFP until it is loaded into a cask for storage in the Indian Point ISFSI. While the IP3 spent fuel is in the IP2 SFP it must meet the IP2 TS for storage of spent fuel. Initially the applicant proposed storing the IP3 spent fuel just as if it were IP2 spent fuel, with no differentiation between the two. Rather than perform an NCS analysis, the applicant initially sought to justify its approach by a comparison of the IP3 and IP2 fuel assembly designs and IP3 and IP2 reactor operation. The applicant did not initially compare the IP3 fuel assembly designs and reactor operation to the IP2 SFP NCS design basis analysis. In response to the NRC staff's requests for additional information, the applicant compared the IP3 fuel assembly designs and reactor operation to the IP2 SFP NCS design basis analysis. In that comparison it was discovered that the IP2 SFP NCS design basis analysis does not fully bound the IP3 fuel assembly designs and reactor operation. Some of the areas where the IP2 SFP NCS design basis analysis is not bounding are: moderator temperature during depletion, use of integral and fixed burnable absorbers, and the use of hafnium inserts. For the IP2 SFP the applicant took credit for soluble boron in the SFP during normal conditions. Accordingly, pursuant to 10 CFR 50.68(b)(4), the k_{eff} of the SFP must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, when flooded with borated water, and the k_{eff} of the SFP must not exceed 1.0, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

To account for the IP2 SFP NCS not fully bounding the IP3 fuel assembly designs and reactor operation, the applicant initially proposed the following license conditions:

1. Only fuel assemblies from IP3 Cycles 1 through 11 will be transferred.
2. Only fuel assemblies with a ^{235}U enrichment ≥ 3.2 and ≤ 4.4 wt% will be transferred.
3. The IP3 fuel will only be stored in IP2 SFP Region 1-1 and Region 1-2.

With these limitations, the applicant evaluated the storage in the IP2 SFP of the specified population of spent fuel, based on the actual conditions experienced by that spent fuel.

The IP2 SFP Region 1 storage racks are a flux trap design that were built with Boraflex inserts in the cell walls for additional neutron absorption. The Boraflex has degraded over time. Based on operating experience, IP2 SFP Region 1-1 is conservatively assumed to have lost all of its Boraflex, and Region 1-2 is assumed to have lost half of its Boraflex.

In the design basis analysis, IP2 SFP Region 1-1 was analyzed for spent fuel storage and enrichment/burnup loading requirements were determined based on the rack design without Boraflex. IP2 SFP Region 1-1 enrichment/burnup loading requirements are listed in current IP2 TS Figure 3.7.13-3. With the assumed 50% Boraflex still in place, IP2 SFP Region 1-2 was analyzed for fresh fuel storage. The IP2 SFP NCS design basis analysis concluded that IP2 SFP Region 1-2 could store fresh fuel with a nominal ^{235}U weight percent (wt%) enrichment of 4.5 in all cells, but enrichments over 4.5 wt% required IFBA to be present in the fuel assembly. IP2 SFP Region 1-2 enrichment/IFBA loading requirements are listed in current IP2 TS Figure 3.7.13-4.

The applicant determined that while the moderator temperature used during the IP2 SFP NCS design basis analysis depletion does not bound the maximum IP3 allowable, it does bound the actual maximum moderator temperature that IP3 Cycles 1 through 11 experienced.

The IP2 SFP NCS design basis analysis did model integral and fixed burnable absorbers, however it did not model the combined use of integral and fixed burnable absorbers in the same fuel assembly, nor did it model the use of hafnium inserts. As shown in the STC NCS, these are more limiting than the use of integral and fixed burnable absorbers independently. The applicant estimated a reactivity penalty of approximately 0.02270 Δk for the IP3 Cycles 1 through 11 spent fuel to account for the combined use of integral and fixed burnable absorbers in the same fuel assembly or the use of hafnium inserts. The applicant's evaluation partially offset this with a 0.01 Δk penalty from the IP2 SFP NCS design basis analysis for the use of fixed burnable absorbers. According to the applicant's estimation this left approximately 0.01270 Δk additional penalty to be applied to the IP3 spent fuel. To offset this penalty the applicant estimated the reactivity margin of the actual IP3 Cycles 1 through 11 spent fuel burnup in comparison to IP2 SFP Region 1-1 and Region 1-2 enrichment/burnup loading requirements. The applicant estimates that the IP3 Cycles 1 through 11 spent fuel has a minimum of 10 GWd/MTU of burnup more than the IP2 SFP Region 1-1 and Region 1-2 enrichment/burnup loading requirements listed in the current IP2 TS Figure 3.7.13-3, and that this provides approximately 0.03086 Δk of margin. By the applicant's estimate, this leaves approximately 0.01816 Δk of margin.

The NRC staff has considered the IP2 SFP NCS design basis analysis. The NRC staff noted that the IP2 SFP NCS design basis analysis 0.01 Δk penalty for the use of fixed burnable absorbers was not applied to the estimation of k_{eff} in the unborated condition, and is therefore unavailable to offset any portion of the approximately 0.02270 Δk IP3 penalty mentioned previously. The NRC staff also noted that the analysis contains the same or similar issues as identified in NRC Information Notice 2011-03, "Nonconservative Criticality Safety Analyses For Fuel Storage" (Reference 6). Additionally, NRC staff noted that the IP2 SFP NCS design basis analysis contains several other items that were only considered in the estimation of k_{eff} in the borated condition and not the unborated condition. Because the IP2 SFP Region 1-1 is a spent fuel analysis and Region 1-2 is a fresh fuel analysis, not all items apply to both Regions. Additionally, the IP2 SFP NCS design basis analysis estimated a Region 1-2 unborated k_{eff} significantly lower than the 10 CFR 50.68(b)(4) requirement of a k_{eff} of 1.0.

With these issues in mind, the NRC staff has considered the margin or lack thereof in the IP2 SFP NCS design basis analysis for IP2 SFP Region 1-1 and Region 1-2. Based on that consideration it appears that the IP2 SFP NCS design basis analysis may not demonstrate compliance in Region 1-1 with the regulatory requirement in 10 CFR 50.68(b)(4) for k_{eff} to "...remain remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water," which is a requirement when credit has been previously taken for soluble boron.

Therefore, the NRC staff has determined it cannot approve the storage of IP3 spent fuel in the IP2 SFP Region 1-1 based on the IP2 SFP NCS design basis analysis enrichment/burnup loading requirements determined therein and listed in the current IP2 TS Figure 3.7.13-3.

Since the IP2 SFP NCS design basis analysis for Region 1-2 had a sufficiently low k_{eff} it was able to accommodate the issues described above. With the issues applied to Region 1-2, the analysis appears to retain approximately 0.038 Δk of margin to the regulatory requirement. Also, the IP3 fuel that will be transferred under this license amendment will have a nominal initial enrichment less than that determined as acceptable for storage without IFBA in Region 1-2 and that fuel will have significant burnup. While a precise estimate cannot be made for that margin, a conservative estimate would be the approximately 0.03086 Δk of margin the applicant estimated for the extra burnup the IP3 spent fuel possesses over the IP2 Region 1-1 enrichment/burnup requirement. These provide approximately 0.06886 Δk of margin during normal conditions with the SFP flooded with unborated water. However, this is based on the assumption that the Boraflex credit model and degradation monitoring are appropriate. Recent staff experience with Boraflex credit modeling and degradation monitoring raise a concern with this assumption. The NRC staff concludes that the 0.06886 Δk of margin would cover the potential concerns regarding Boraflex, and the NRC staff concludes that there is reasonable assurance that storage of the limited IP3 spent fuel population in IP2 SFP Region 1-2 will meet the regulatory requirement in 10 CFR 50.68(b)(4) for k_{eff} to "...remain remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water".

During accident conditions, Boraflex that has suffered 50% degradation may be thin enough that it is possible that during a seismic event pieces of Boraflex could slide past one another and settle to the bottom of the Boraflex enclosure leaving the topmost portion of the fuel assemblies in an essentially unpoisoned storage rack. In this accident, IP2 SFP Region 1-2 essentially becomes Region 1-1 in the normal condition borated analysis. Nonetheless, despite the apparent short-comings in the unborated portion, the IP2 SFP NCS design basis analysis does appear to cover these issues, in determining whether the requirement for k_{eff} "...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" is met in the soluble boron credit portion for Region 1-1. The IP2 SFP NCS design basis analysis indicates Region 1-1 requires 786 ppm of soluble boron under normal conditions to meet the regulatory requirement. While the double contingency principle indicates multiple independent accidents do not have to be analyzed, a seismic event may also result in a loss of SFP cooling, giving these events a common cause, and therefore requiring analysis. The IP2 SFP NCS design basis analysis indicates Region 1-1 requires an additional 110 ppm of soluble boron under abnormal conditions to meet the regulatory requirement that k_{eff} must not exceed 0.95 if flooded with borated water. This would bring the total soluble boron requirement to 896 ppm. The current IP2 TS 3.7.12 requires the IP2 SFP soluble boron concentration to meet or exceed 2000 ppm. This provides significant margin, supporting the staff's conclusion that the regulatory requirement is met in Region 1-2 during normal conditions with the SFP flooded with borated water.

Therefore, the NRC staff approves the storage of IP3 spent fuel in IP2 SFP Region 1-2 based on the IP2 SFP NCS design basis analysis enrichment/burnup loading requirements determined therein and listed in IP2 TS Figure 3.7.13-4 and the IP3 spent fuel limitations discussed above.

3.4.5 Conclusion

The NRC staff's review of the STC nuclear criticality safety analysis documented in the SAR identified several non-conservative considerations. Those items were evaluated against the stated margin and the regulatory limit, and what the NRC staff believes to be an appropriate amount of margin attributable to the isotopic compositions modeled in the analysis compared to the actual isotopic compositions of the depleted fuel. Since not all neutron absorbers were modeled, there is some margin associated with the non-modeled absorbers. Most of the comparisons in the SAR were performed with a ^{235}U 5.0 wt% enriched fuel assembly with 50 GWD/MTU of burnup. This modeled assumption does not exactly represent any burnup/enrichment combination used to develop a loading curve for the STC, but is reasonably close to the ^{235}U 5.0 wt% enriched fuel assembly with 46.4 GWD/MTU of burnup used to develop loading curves for the STC. While the actual values may be expected to vary with enrichment and burnup, and while the exact variation is unknown, using the Configuration 1A selection of a 5.0 wt% enriched fuel assembly with 46.4 GWD/MTU burnup provides a reasonable estimate of whether the SAR analysis demonstrates compliance with the regulatory limit, as that fuel assembly starts with the lowest margin to the regulatory limit.

As previously noted, it appears the applicant's analysis did not always use the most reactive fuel assembly. Table 4.7.5 in the SAR indicates that for the Configuration 1A 5.0/46.4 combination, a Type 2 fuel assembly may be more reactive by as much as 0.00140 Δk. Also, as previously noted it appears the analysis did not consider the uncertainty in the STC fuel basket tolerance bias. Table 4.7.9.a in the SAR indicates that for the Configuration 1A 5.0/46.4 combination, the STC fuel basket bias may be larger by as much as 0.00090 Δk.

In the aggregate with these items considered, the SAR Table 4.7.1 estimation of k_{eff} for the Configuration 1A 5.0/46.4 combination would not provide reasonable assurance that k_{eff} is ≤ 0.95 at a 95 percent probability, 95 percent confidence level (95/95 level), if flooded with unborated water. However, with consideration of the margin available due to the conservatism in the analysis due to not having modeled all neutron absorbing nuclides that are actually present in the depleted fuel, the NRC staff concludes that the SAR analysis and subsequent TS controls provide reasonable assurance that the STC meets the regulatory requirement during normal operations for k_{eff} to be ≤ 0.95 at a 95 percent probability, 95 percent confidence level (95/95 level), if flooded with unborated water.

In the accident analysis, the STC NCS estimates that 1053 ppm of soluble boron is needed for k_{eff} to be ≤ 0.95 at a 95 percent probability, 95 percent confidence level (95/95 level), if flooded with borated water. The STC TS require 2000 ppm of soluble boron. The NRC staff concludes that the SAR analysis and subsequent TS controls provide reasonable assurance that the STC meets the regulatory requirement for accident conditions for k_{eff} to be ≤ 0.95 at a 95 percent probability, 95 percent confidence level (95/95 level), if flooded with borated water.

The IP2 SFP NCS design basis analysis does not fully bound the IP3 fuel assembly designs and reactor operation. Based on the NRC staff's review of the IP2 SFP NCS design basis analysis, the IP3 spent fuel is acceptable for storage in the IP2 SFP provided the following limitations are met:

1. Only fuel assemblies from IP3 Cycles 1 through 11 will be transferred.
2. Only fuel assemblies with a ^{235}U enrichment ≥ 3.2 and ≤ 4.4 wt% will be transferred.
3. The IP3 fuel will only be stored in IP2 SFP Region 1-2.

These conditions have been placed in the IP2 and IP3 licenses. With these license conditions in place, there is reasonable assurance that the LAR complies with the requirements of GDC-62 and 10 CFR 50.68.

3.5 Thermal-Hydraulic Evaluation

The NRC staff reviewed Chapter 5, "Thermal-Hydraulic Evaluation" of licensing report HI-2094289 (SAR) and associated thermal calculation report HI-2084146 "Thermal Hydraulic Analysis of IP3 Shielded Transfer Cask" to determine cask design compliance with the thermal-hydraulic requirements of GDC 61 of Appendix A to 10 CFR Part 50. The staff also audited some of the supporting thermal analysis files.

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 Holtec International Final Safety Analysis Report for HI-STORM 100 Cask System, Revision 7, dated August 9, 2008, 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 SAR Chapter 3, 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.

3.5.1 Thermal Design

The on-site fuel transfer casks consist of the STC normally situated inside a vertically oriented transfer cask (HI-TRAC), both equipped with bolted lids. The spent nuclear fuel (SNF) assemblies reside inside the STC. The STC contains a stainless-steel honeycomb fuel basket with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies. The fuel basket panels are equipped with neutron absorbing panels sandwiched between a stainless steel sheathing plate and the fuel basket panel, along the entire length of the active fuel region. The STC is water filled, which emulates the wet storage environment in the IP2 and IP3 SFPs. In this manner fuel temperature excursions during loading, transfer and unload-to-pool are minimized. Within the STC the water environment sustains a closed loop thermosiphon action, removing SNF heat by an upward flow of water through the storage cells. Thermosiphon action is defined as buoyancy induced global circulation of water in the STC. The STC is externally cooled by the water filled HI-TRAC annulus. The HI-TRAC annulus is cooled by the so-called "Rayleigh effect" defined as natural circulation in differentially heated cavities. The Rayleigh effect transports heat laterally across the HI-TRAC annulus. The heat reaching the HI-TRAC inner surface is transmitted laterally across the HI-TRAC steel-lead-steel body by conduction. The HI-TRAC annulus is equipped with an aluminum centering assembly principally engineered to cushion the STC under a hypothetical tipover accident. The HI-TRAC body is water jacketed to provide neutron shielding and to dissipate heat by natural circulation. The HI-TRAC is externally cooled by radiative heat transfer and natural convection heat transfer to the environment.

To protect the casks from excessive hydraulic pressures, an air space is provided under the HI-TRAC lid and a steam-filled space is provided under the STC lid. The minimum heights of the open spaces are 7.5 inches for the STC lid and 9.3 inches for the HI-TRAC lid.

The NRC staff reviewed 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.

3.5.2 Thermal Properties of Materials

Materials present in the STC are fuel assemblies (primarily uranium dioxide and zirconium alloy with relatively small amounts of fission products and helium gas inside the fuel rods), carbon steel, stainless steel, Metamic (boron carbide particles in an aluminum matrix), lead, steam, and water. Materials present in the HI-TRAC transfer cask are carbon steel, lead, air, water, and aluminum. The SAR provided material thermal properties such as thermal conductivity, density, specific heat, viscosity, and emissivity. The NRC staff found these properties to be accurate for the materials specified. The applicant specifies the natural convection heat transfer coefficient as a function of the product of Grashof and Prandtl numbers. This product is a function of length scale, surface-to-ambient temperature difference, and air properties. The thermal properties used for the analysis of the package (the STC placed inside the HI-TRAC) were appropriate for the materials specified.

The NRC staff reviewed the thermal properties used for the package analyses and determined that they were appropriate for the materials specified.

3.5.3 Thermal Evaluation of Fuel Transfer Operation

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

Condition	Value
Maximum decay heat per fuel assembly	1105.2 watt (W) (four interior cells) 650 W (eight peripheral cells)
Total decay heat	9.621 kW
Ambient temperature	100°F
Solar insolation	As specified in 10 CFR Part 71

3.5.3.1 Description of the 3-D Thermal Model

The applicant used the Fluent program to evaluate the thermal performance of the STC. Fluent is a finite volume computational fluid dynamics (CFD) program with capabilities to predict fluid flow and heat transfer phenomena in two and three dimensions (3-D). The applicant developed a 3-D Fluent CFD model of the STC and the transfer cask. The composite cell walls (made up of stainless steel panels, neutron absorber panels and stainless steel sheathing) were modeled as a homogeneous panel with equivalent orthotropic (thru-thickness and parallel plates direction) thermal conductivities. The fuel rods and the interstitial water are replaced with an equivalent square homogeneous section characterized by an effective thermal conductivity and porous media flow resistance factor. The rest of the STC components (including the basket cross section, cutouts at the bottom of the basket wall to allow water circulation, top plenum, and downcomer flow passages) were individually modeled in the 3-D analysis. The HI-TRAC annulus, steel-lead-steel layers, top lid, bottom pool lid, and water jacket were individually modeled. Heat dissipation by the aluminum centering assembly placed in the HI-TRAC annulus was neglected when the HI-TRAC annulus is filled with water.

The applicant performed a grid sensitivity study on the water jacket region, annulus region between the STC and the HI-TRAC and the grid size in the axial direction in the fuel region. Three meshes were used in the grid study, with mesh 1 being the coarsest and mesh 3 the finest. All sensitivity analyses were performed for the case with design basis maximum heat load. Based on the grid sensitivity study, the applicant determined that the thermal solution is quite sensitive to the grid density in the annulus region. The grid sensitivity results showed that mesh 3 was reasonably converged and mesh 1 predicted the highest peak cladding temperature. Convergence means that subsequent iterations of the model calculations produce essentially the same answer, showing that the initial assumptions and repeated calculations lead to a stable result. Based on results from the grid sensitivity study, mesh 1 was used for all steady state calculations, while mesh 3 was used for all transient calculations to predict more accurate results.

3.5.3.2 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. As indicated above, mesh 1 is used 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 from SAR Table 3.2.1.

3.5.3.3 Evaluation of the STC Without the HI-TRAC

The STC is moved without the HI-TRAC when lifting it into and out of the spent fuel pools. There is the possibility that a crane malfunction could result in keeping the STC in this configuration for an extended period of time. The evaluation of the bare STC with the maximum heat load shows that the average outer surface temperature of the STC is 83°C (181°F). During on-site transfer operations, the average outer surface of the STC located within the HI-TRAC is 91°C (196°F). The bare STC components and cavity temperatures will, therefore, be lower than those reported in SAR Table 5.3.1. The evaluation of the bare STC with the maximum heat load bounds the configurations of a loaded STC, with or without the STC lid, being moved in the FSB outside of the HI-TRAC.

3.5.3.4 Detection of Misloaded Fuel With High Decay Heat

During its review of the LAR, the NRC staff was concerned that if a fuel assembly with a higher decay heat than the design basis decay heat was loaded in the STC, it would invalidate the thermal evaluations and possibly result in rupture of the STC, as there are no pressure relief valves. Therefore, the applicant has implemented a test for a possible misloaded fuel assembly with high decay heat. The misload test requires the pressure inside the STC to be monitored for a minimum duration of 24 hours after the STC lid is torqued, the open space above the water is filled with steam, and the STC lid vent valve is closed. Pressure monitoring is adopted to conduct the fuel misload test because the vapor pressure of water rises sharply with temperature, thereby providing a sensitive means to detect gross fuel misloads. A transient calculation is performed using the Fluent thermal model defined in SAR Section 5.3.1 and the STC pressure rise is computed. The 48-hour STC pressure rise under design basis heat load is graphed in SAR Figure 5.3.2. The STC pressure rise is computed as the increase in STC pressure against the initial STC pressure (at time = 0). The rate of change of STC pressure with time for design basis heat is graphed in SAR Figure 5.3.4. From the figure, the maximum permissible rate of change of STC pressure with the design basis heat load was determined and is reported in SAR Table 5.3.3. SAR Figure 5.3.4 shows that the rate of change of STC pressure specified in SAR Table 5.3.3 can be used to differentiate between a design basis heat load and a severe misload during the STC pressure rise surveillance.

3.5.3.5 NRC Staff Conclusions

The NRC staff reviewed the description of the 3-D thermal model and concludes that the description provides sufficient details to make a determination of the adequacy of the model. The NRC staff finds that the model is acceptable for this application. The staff reviewed the calculated maximum temperatures during typical fuel transfer conditions and with the bare STC using the design basis decay heat, and determined the predicted temperatures are below the allowable material limit. The staff also reviewed the applicant's approach for detecting the misload of fuel assemblies with decay heat greater than permitted and found it acceptable.

3.5.4 Thermal Evaluation During Accident Conditions

The applicant performed the evaluation of several postulated accidents during transfer operations to demonstrate that the STC will remain within design basis conditions. The following criteria are demonstrated to ensure material integrity:

1) The fuel cladding must remain below the ISG-11 temperature limit guidance.
2) The STC vessel temperature and pressure must remain below accident limits.
3) HI-TRAC pressure boundary temperature and pressure must remain below accident limits.

The NRC staff's evaluation of postulated accidents (rupture of the HI-TRAC water jacket, 50-gallon transporter fuel tank rupture and fire, simultaneous loss of water from the water jacket and HI-TRAC annulus, fuel misload, hypothetical tipover, and crane malfunction) is provided below.

3.5.4.1 HI-TRAC Jacket Water Loss

For a bounding analysis, the applicant assumed that all HI-TRAC water jacket compartments are drained of water and replaced with air. Heat dissipation by conduction and radiation in the air space is included in the analysis. The HI-TRAC is assumed to have the maximum thermal payload (SAR Table 5.0.1) and assumed to have reached steady state maximum temperatures. Under this array of adverse conditions, the maximum temperatures and pressures are computed and reported in SAR Tables 5.4.1 and 5.4.2. The staff reviewed the analysis and assumptions and confirms from the results of jacket water loss evaluation, the cladding, STC and HI-TRAC component temperatures are below design limits and the co-incident STC and HI-TRAC pressures are bounded by the vessel accident pressure limits.

3.5.4.2 Fire

The fire event is defined as rupture of an on-site transport vehicle fuel tank filled to capacity and ignition of spilled fuel. The applicant stated that the fuel tank capacity is limited to 50 gallons. The fuel tank fire is conservatively assumed to surround the HI-TRAC in the manner described in 10 CFR Part 71. All exposed transfer cask surfaces are heated by radiation and convection heat transfer from the fire. Guidance from NUREG-1536 and 10 CFR Part 71 conservatively bounds the consequences of the postulated fire event. The staff reviewed the analysis and assumptions and confirms that the fuel cladding temperature is within the SFST-ISG-11 limits (SAR Table 3.1.1). The maximum temperatures of the basket structural materials are within design limits (SAR Table 3.1.1). The maximum temperature of the METAMIC neutron absorber is within design limits (SAR Table 3.1.1). The maximum temperatures of the STC pressure boundary materials are within design limits (SAR Table 3.1.1). The maximum STC and HI-TRAC pressures are within design limits (SAR Table 3.2.1).

3.5.4.3 Simultaneous Loss of Water from the HI-TRAC Water Jacket and HI-TRAC Annulus

The integrity of fuel cladding and STC pressure boundary integrity is evaluated under a postulated simultaneous loss of water from the water jacket and HI-TRAC annulus. The HI-TRAC is equipped with an array of water compartments filled with water. For a conservatively bounding analysis, the applicant assumed that all water jacket compartments and the HI-TRAC annulus are drained of water and replaced with air. The HI-TRAC is assumed to have the maximum thermal payload (SAR Table 5.0.1) and assumed to have reached steady state maximum temperatures. Under this array of adverse conditions, the maximum temperatures and pressures are computed and reported in SAR Tables 5.4.5 and 5.4.6. The staff reviewed the analysis and assumptions and confirmed the spent fuel cladding, STC and HI-TRAC component temperatures remain below design limits and the co-incident STC pressure is bounded by the vessel accident pressure limit.

3.5.4.4 Fuel Misload

To provide assurance that the STC integrity is not challenged, the applicant defined a hypothetical misload as an event when every storage location is loaded with fuel generating two times the maximum permitted heat load. The misload event is evaluated with the STC placed in the HI-TRAC, the STC lid vent valve closed, and an assumption that maximum steady state temperatures and co-incident pressures are reached. Under this array of adverse conditions, the maximum temperatures and pressures are computed and reported in SAR Tables 5.4.7 and 5.4.8. The staff reviewed the analysis and assumptions and confirms that the cladding, STC and HI-TRAC component temperatures under this adverse fuel misload condition remain below design limits and the co-incident STC pressure is bounded by the vessel accident pressure limit.

3.5.4.5 Hypothetical Tipover

The tipover accident is defined in SAR Chapter 3 as a non-mechanistic event to demonstrate structural robustness of the STC. The applicant analyzed this event to evaluate the thermal design of the STC to provide protection of the loaded fuel. The staff reviewed the analysis and assumptions and confirmed that the cladding, STC and HI-TRAC component temperatures remain below design limits (SAR Table 3.1.1) and the co-incident STC and HI-TRAC pressures are bounded by the accident pressure limits (SAR Table 3.2.1).

3.5.4.6 Crane Malfunction

The applicant postulated a crane malfunction accident as an event wherein the IP3 crane stops operation for an extended duration co-incident with a fuel misload error while the loaded STC is out of the SFP but not yet in the HI-TRAC. The co-incident fuel misloading is defined as a condition wherein all fuel storage locations are loaded with fuel generating two times the maximum permitted heat load. Under this scenario the STC is assumed to be initially flooded with 100°F pool water. This assumption reasonably bounds pool water temperature during post-outage fuel loading operations. As an additional measure of conservatism the STC is assumed to be insulated and the water subjected to adiabatic heating. A significant pressure rise under the postulated crane malfunction event is not credible because the STC lid vent is open to atmosphere and the lid is positioned unbolted above the STC with a small gap. However, to avoid boiling and loss of water inventory, the applicant calculated a minimum available time to implement corrective actions prior to the STC reaching boiling temperature. The minimum time (time to boil) and decay heat are 17.8 hours and 19.2 kW, respectively. This evaluation provides reasonable assurance that plant operators have adequate margins to correct a crane malfunction or implement steps to manually lower the STC into the pool. The staff reviewed the analysis and assumptions and agrees that calculated time to boil is adequate for correcting the crane malfunction or implementing steps to manually lower the STC into the pool (if needed).

SAR Tables 5.4.1 through 5.4.10 include the maximum temperatures and pressures predicted for each of the postulated accident events. The fire and crane malfunction accident analyses are based on transient calculations. The other postulated accidents are assumed to reach thermal equilibrium. SAR tables 5.4.1 through 5.4.10 show that the calculated maximum temperatures and pressure are all below the allowable accident limits.

The NRC staff reviewed the applicant'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.

3.5.5 Confirmatory Analysis

The NRC staff reviewed the applicant'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 applicant'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 assured that the applicant performed appropriate sensitivity analysis calculations to obtain mesh-independent results that would provide bounding predictions for all analyzed conditions during normal fuel transfer and accidents. The staff also independently performed a number of sensitivity calculations. Finally, through a request for additional information (RAI), the staff received the information needed to make a safety determination on the adequacy of the fuel transfer thermal design.

3.5.6 Conclusion

Based on the applicant's thermal evaluation and the staff's review of the statements and analysis methods employed in the application, the NRC staff concludes that the thermal aspects of the proposed design have been adequately described and meet the regulatory requirements of 10 CFR Part 50.

3.6 Structural Evaluation

3.6.1 Introduction and Regulatory Evaluation

Chapter 6 of the SAR identifies and describes the structural components of the STC. The objective 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.

This LAR is a first-of-a-kind application, as this is the first fuel cask that an applicant has 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 structures, systems, and components for different portions of the LAR.

3.6.2 Structural Design and Structural Evaluation Details

A detailed description of the STC and HI-TRAC is provided in Section 1.3 of the SAR. The STC is a newly designed component for this LAR; the HI-TRAC transfer cask is an existing piece of equipment that is part of the HI-STORM 100 Dry Cask Storage System that is certified by the NRC under docket 72-1014, and is currently in use at Indian Point. The licensing drawings for the STC and HI-TRAC 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 Section 1.3.1 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). The NRC staff questioned the applicant about the differences between Subsection ND and NB or NC (refer to RAI Response 8-4 in Entergy's letter dated July 28, 2011). The NRC staff determined that

ASME Code, Section III, Division 1, Subsection ND is adequate and applicable based on the applicant's code reconciliation between Subsection ND and NC and post-fabrication examination and acceptance testing. The list of ASME Code Alternatives for the STC is tabulated in the IP2 and IP3 licenses, in Appendix C, Table 4.1.3-1.

As previously mentioned, the HI-TRAC transfer cask, which was certified by the NRC under docket 72-1014, has already been in use for IP2 dry cask storage operations. However, the dry cask storage system uses a vented lid on the HI-TRAC, which prevents the development of internal pressure. The applicant has provided information in the application to demonstrate that the HI-TRAC, with the new solid lid design, meets the design criteria listed in ASME Code, Section III, Division 1, Subsection ND.

In January 2012, the applicant stated that a second HI-TRAC transfer cask will be built for the transfer of spent fuel assemblies from IP3 to IP2. The new HI-TRAC will be built to the same specifications as those stated in the SAR. Therefore, either the existing or a new HI-TRAC transfer cask will be used in the proposed fuel transfer.

The STC and/or HI-TRAC systems were analyzed against 9 different load cases, as stated in Section 6.2 of the SAR and discussed below.

3.6.2.1 Load Case 1: Design Pressure

The STC normal pressure limit is 50 psig, while the accident pressure limit is 90 psig. The applicant'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, by use of Appendix I of the ASME Code, the number of cycles (conservatively estimated at 500) and the corresponding stress amplitude due to fatigue, would not be sufficient to approach conditions for a fatigue failure.

Further, the NRC staff noted that per Table 8.5.1 of the SAR, the STC closure bolts will be replaced every 240 bolting cycles, so fatigue will not be an applicable failure mechanism.

The HI-TRAC normal pressure limit is 30 psig, while the accident pressure limit is 50 psig. The pressure stress loading is bounded by the normal lifting and handling (which also includes normal pressure loading) operations.

3.6.2.2 Load Case 2: Normal Operating Pressure Plus Temperature

The STC is filled with borated water, and the annulus region between the STC and HI-TRAC is filled with unborated water, to enhance heat transfer capability and radiation shielding. 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 pressure plus the effect of temperature is determined to be minimal, and no significant thermal stresses will be generated.

3.6.2.3 Load Case 3: Normal Handling

All applicable lift points on the STC and HI-TRAC have been designed to satisfy the requirements of ANSI N14.6 and NUREG-0612. Prior to initial use, the STC will be tested to

300% of the maximum design lifting load of 80000 lb 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 by ANSI N14.6, per Section 8.5.3.3 of the SAR.

The applicant 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 (CMAA) specifications. The applicant analyzed the HI-TRAC lifting trunnions and structures as reported in Section 3.4.3 of the HI-STORM 100 FSAR (which was previously approved by the NRC) for a bounding weight of 200,000 lb (i.e., 10,000 lb greater than the 190,000 lb for a HI-TRAC with a fully loaded STC inside).

The maximum gross weight of the fully loaded STC is 40 tons. The applicant 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 of Holtec Report HI-2084118 (proprietary). The applicant provided the detailed STC trunnion and STC closure lid lifting analysis in Appendix A of HI-2084118, "Shielded Transfer Canister Structural Calculation Package" (proprietary). The detailed STC baseplate and closure lid stress analysis are provided in Appendix B of HI-2084118.

The HI-TRAC pool bottom lid and top lid lifting (plus internal pressure) structural analysis is provided in Appendix C of HI-2084118.

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 have acceptable safety margins with conservatism (using dynamic load factors and following appropriate codes and standards).

3.6.2.4 Load Case 4: Fuel Assembly Drop

A fuel assembly drop accident analysis was performed by the applicant in order to determine the effect of a 2000 lb fuel assembly dropping 36 inches onto the STC fuel basket during handling operations in the spent fuel pool. 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 the fuel will remain in a subcritical configuration. Details of the analysis are provided in Appendix D of HI-2084118 (proprietary).

3.6.2.5 Load Case 5: HI-TRAC Vertical Drop Accident

A vertical drop accident analysis was performed by the applicant to ensure that a loaded STC, inside the HI-TRAC, will survive a 6 inch drop when the HI-TRAC is being lifted by the VCT. The analysis was done in LS-DYNA. The applicant provided details of the analysis in HI-2094345, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation" (proprietary). The acceptance criteria provided was determined to be acceptable. The analysis results conclude that the loaded STC and HI-TRAC 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 (with a loaded STC inside) is not permitted to be lifted above 6 inches unless certain conditions have been met (e.g. a N14.6 lifting device is being used which has redundant drop protection features). Therefore, the 6 inch drop is a bounding drop height.

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

The applicant analyzed a VCT loaded with a HI-TRAC with a seismic event equal to the Indian Point site design basis earthquake. The approach used to analyze the systems is consistent with what was done in the HI-STORM 100 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 tipover during a design basis earthquake (DBE), and will remain stable during operations. Note that a tipover analysis (Load Case 9) was also done, and conservatively concluded that in the event of a tipover, a loaded STC and HI-TRAC will remain below acceptable stress/deceleration limits.

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

In Section 6.2.7 of the SAR, the applicant analyzed the STC sitting on the floor of the SFP with no crane attached, with a seismic event equal to the Indian Point design basis earthquake (DBE). The IP2 and IP3 SFPs are each one large pool, with one corner designed for placing the spent fuel transfer cask. The concern is that the STC could tip over 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 which have been loaded into the STC. The applicant used the criteria for static equilibrium to show that the STC will not slide or tip over in the SFP during the DBE.

During the course of its review, the NRC staff was initially unable to verify if the DBE is applicable to the loading/unloading zones of the fuel transfer. By letter dated April 20, 2010 (Accession No. ML101020486), the staff requested that the applicant, by RAI 6-7, define and evaluate the applicable seismic stability loading and unloading conditions for the loading/securing of the loaded STC and HI-TRAC onto the VCT, as well as the loading/securing of the STC onto the HI-TRAC. By letter dated October 5, 2010, the applicant stated 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 (ZPA) 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 (0.15g in the horizontal direction and 0.10g in the vertical direction).

The NRC staff reviewed the applicant's response against the UFSARs for IP2 and IP3 and confirmed that the ground accelerations for the DBE of 0.15g horizontal and 0.10g vertical were used to analyze the no loss-of-function concept. Therefore, the NRC staff found the applicant's response to RAI 6-7 acceptable. On the basis of its review, the NRC staff finds that this analysis of the STC is acceptable.

3.6.2.8 Load Case 9: Non-Mechanistic Tipover of Loaded HI-TRAC Cask

The applicant performed a tipover analysis to ensure that a loaded STC, inside the HI-TRAC, will survive a tipover event when enroute 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, and thus provides no energy absorbing capability during the event. The analysis is detailed in Holtec Report HI-2104706 (proprietary). 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 the tipover, and the stresses/decelerations induced in the HI-TRAC and STC were below the acceptable limits. The issue of hydride reorientation in the spent fuel will not occur due to the direct load, temperature, pressure, and wet (borated water) conditions of the transfer of the high burnup fuel assemblies.

In summary, the NRC staff concludes that the structural analyses for the STC and the HI-TRAC systems for the 9 different load cases were appropriate, and the results of those analyses are acceptable.

3.6.2.9 Other Structural Reviews

Prior to initial use, the applicant will leak test the entire confinement boundary of the STC to leaktight criteria per ANSI N14.5, perform spot radiography examination of all the STC boundary welds (per ND-5200), and perform the ASME code-required 125% hydrostatic pressure test. The HI-TRAC is also required to undergo the hydrostatic pressure test, however the HI-TRAC will not be leak tested to the full extent of ANSI N14.5. The top lid seals of the HI-TRAC are required to be tested to the rate of 1.0 X 10^{-3} ref·cm^3/s before transfer can take place. Detailed information regarding the STC acceptance testing is contained in Section 8.4.3 of the SAR. Detailed information regarding the STC and HI-TRAC maintenance and inspection program is described in Section 8.5 of the SAR. The applicant's commitment to perform post fabrication acceptance tests allows confirmation that ASME Code, Section III, Division 1, Subsection ND is adequate and applicable to the STC design, along with the applicant's code reconciliation between Subsection ND and NC.

Table 8.5.1 of the SAR tabulates the Maintenance and Inspection Program Schedule. Of note, the STC will be tested to the full extent of ANSI N14.5 for post-fabrication, pre-transfer, annual, and post-maintenance activities.

3.6.3 Conclusion

Based on the applicant's structural evaluation and the staff's review of the statements and methodologies employed in the application, the NRC staff concludes that the structural aspects of the proposed design have been adequately described and meet the regulatory requirements of 10 CFR Part 50.

3.7 Shielding And Radiation Protection Evaluation

The objective of the shielding and radiation protection review is to ensure that there is adequate protection of the public and workers against radiation from the proposed spent fuel transfer. The proposed transfer operations use an STC and are referred to as wet transfer operations because the spent fuel and non-fuel hardware are maintained in a borated water environment for the duration of the operations. The NRC staff's review evaluates whether the proposed shielding features and operations provide adequate protection against radiation for the operating staff and members of the public, and that exposures (from both direct radiation and any effluents) will satisfy regulatory requirements during normal operating, off-normal (e.g., crane hang-up and vertical cask transporter breakdown), and design-basis accident conditions. The review includes consideration of the shielding and radiation protection design description, radiation source definition, shielding model specification and shielding analyses for the STC and the HI-TRAC transfer cask used to move the STC between the IP3 and IP2 SFPs. The review then evaluates compliance with both ALARA requirements for system design and operations, and dose limits for occupational workers and members of the public. Note that the phrase "members of the public" also refers to people at the plant site who are not occupational radiation workers.

The proposed transfer system design and operations have been submitted as part of a 10 CFR Part 50 license amendment request. The applicant recognized the unique aspects of the proposed activities and that the associated hardware and operations are similar to those employed as part of 10 CFR Part 72 dry cask storage activities. Thus, for purposes of the shielding and radiation protection evaluation, the applicant considered the regulatory requirements in 10 CFR Part 72 and 10 CFR Part 20. The applicant proposed to demonstrate compliance with the most restrictive requirements of these regulations for those instances where there is more than one limit for the same condition. With regard to 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; thus, compliance with these requirements assures compliance with the requirements of 10 CFR Part 20 related to doses at or beyond the controlled area boundary (i.e., doses to members of the public in the unrestricted area). The NRC staff has considered this proposed approach and finds it to be acceptable with respect to the shielding and radiation protection evaluation. Thus, for purposes of this review, the regulatory requirements against which the operations are reviewed include 10 CFR 72.104 and 72.106(b). The requirements in 10 CFR 20.1301(e) refer to the limits 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). In addition, staff also considered the regulations in 10 CFR 72.122(b) and (c), 72.126 and 72.128. The 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases and radiation from other uranium fuel-cycle operations. Thus, the review described in this section relies upon the results of the review of the confinement evaluation (see section 3.8 of this safety evaluation) for those aspects involving effluents. Given the selected licensing approach, the review used the guidance provided in NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" and NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," as appropriate.

The applicant proposed to amend the licenses for both IP2 and IP3 to allow for the transfer of spent fuel from the IP3 SFP to the IP2 SFP. The proposed amendment to the IP2 and IP3 licenses includes the addition of an Appendix C, "Inter-Unit Fuel Transfer Technical Specifications," to each license, that captures important system design and operations information and requirements for the proposed actions, similar to what is done for 10 CFR Part 72 dry storage system certificates of compliance.

The transfer operations will be conducted using an STC to remove spent fuel from the IP3 SFP and load it into the IP2 SFP. The STC will be transferred between the buildings that house the respective units' SFPs in a HI-TRAC transfer cask. The HI-TRAC 100D transfer cask is part of the HI-STORM 100 system certified for spent fuel dry storage and in use at IP2. Certain unique operations and equipment (e.g., the STC) have been proposed to address the limited capacity of the IP3 FSB crane, which lacks sufficient capability to handle the transfer casks of approved dry storage systems. Thus, movement of the IP3 spent fuel into dry storage cannot be performed without the fuel first being moved to the IP2 SFP, where there is sufficient crane capacity to lift the approved dry storage system transfer casks.

3.7.1 Design Description

3.7.1.1 Design Criteria

The applicant considered the regulatory dose requirements in 10 CFR Part 20 and 10 CFR Part 72. For those instances where the regulations specified dose limits for the same conditions, the more restrictive requirements were invoked as design criteria. Since 10 CFR Part 72 limits for normal operations and off-normal conditions (10 CFR 72.104(a)) and for accident conditions (10 CFR 72.106(b)) are more restrictive than the limits in 10 CFR Parts 20 for doses at the controlled area boundary, these limits were set as acceptance criteria. The applicant also identified maintaining of dose rates As Low As is Reasonably Achievable (ALARA) for both occupational exposure and dose to members of the public (10 CFR 20.1201 and 10 CFR 72.104) as a design objective and criterion. Additionally, the applicant identified the dose limits for members of the public specified in 10 CFR 20.1301(a) and (b) as additional design criteria for the transfer operations. The NRC staff reviewed these criteria and finds that these criteria, along with the requirements of 10 CFR 20.1301(e), are acceptable and appropriate for this application because compliance with these criteria will assure compliance with applicable regulatory dose requirements for these types of operations.

3.7.1.2 Design Features

The equipment to be used for the wet transfer operations includes the newly-designed STC. It also includes the HI-TRAC 100D (HI-TRAC) transfer cask that is part of the approved HI-STORM 100 dry storage system in use at IP2 for dry storage operations. These two devices provide shielding for the wet transfer operations. The designs of the STC and the HI-TRAC are described in Chapter 1 of the SAR and in the technical drawings included in Section 1.5 of the SAR.

The STC shielding design includes a thick steel base, a multi-shell wall comprised of an inner steel shell and an outer steel shell with lead in between, a thick steel lid that also has an area of lead (1.5 inch thick with a 32 inch nominal radius) that covers most of the STC cavity (42 inch nominal radius). The STC cavity contains a structural support basket to hold spent fuel assemblies in place. See Section 3.2.1 of this safety evaluation for a description of the fuel assemblies. The minimum dimensions of the STC radial shielding are specified in the proposed TS, Appendix C, Part I, Section 1.0. The technical drawing shows a smaller minimum dimension for the outer steel shell (3/8 inch). The STC dimensions in the TS are used for calculating dose rates for the STC while outside of the SFP and not in the HI-TRAC, and the minimum STC shell thicknesses in the technical drawings are used for calculating dose rates for the loaded HI-TRAC. The as-built STC meets the dimensions specified in the TS and any replacement STC that may be built in the future must also meet the dimensions in the TS. Thus, the analyses are acceptable and bounding with regard to the STC shielding dimensions. Based on these considerations, the NRC staff finds that the dimensions in the proposed TS are acceptable for describing the STC radial shielding design. The NRC staff notes that the STC also has four steel radial ribs that extend along the axial height of the shells and through the lead shielding. These ribs represent areas of potential radiation streaming. Additionally, the vent and drain ports and the gap present between the STC lid and top flange during lifting operations are also areas of potential radiation streaming. The STC design was modified to include shield blocks and a shield ring, attached to the lid, to mitigate radiation streaming in these areas on and near the lid.

The STC shielding design also utilizes the neutron absorber plates in the STC basket and the soluble boron in the water (spent fuel pool water) used to fill the STC. The NRC staff notes that reliance on these features for shielding, particularly the soluble boron, is unusual as compared to 10 CFR Part 72 dry storage systems. The staff notes, however, that these features are routinely relied upon for purposes of criticality control for dry storage system loading (and, if needed, unloading) operations and that TS are implemented to ensure proper control on the fabrication and installation of the absorber plates and the soluble boron concentration. The same is true for the proposed STC design and operations. Thus, staff finds that reliance on the absorber plates and soluble boron for shielding is acceptable since the appropriate TS (i.e., TS 3.1.1, "Boron Concentration" and TS 4.1.2.1.f-I and TS 5.2 with regard to the Metamic absorber panels) specify that these features are provided and have the necessary shielding-related properties.

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 HI-TRAC transfer cask's shielding design includes a solid steel top lid, a multi-shell wall comprised of an inner steel shell and an outer steel shell with lead in between and a steel water-filled jacket attached to the outside of the shell, and a base lid comprised of steel and lead (hereafter referred to as a pool lid) at the axial base of the transfer cask. The steel and lead shells in the wall provide for gamma shielding while the water jacket provides for neutron shielding. Additional shielding is provided by the unborated water in the annulus between the HI-TRAC's inner shell and the STC. Given the size of the annulus and the potential effects from the STC being off-centered, the design includes a means to keep the STC centered within the HI-TRAC cavity. Initially, the centering feature was integral to the STC itself with the previously mentioned STC steel ribs extending outside the STC outer shell near the base and top of the STC. However, the applicant modified the design to the currently proposed means of centering the STC, using a separate aluminum centering device; the steel ribs no longer extend beyond the STC outer shell surface. In an accident condition, this centering device also acts like an impact limiter for the STC. The HI-TRAC is also designed with a bottom missile shield (BMS) to protect the bottom flange area where the pool lid and HI-TRAC wall meet from impacts due to tornado missiles. The proposed TS, Appendix C, Part I, Section 1.0 includes a description of the HI-TRAC (including the minimum radial shielding dimensions).

The applicant plans to use existing equipment that can facilitate remote operations while transferring the STC between the IP2 and IP3 SFPs and the HI-TRAC. For the purpose of these operations, remote operations means operators are located at significant distances from the STC but are still in the same room as the STC (i.e., operators are not in a separate room or building). Performance of these activities is in keeping with ALARA practices, helping to reduce occupational exposures during handling of the STC by allowing personnel to remain at greater distances from the STC. This equipment includes means to remotely operate the building cranes and guide the insertion of the STC into the HI-TRAC. Both the STC base and the top of the centering device are designed so that the STC may be easily inserted into the centering device in the HI-TRAC.

The NRC staff has reviewed the description of the wet transfer operations equipment provided by the applicant. Based upon that review, the staff finds that the applicant has adequately described the equipment design to allow for evaluation of the transfer operations with respect to shielding and radiation protection. A description of the required equipment and significant parameters of the equipment design are provided in proposed Appendix C for the IP2 and IP3 licenses. These parameters include the minimum radial shielding dimensions, important parameters of the Metamic neutron absorber plates and controls for the soluble boron concentration in the STC.

3.7.2 Radiation Source Definition

The proposed radioactive contents of the STC include IP3 spent fuel assemblies. IP3 operates with Westinghouse 15x15 assemblies. The fuel rods in these assemblies may or may not have axial blankets of natural or slightly enriched uranium. The fuel rod cladding material, including the guide tubes, is zirconium alloy. The assembly hardware includes steel and inconel, including inconel grid spacers for some assemblies. The maximum uranium mass loading, an important parameter for the radiation source term, is 473 kg (uranium) per assembly. For purposes of calculating the assembly radiation source term, the applicant used the characteristics of the Babcock & Wilcox (B&W) 15x15 fuel assembly. This assembly has been shown to bound the source term from the Westinghouse 15x15 assembly in analyses for the HI-STORM 100 FSAR. The B&W 15x15 has a maximum uranium mass loading of about 495.5 kg per assembly. Therefore, use of this assembly type is conservative and acceptable to staff for determining the assembly source term for the STC. The TS include assembly specifications important to the source term in proposed Appendix C, Part II, Table 4.1.1-1.

The applicant initially proposed that the assembly contents be limited by decay heat, a maximum assembly average burnup and a minimum cooling (or post-irradiation) time. The decay heat limit is based upon the decay heat restrictions on different zones of STC fuel basket locations; the outer 8 locations are limited to a total of 650 W per location while the inner 4 locations are limited to a total of 1105 W per location. Assemblies of any combination of cooling time greater than 5 years and assembly average burnup less than 55 gigawatt days per metric ton of uranium (GWd/MTU) that resulted in an assembly decay heat less than the limit for the selected basket location would be allowed to be transferred in the STC. Thus, the assembly contents were to be primarily defined by the decay heat limits.

The NRC staff considers that decay heat alone is not normally sufficient to define an assembly's radiation source term. Assemblies having the same decay heat may have significantly different radiation source terms. Thus, to define the contents by decay heat alone, analyses would be needed to determine the bounding radiation source term. These analyses were not done as part of this application. Therefore, the applicant modified its proposal to include assembly content limits in terms of combinations of maximum assembly average burnup, minimum cooling time and minimum initial assembly average enrichment. These limits are included in the proposed TS (see Appendix C, Part II, Limiting Conditions for Operation (LCO) 3.1.2, Table 3.1.2-3). The applicant revised its analyses to use these parameter combinations to determine the bounding source terms for the STC assembly contents. The staff finds this method of defining the allowable spent fuel contents to be acceptable since it provides limits on significant parameters for determining the radiation source term.

In its source term analyses, the applicant assumed a uniform enrichment along the full length of the assembly active fuel zone (i.e., the axial blankets were not modeled). The NRC staff notes that the presence of blankets affects the burnup profile and increases the burnup in the axial middle area of the assembly, increasing the source term for assemblies with long axial blankets (i.e., longer than about 6 inches). The applicant evaluated the impact of axial blankets for the IP3 assemblies and found that the dose rates would increase for these assemblies by less than 3%. The applicant calculated source terms using the B&W 15x15 assembly characteristics. Based on the difference in uranium mass loading versus the IP3 assembly type (i.e., Westinghouse 15x15), the staff finds that the conservatism of using the B&W 15x15 source term is sufficient to cover any effect of axial blankets.

The STC proposed contents also include non-fuel hardware (NFH), which can be inserted into the top of a fuel assembly. Those items important for shielding are the burnable poison rod assemblies (BPRAs), wet annular burnable absorbers (WABAs), rod cluster control assemblies (RCCAs), hafnium flux suppressors, thimble plug devices (TPDs), and neutron source assemblies (NSAs). While there are some TPD designs that have water displacement rods that extend into the active fuel zone, the TPDs in use at IP3 do not have these rods. NFH is, with the exception of NSAs, a contributor to the gamma source; NSAs may contribute to both the neutron and the gamma sources. IP3 uses two types of NSAs, plutonium-beryllium (Pu-Be) primary sources and antimony-beryllium (Sb-Be) secondary sources. The short half-life of the antimony isotope (about two months) of the secondary source means that, given the proposed minimum cooling time, this type of NSA will not contribute to the neutron source. The Pu-Be source has a significantly long half-life and so will be a significant neutron source. The proposed technical specifications (TS) include limits for NFH (see Appendix C, Part II, LCO 3.1.2, Table 3.1.2-2), and do not allow NSAs to be transferred in the STC based on IP2 SFP storage analyses.

The applicant used the SAS2H and ORIGEN-S modules of the SCALE computer code, Version 4.3, to calculate the source terms for the spent fuel assemblies and the NFH. The calculations were done with the 44-group cross section library. For the source term from assembly hardware and NFH, scaling factors were applied to account for the variations in the neutron flux at the locations of the hardware and NFH components during irradiation in the reactor and the different masses of steel and inconel in those components. The scaling factors and material masses used in these calculations are the same as those used in the HI-STORM 100 system analyses, which also address assemblies of the type used at IP3. The NRC staff has reviewed and found these scaling factors and material masses to be acceptable for various NRC-approved Certificate of Compliance (CoC) amendments for the HI-STORM 100 system (e.g., see Sections 5.2.1 and 5.2.3 of the safety evaluation report for the initial CoC [Ref. 8]). Therefore, the staff finds these scaling factors and material masses for the assembly hardware and NFH components to be acceptable.

The NRC staff notes that there are more updated versions of the SCALE code available. The staff also is aware that the burnup range for which these SCALE modules have been validated is limited to low burnup fuel and that the code developer is no longer supporting the SAS2H module as part of the SCALE code. However, the staff also recognizes that this code, code version and cross section library are the same ones used for calculating source terms for the HI-STORM 100 dry storage system. The staff compared the proposed STC contents with the contents for the HI-STORM 100 system. While the enrichments used for the assembly burnups differ (the enrichments for the STC contents analysis is generally lower for the same burnup than for the HI-STORM 100), the enrichments, burnups and cooling times are all within the range of those analyzed in the HI-STORM 100 FSAR. The proposed burnup limits for NFH are also within the range of the limits proposed for the HI-STORM 100, though with increased cooling times to account for differences in the assumed cobalt levels in the NFH materials. In addition, the maximum burnup analyzed for the STC spent fuel assemblies does not extend significantly into the high burnup regime, so that any future extension of burnup may necessitate evaluation of source term uncertainties similar to what was done for the higher burnups allowed for the HI-STORM 100 system. Considering these factors, the staff finds that the use of the selected code, code version and code modules is acceptable for calculating the radiation source terms (strength and spectra).

3.7.2.1 Gamma Source

The gamma sources for the fuel assemblies are provided in Table 7.2.1 of the SAR. While this table does not show the gamma source for each proposed limiting combination of burnup, enrichment and cooling time, it does describe the source for most combinations, including those that result in the bounding dose rates. The gamma source from the hardware and NFH is stated in Tables 7.2.2, 7.2.5 and 7.2.7 of the SAR. The gamma source from the assembly hardware and the NFH is assumed to be all from Cobalt-60 in the steel and inconel components, with the exception of the RCCAs. The RCCA gamma source includes the activated silver-indium-cadmium (AgInCd) neutron absorber material. The source for RCCAs was calculated using the same assumptions regarding operations with RCCAs as is done in the HI-STORM 100 analyses. The staff finds this to be acceptable as these assumptions are bounding for the operations with RCCAs at IP3.

The amount of cobalt assumed for the steel and inconel components of the assembly hardware and NFH can have a significant impact on the dose rates. For inconel components, the applicant assumed a cobalt amount of 4.7 g/kg inconel. This assumption is consistent with analyses performed for dry storage systems. For steel components, different amounts of cobalt were assumed depending upon the age of the assembly and NFH. The basis for using different amounts is that in the late 1980s, industry made an effort to reduce the amount of cobalt in assembly hardware. Prior to that time, the amount of cobalt was significantly higher, as determined from measurements of various assembly hardware components; it is expected that this is also the case for NFH. Based on measurements of cobalt obtained from the fuel manufacturer for the assemblies in use at IP3, the applicant analyzed the assembly hardware to have 1.2 g cobalt per kilogram of steel for assemblies fabricated prior to 1989. For assemblies fabricated after that year, the amount was reduced to 0.5 g cobalt/kg steel, a value which has been used in dry storage system analyses. The higher cobalt amount was also applied to BPRAs and TPDs. Based on the proposed limits for burnup and cooling time, the staff finds that this is conservative for BPRAs and for most of the burnup and cooling time limits for TPDs, and is consistent for TPDs with burnups and cooling times that indicate the TPDs were fabricated prior to efforts to reduce cobalt levels (taken to be 1989 for this application). The staff finds the amount of cobalt used in the analyses to be acceptable since it is based on data provided by the manufacturer of the fuel assemblies used at IP3.

The cobalt amount in the steel of NSAs was initially assumed to be 0.5 g/kg. Based on the proposed burnup and cooling limits for NSAs, the staff had concerns that this assumption was not adequate, in that the burnup and cooling time limit is more consistent with NSAs that would have been fabricated prior to 1989, implying that the higher cobalt amount was more appropriate. The applicant modified its analyses to consider the cobalt amount in the NSA steel to be 1.2 g/kg, consistent with the treatment of the other NFH and the hardware of older assemblies. Due to the increase in cobalt assumed in the NSA steel, the minimum cooling time for NSAs was extended. The staff finds the higher cobalt amount used for the NSAs to be acceptable since it is based on data provided by the manufacturer of the fuel assemblies used at IP3. However, the proposed technical specifications (TS) (see Appendix C, Part II, LCO 3.1.2, Table 3.1.2-2) do not allow NSAs to be transferred in the STC due to IP2 SFP storage analyses.

3.7.2.2 Neutron Sources

The neutron sources for the fuel assemblies are provided in Table 7.2.3 of the SAR. The neutron source spectra in that table are for the same burnup, enrichment and cooling time combinations for which gamma source spectra are provided in Table 7.2.1. In addition to the fuel neutron source, some types of NSAs may also be a significant source of neutrons as described previously. Pu-Be type NSAs will have a source strength similar to that of an assembly. The applicant used the typical initial Pu-Be source strength for the shielding analyses to account for the contribution of NSAs to dose rates. The staff finds this to be acceptable as there will be some decay of the source strength (the neutron production rate) over the course of the NSA's use in the reactor and its post-irradiation cooling time; therefore, sources with initial strengths that are somewhat above the typical initial source strength will have strengths that are similar to that assumed in the analysis. However, the proposed technical specifications (TS) (see Appendix C, Part II, LCO 3.1.2, Table 3.1.2-2) do not allow NSAs to be transferred in the STC due to IP2 SFP storage analyses.

3.7.3 Shielding Model Specification

The applicant described the shielding model, including the configurations of the shielding, the source and the material properties, primarily in Section 7.3 of the SAR. Other sections in Chapter 7 of that report contain additional information, including loading patterns. The staff reviewed this information during its revew, including the structural, thermal, and materials areas, and the descriptions of the transfer operations, as discussed in the following sections.

3.7.3.1 Configuration of the Shielding and Source

3.7.3.1.1 Source Configuration

The configuration of the source is based upon the proposed loading restrictions in the TS. TS LCO 3.1.2, together with TS Figure 3.1.2-1 and TS Tables 3.1.2-2 and 3.1.2-3, defines restrictions on the allowable contents per STC basket cell. For fuel assemblies, the basket is divided into two zones or regions: the central four cells and the remaining outer eight cells. The TS defines six loading configurations for assemblies. The sixth configuration was added to allow for assemblies with inconel grid assemblies in the active fuel zone to be loaded in any basket cell; the analyses for the other five configurations assume these assemblies are only allowed in the inner four cells. Dose rates were calculated for these configurations at 1 meter distance from the STC surfaces and the HI-TRAC surfaces. The configurations yielding bounding dose rates, configurations 3 and 4 (see Table 7.1.1 of the SAR and TS Table 3.1.2-3), were used for subsequent calculations. Configuration 4 was bounding for all surfaces of the STC. For the HI-TRAC, configuration 3 was bounding for the radial surface and configuration 4 was bounding for the top and bottom surfaces; thus, both configurations are used to determine the bounding HI-TRAC dose rates. All analyses are for intact fuel assemblies, with the fuel retaining its basic configuration for all normal, off-normal and accident conditions; only intact fuel assemblies are allowed to be transferred in the STC. The TSs provide the definition of intact fuel assemblies (see proposed operating license Appendix C, Part II, Section 1.1).

For the NFH configurations, an STC (and HI-TRAC) containing assemblies with BPRAs was found to be generally bounding for dose rates around the side and above the top. The configuration with RCCAs replacing the BPRAs in the inner four basket compartments was generally bounding for dose rates below the base. In some cases other configurations resulted in higher dose rates; however, the differences were negligible and/or were localized (e.g., at the STC steel radial ribs). Thus, dose analyses were based on loading configurations with BPRAs, or BPRAs and RCCAs as appropriate.

The NRC staff finds that using the source configurations (spent fuel assemblies and NFH) that result in bounding dose rates is an appropriate approach to the shielding analysis. Given the function of the STC and the wet transfer operations, it is expected that at least the assembly contents definitions are based upon the spent fuel inventory in the IP3 spent fuel pool. Thus, bounding dose rates for the transfer system and operations provides an envelope for the radiological conditions which may be expected when handling the loaded STC and HI-TRAC. With the contents definitions specified in the TS, the staff finds that the use of a 'regionalized' loading configuration for the spent fuel contents and the currently analyzed NFH configurations will be bounding for shielding purposes for the transfer operations.

One feature which is unique to the wet transfer operations, as compared to similar activities performed for dry storage operations, is the presence of borated water for the entire transfer process. Compared to the dry gases present during the later stages of loading operations for dry storage, the borated water can act as a moderator which increases the overall reactivity of the fuel assembly. Similar to dry storage shielding analyses, the applicant modeled the source as discrete axial zones, with each zone a homogenized mixture of the materials present in that zone. It was not clear to the staff that modeling the source zone as a homogenized material zone given the presence of moderator is appropriate, since this kind of modeling assumption can be non-conservative with respect to subcritical multiplication and the potential impact on neutron dose rates. Therefore, the applicant subsequently evaluated the impacts on neutron dose rates of modeling the fuel homogenized with the moderator versus heterogeneously modeling the assembly. That analysis indicates that the neutron dose rates for the STC are basically the same for these two modeling assumptions. This conclusion, though not explicitly analyzed, is extended to the HI-TRAC neutron dose rates. The staff recognizes that the configuration of source materials and moderator not only affects neutron multiplication (since the water moderator thermalizes the neutron spectrum), but it also affects the shielding behavior of the materials involved. These two aspects of the model can have different impacts on the neutron dose, the extent of which depends on the system. Based on these considerations and the applicant's analysis, the staff finds that homogenization of the fuel with the moderator (which in the current analysis credits the soluble boron in the STC cavity water), is acceptable.

Two other aspects of the source configuration considered by the staff are the axial burnup profile and the physical extent of the source region. As stated earlier, the applicant did not consider axial blankets as part of its source term calculations. This consideration also extends to the burnup profile, for which the applicant did not include the effects of axial blankets. This results in greater source strength toward the axial ends of the assembly and slightly less strength along the rest of the axial length of the fuel. The impact of this modeling assumption is included as part of the dose rate effect described earlier (see Section 3.7.2 of this safety evaluation, which also describes the acceptability of this assumption). The applicant used the characteristics of the B&W 15x15 assembly for the shielding analyses with the exception of the axial configuration for calculations with the loaded STC outside the pool and HI-TRAC. For that scenario, the applicant used the axial configuration of the Westinghouse 15x15 assembly. The staff finds this approach acceptable since the B&W 15x15 assembly is bounding in terms of shielding, as compared to the Westinghouse 15x15 assembly used at IP3.

3.7.3.1.2 Shielding Configuration

The applicant used a single model for the loaded STC during handling between the SFP and the HI-TRAC. This model is used for the normal, off-normal, and accident conditions for the loaded STC. All handling of the STC will be done in the FSBs by means of the installed cranes. Per the applicant's commitments, the cranes to be used have been upgraded to single-failure proof cranes, thus any concerns due to drop accidents are resolved. Other accidents with the STC by itself do not affect its shielding capabilities.

Section 3.7.1.2 of this safety evaluation describes the design features important to shielding and radiation protection. The applicant's model of the STC includes these design features. The model also accounts for the steam gap that is required for purposes of overpressure protection, using a gap that is larger than that which is specified by the TS. As already stated, the model uses as-built dimensions for the radial shielding. The model also takes credit for 2000 ppm of soluble boron in the STC cavity water. This credit for soluble boron is not used in dry storage shielding evaluations but is acceptable given that this represents reality, the boron concentration is controlled by TS, and there is no credible way for loss of this water to occur. The staff finds the other aspects of the model to be consistent with the STC design as described in the report, the technical drawings and the TS.

The applicant used several different calculation models to analyze normal, off-normal and accident conditions with a loaded HI-TRAC (i.e., a loaded STC inside the HI-TRAC). These models use the minimum dimensions of the STC's radial shielding that are specified in the technical drawings. Additionally, the applicant does not take credit for the soluble boron in the water in the STC cavity for the configuration of the STC in the HI-TRAC. The dimensions of the HI-TRAC are consistent with those specified in the technical drawings and the TS and include the steel ribs that are in the water jacket. The models neglect the HI-TRAC lid, the centering device and the BMS. The normal conditions model includes the required air gap in the HI-TRAC cavity; the gap size is consistent with that specified in the TS. The shielding configuration is the same for normal and off-normal conditions (HI-TRAC transporter breakdown); therefore, there is not a separate off-normal conditions model. Based on its review of the HI-TRAC design and technical drawings, the staff finds the HI-TRAC calculation models to be acceptable.

The applicant defined three different accident models. These are: 1) loss of the water in the HI-TRAC water jacket (which functions as a neutron shield), 2) loss of water in the HI-TRAC water jacket and the HI-TRAC cavity and 3) loss of water in the HI-TRAC water jacket, the HI-TRAC cavity and the STC cavity with the STC shifted off-center toward the wall of the HI-TRAC cavity. The first model addresses any accident scenario that may impact the HI-TRAC water jacket. The second and third models address effects of a tip-over accident. In a tip-over accident, the centering device acts as an impact limiter and will crush. Additionally, while the cask is on its side, some of the fuel may become uncovered as the steam gap shifts from above the fuel to along the STC side and the air gap in the HI-TRAC shifts from above the STC to along the side of the HI-TRAC inner shell. The applicant's third model conservatively accounts for such a scenario with the entire configuration void of any water and the STC shifted to be in contact with the HI-TRAC cavity wall. The actual accident analyses demonstrate no loss of water from the STC, and no leakage from the STC, so assuming the STC is empty of water is very conservative. Based on its review of the accident scenarios, staff finds these models capture the shielding impacts of those scenarios.

3.7.3.2 Material Properties

The materials relied upon for shielding during the transfer operations include the steel and lead in the STC's and HI-TRAC's walls, bases and lids, the STC basket steel and Metamic neutron poisons, the borated water in the STC cavity and the water in the HI-TRAC cavity and neutron shield (or water) jacket. The staff's review of the materials properties of the STC and HI-TRAC components is described in Section 3.8 of this safety evaluation. In its review, the staff gave particular attention to nonstandard materials (e.g., Metamic). The staff noted that credit for soluble boron and neutron absorber plates is not typically used in dry storage shielding evaluations but is acceptable for this application because the characteristics important to shielding are the same as those that are important for criticality. Those characteristics are captured in the proposed TS (e.g., TS LCO 3.1.1, TS 4.1.2.1.f to 4.2.1.i, and TS 5.2). Section 3.8 of this safety evaluation adequately describes the acceptance tests and maintenance program for materials and features important to shielding.

The NRC staff reviewed the description of the materials used in the radiation shielding code input files. This review included a sample input file that was provided during the early stages of the review. Based on its review, the staff finds that the material properties used in the shielding code are consistent with the materials descriptions given in the SAR and the shielding calculation package. The densities of the materials that represent mixtures (e.g., fuel homogenized with water, Metamic, etc.) are reasonable and consistent with the densities used in analyses for dry storage shielding calculations, as applicable. The density of the water (borated and non-borated) is reduced to account for the elevated system temperatures. Based on its review, the staff finds that the materials properties used in the shielding analysis are acceptable.

3.7.4 Shielding and Radiation Protection Analyses

Section 7.4 of the SAR describes the shielding analysis method and the dose rates determined for the loaded STC and the loaded HI-TRAC under different operating conditions. Section 7.4 of the SAR also describes the radiation protection evaluation, using the calculated dose rates to provide estimates of occupational and public exposures. The radiation protection evaluation relies upon the shielding analyses and the effluent confinement evaluation (see Section 3.8 of this safety evaluation for evaluation of the confinement design). Following the applicant's submission of the LAR, the confinement design was modified so that the STC is now designed and operated to meet the conditions for being "leak-tight" as defined in ANSI N14.5. Therefore, there is no effluent contribution from the wet transfer operations; accordingly, the radiation protection evaluation only needs to consider the direct radiation occurring from the operations. This evaluation does, however, consider the dose contributions from other site operations, including the Independent Spent Fuel Storage Installation (ISFSI), in determining compliance with 10 CFR 72.104(a) annual dose limits.

3.7.4.1 Computer Codes

The applicant used MCNP5, a three-dimensional Monte Carlo code to calculate dose rates for the transfer operations. This code was developed by Los Alamos National Laboratory (LANL) and is used extensively for shielding analyses, including benchmarking of the code against experimental data. The code's capabilities are such that the STC and HI-TRAC features can be modeled explicitly and in detail, allowing for evaluation of potential streaming paths. The code uses continuous energy cross section libraries, based on ENDF/B-V and B-VI data, to calculate the interactions of radiation in the models. MCNP5 is used as part of a two-step process to determine dose rates as explained in Section 7.4.1 of the SAR. This analysis method is the same as is used for the HI-STORM 100 shielding analyses; those analyses also use MCNP5. Based on the foregoing considerations, staff finds that MCNP5 is an acceptable code for use in this application and that the analysis method is acceptable.

3.7.4.2 Flux-to-Dose-Rate Conversion

The applicant used the conversion factors in ANSI/ANS 6.1.1-1977 to calculate dose rates from fluxes determined in MCNP5. These conversion factors were included in the MCNP5 input so that the conversion is done as part of the MCNP5 calculation. These conversion factors are those that are accepted by staff for dry storage system analyses, as described in the dry storage system review guidance (see Section 6.5.4.2 of NUREG-1536).

The applicable regulatory limits of 10 CFR 20.1301(a) and (e)(citing 40 CFR190) are in quantities of "effective dose equivalent" and "dose equivalent," respectively. The applicant calculated the doses and dose rates in units of mrem and rem of "dose equivalent." This is acceptable to the NRC staff, as provided in NRC guidance in Regulatory Guide 8.40.

Based on these considerations and the similarity of the operations in the proposed amendment to various operations involved in spent fuel dry storage, the staff finds the use of these conversion factors to be acceptable.

3.7.4.3 Dose Rates

The applicant calculated dose rates at the surface of and at various distances from the loaded STC and the loaded HI-TRAC. The distances include those that are representative of the locations where operations personnel will be working or are located in relation to the STC and HI-TRAC. The distances also include those that are representative of distances to the boundaries of restricted areas for controlling dose to members of the public on site (including the applicant's employees who are not trained as occupational radiation workers) and distances to the controlled area boundary for purposes of evaluating annual dose against the 10 CFR 72.104(a) and 10 CFR 20.1301 limits. Section 7.4 of the SAR has several tables of dose rates for the STC and the HI-TRAC.

The dose rates and the distances for which dose rates are reported in the SAR were modified over the course of the staff's review to address staff questions. Initially, only average dose rates (averaged over the surface of the STC and HI-TRAC and at distances from the surfaces) were reported. The STC dose rates were determined for a regionalized loading pattern that was intended to be representative of the contents of the STC. HI-TRAC dose rates were determined for both a regionalized and a uniform loading pattern. The regionalized loading pattern was used for evaluating normal and off-normal conditions and the uniform loading pattern was used for evaluating accident conditions (i.e., loss of the HI-TRAC's water neutron shield). The contents were specified by a single maximum burnup and a single minimum cooling time and decay heat limits for the outer and inner basket compartments. A single minimum enrichment was assumed for these dose rates. Minimum enrichment is used since irradiation of lower enrichment fuel yields higher concentrations of actinides for the same burnup, which translates into an increased neutron source. The initial dose rates were based only upon the spent fuel contents.

The staff questioned the initially provided dose rates given this initial definition of the allowable contents and its understanding from other sections of the SAR that non-fuel hardware would also be present in the STC; it was not clear that the selected regionalized loading pattern would adequately capture the dose rates from the allowable contents. The staff also found that the dose rates should include the contributions from the NFH to be transferred. Also, given the system design, particularly for the STC, the staff found that the dose rate analyses should account for the azimuthal variation of dose rates and streaming paths at the vent and drain port areas of the STC lid.

In response to the staff's concerns, the applicant modified its dose rate analyses to account for NFH and dose rate variations. For NFH, the applicant eventually added limits to the NFH contents in the TS (see Section 3.7.2 of this safety evaluation) so that the analysis with BPRAs remains generally bounding. The configuration with RCCAs is bounding below the STC base and yields slightly higher dose rates on or below the base of the HI-TRAC. Additionally, the applicant evaluated for the azimuthal dose rate variations and reported the highest dose rates. The applicant also provided dose rates at the vent and drain ports, accounting for a design modification that added steel shield blocks at the locations on the underside of the STC lid. The shield blocks were added to reduce the streaming at these locations.

The NRC staff reviewed the revised dose rates resulting from the applicant's modified analyses, which were added to the SAR. The dose rates at and near the surface had increased significantly, with some increasing by more than double (e.g., STC radial surface total with BPRAs increased from about 3.6 rem/hr to about 6.7 rem/hr for a regionalized loading pattern). These revised dose rates are significantly higher than the dose rates on transfer casks for dry storage systems currently approved under 10 CFR Part 72. This raised a concern as to the adequacy of the radiation protection provided by the design and the need to consider additional features or operations controls to ensure adequate protection. Additionally, staff noted that dose rates reported at varying distances either did not change or decreased; thus, the dose rates seemed to be inconsistent. Further, the staff was concerned that the dose rates for the regionalized loading pattern may not adequately capture the dose rates for the allowable contents.

The applicant significantly revised its shielding analyses to address the staff's concerns. The revised analyses also account for changes to TS 3.1.2 on allowable contents of the STC (described in Section 7.2 of the SAR) and the STC design. Dose rates were re-calculated for the loading configurations allowed in the TS to identify the configurations that result in bounding dose rates. The remaining dose rate analyses are for the bounding configurations. The dose rate analyses include an additional streaming path at the gap between the STC lid and top flange that is present during handling of the STC. The STC lid was modified to include a steel ring to reduce this streaming; this ring is credited in the analysis. The analyses include axial and azimuthal variations in the dose rates at and near the radial surface of the STC and azimuthal variations at and near the radial surface of the HI-TRAC. Configurations for the dose rate analyses were modified, as described in Section 3.7.3.1.2 of this safety evaluation. The distances at which dose rates are provided was expanded to account for the varying distances from the STC and the HI-TRAC at which operations personnel are expected to be present during the different stages of the operations. Varying distances are also included to estimate doses to members of the public. Based on its review, the staff finds the applicant's revised shielding analyses to be acceptable.

The NRC staff reviewed the revised dose rates provided in the SAR. The average STC surface dose rates, as revised, are approximately 3.2 rem/hr. These dose rates are within the range of dose rates on transfer casks for approved dry storage systems; therefore, staff concerns with regard to high dose rates are alleviated. The analyses identified significant radiation streaming and dose rates near the STC's radial ribs; dose rates dropped significantly moving only a few degrees around the cask surface away from the rib. The staff finds the analyses address the staff's concerns, providing bounding estimates for the allowable contents and for the different configurations of the loaded STC and the loaded HI-TRAC under different conditions of operations. The analyses provide dose rates at distances that are useful for estimating occupational exposure and exposures for members of the public for normal, off-normal and accident conditions. The staff also finds, based upon the information provided in the application and staff's experience with evaluations of dry storage systems, that the dose rates are reasonable and their variation is consistent with the shielding configuration of the STC and the HI-TRAC.

3.7.4.4 Occupational Exposures

The applicant provided an estimate of the occupational exposures that would result from a single operation sequence to transfer 12 assemblies from the IP3 SFP to the IP2 SFP. This estimate is given in Table 7.4.22 of the SAR. The table also gives the dose estimates for the different stages of the operations sequence, describing the activities, number of personnel, personnel locations and distance(s) from the STC or HI-TRAC, and estimated duration of each stage.

The final estimates account for the applicant's modifications to the shielding analyses discussed above and use the bounding dose rates from those analyses. The estimates include realistic assumptions regarding personnel locations with respect to the STC and the HI-TRAC and the duration of personnel's time at those locations in order to complete the tasks required to perform the various transfer operations. Thus, while a particular operational activity may take over an hour to complete, personnel actions to complete that activity may require personnel to be at the nearest location to the STC or HI-TRAC for no more than 15 minutes. Additionally, personnel locations account for applicant actions to keep exposures ALARA. These include using remote operations to handle the STC outside of the spent fuel pool and the HI-TRAC, including use of extension nozzles when doing a wash down of the loaded STC coming out of the IP3 SFP.

The NRC staff reviewed the occupational dose estimates provided in the SAR. From the person-rem values reported in Table 7.4.22, the staff identified the dose rates assumed for the activity and identified the corresponding position relative to the STC and the HI-TRAC where that dose rate was calculated. The staff also considered the activity description given in Table 7.4.22

and the operations descriptions given in Chapter 10 of the SAR. The staff also considered the estimated activity durations and the durations of personnel at the closest distance(s) to the STC and HI-TRAC. The staff compared these time estimates with those seen for similar activities in dry storage operations. This comparison indicates that the time estimates in Table 7.4.22 are reasonable. While the comparison indicates that a few operations may take somewhat more time than is estimated in the SAR, the impact to exposure estimates would be small because either the increase in estimated time does not result in a dramatically extended time being necessary to complete the operation or the dose rates for that operation are quite small.

The NRC staff also reviewed the dose estimates for the operations and finds that the dose estimates for the operations activities are reasonable. As part of its review, staff questioned some of the assumptions, such as personnel locations for some operations steps, in the applicant's evaluation and performed a confirmatory dose estimate analysis. This confirmatory estimate was based on staff's expectation of reasonable locations for personnel during operations and dose rates at those locations for the operations as described in Table 7.4.22 of the SAR. The staff's estimate for total exposure does differ noticeably from that given by the applicant in Table 7.4.22. The staff notes, however, that the dose estimates are less than the dose estimates for dry storage operations given in the HI-STORM 100 FSAR for approved HI-STORM 100 systems, due to the lower radiation source term, differences in assumptions regarding personnel locations (e.g., remote handling for the STC) and water being always present (vs. dry operations outside the SFP for various operations with the HI-TRAC in dry storage loading operations). Based on its review and its confirmatory analyses, the staff finds the applicant's estimated dose during wet transfer operations to be reasonable and to comply with the requirements of 10 CFR Part 20, including ALARA. Further, as discussed in Section 3.7.4.7 of this safety evaluation, the operations adequately incorporate procedures and engineering controls to achieve occupational doses that are ALARA.

As indicated previously, the applicant's evaluation, at one time during the review, included dose rates at and near the STC surface that were significantly higher than has been seen for currently approved dry storage systems. Considering those dose rates, it was expected that remote operations would be necessary for safe handling of the STC. Thus, a concern arose for how an off-normal event could impact occupational exposures. The event considered to be of concern was a malfunction, or "hang-up," of the crane with the STC completely exposed and the necessary actions to recover from this event (e.g., manual operation of the crane or crane repair). While the applicant did not provide a detailed evaluation of total personnel exposures for this event, the applicant did include additional dose rate calculations for personnel involved in recovery from the event based on their expected locations as well as an estimated time to complete recovery actions.

In its review of the evaluation, the staff questioned whether 4 hours was an appropriate estimate of the time needed for recovery from the "hang-up" event. The applicant explained that this estimate accounts for staging of personnel and equipment as well as its development of procedures for this scenario beforehand and inclusion of this scenario in pre-operational personnel training. Based on these considerations, the staff finds that the 4-hour time estimate is reasonable. Development of procedures and pre-operational training will enhance the personnel's ability to respond in the most effective and efficient manner to a "hang-up" event, and staging of equipment and personnel will also help to minimize the required time. The staff finds that the applicant's additional dose rate calculations for estimating personnel exposure over the entire event duration are reasonably bounding. For example, while the primary crane operator may pass much nearer to the STC while moving to the crane trolley, the time at this close proximity would be expected to be very short given the pre-operational testing and training, the pre-job briefings, radiological protection personnel providing job coverage, and use of real-time electronic personal dosimeters with alarms. Additionally, the distance from the STC to the crane trolley will increase as the STC is lowered into either the SFP or the HI-TRAC. Given that the STC dose rates are shown to be in the range of those calculated for transfer cask operations for currently approved dry storage systems, the analysis for occupational exposure due to off-normal events is acceptable.

3.7.4.5 Dose to Onsite Members of the Public (Including Non-Occupational Workers)

The applicant provided an evaluation of dose to members of the public who may be on-site, including the applicant's non-occupational workers. This evaluation included analyses to estimate dose for a loaded STC when being moved between the SFP and the HI-TRAC and for a loaded HI-TRAC. Analyses were performed for both normal and off-normal conditions. The off-normal conditions include a 4-hour crane "hang-up" and a breakdown of the HI-TRAC transporter. The analyses assume that an individual is at 60 meters from the exposed STC and do not take credit for any shielding afforded by the FSB. For the scenarios with the loaded HI-TRAC, a member of the public is assumed to be onsite at a distance of 20 meters. The applicant will have controls in place that provide reasonable assurance that members of the public will not exceed their dose limits. The applicant used conservative assumptions regarding the occupancy times, having the same individual at this distance for the duration of the normal or off-normal conditions for every transfer operation. In order to be at these same distances for an entire fuel transfer operation, the individual would need to be near the IP3 FSB, then follow the loaded HI-TRAC toward the IP2 FSB, and then remain near the IP2 FSB for the rest of the operation. The analyses are based on 16 transfers per year at which the individual is present. The analyses indicate the dose to onsite members of the public will be within the public dose limits in 10 CFR 20.1301(a) and (b).

Additionally, the applicant described real-time radiological controls that will be used to limit exposures to onsite members of the public. These controls will be instituted in accordance with the IPEC radiation protection program and include limiting access to the FSB to only personnel involved with the transfer operations and controlling access to areas around the building and haul path during the operations. The applicant indicated that access to a plant walkway that crosses over the haul path will also be controlled during the fuel transfers. Based on the conservative analysis provided by the applicant and the description of the actions to be taken for protection of members of the public on-site, which are in accordance with the approved IPEC radiation protection program, the staff finds that the dose limits for on-site members of the public in 10 CFR 20.1301(a) and (b) will be met.

3.7.4.6 Public Exposures At or Beyond the Controlled Area Boundary

The applicant provided an evaluation of the dose at the controlled area boundary for normal, off-normal and accident conditions of operations. Normal and off-normal conditions were evaluated against the annual dose limits in 10 CFR 20.1301(e) and 10 CFR 72.104(a), and the accident conditions were evaluated against the dose limits in 10 CFR 72.106(b). The normal and off-normal condition evaluations include the contributions from the ISFSI and the other IPEC operations. Based on the applicant's planned number of transfers in 2012, the evaluations assume 16 transfers in a year.

Information relative to the ISFSI contribution and the other IPEC operations are taken from evaluations done to support dry storage operations, including the evaluation report required by 10 CFR 72.212. The distance to the ISFSI controlled area boundary is taken to be the shortest distance to the IPEC owner-controlled area boundary, which is the Hudson River located 137 meters from the ISFSI edge. Since the river forms the controlled area boundary, the applicant assumed 500 hours occupancy at this location. The Hudson River is also the nearest part of the IPEC owner-controlled area boundary to any point of the entire wet transfer operations, approximately 160 meters. Thus, this distance (i.e., 160 meters) is used as the controlled area boundary for the wet transfer operations. The occupancy time was assumed to be equivalent to an individual being present at this distance for 8 hours for each transfer operation. The applicant also assumed an occupancy time of 500 hours to evaluate the doses for off-normal conditions and for accident conditions. The results are provided in SAR Tables 7.4.16 through 7.4.20 and demonstrate that the limits in 10 CFR 72.104(a) and 72.106(b) will not be exceeded. The applicant did not evaluate doses at or beyond the other owner-controlled area boundaries since the distances are much greater than the distances in the applicant's evaluation.

The NRC staff considers that a 500 hour occupancy at the nearest boundary is a reasonably conservative assumption, since that boundary is the IPEC shore on the Hudson River. The person would have to be in a boat in order to approach the plant boundary, as the owner-controlled area fences prohibit access by land. Fisherman and boaters are only infrequently seen near the plant, which supports the staff's decision that a 500 hour occupancy is a reasonably conservative assumption. Buoys are also placed in the Hudson River, further out than the 160 meters, to warn boats against entering the plant's security zone. Therefore, individuals on the river would actually be at greater distances from the transfer operations, which further reduces exposures. The staff also considered the distances to locations that include the nearest residence, workplace and recreational area. While the distances to these locations are greater than the distances from the ISFSI or transfer operations to the Hudson River, the occupancy times at those locations would be significantly longer (8760 hours for a resident and 2000 hours for someone working) and allow for correspondingly greater exposure durations for ISFSI operations, wet transfer operations and other IPEC operations.

The NRC staff also considered additional time required to complete 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 FSB when the STC is in the HI-TRAC. Thus, each transfer lasts more than the 8 hours considered by the applicant. The staff also considered the impact of the occurrence of an off-normal event (i.e., crane "hang-up") at each FSB and along the haul path (i.e., transporter breakdown) all in the same year, which would increase the total exposure time.

The impacts of this additional time assumption at each of the selected locations (the applicant's selected controlled area boundary distance and the nearest residence, workplace and recreational area) was evaluated. Taking into account these factors, the staff's evaluation indicates that the limits in 10 CFR 20.1301(e) and 10 CFR 72.104(a) will not be exceeded.

For accident conditions, the applicant analyzed the dose at the controlled area boundary for the three configurations described in Section 3.7.3.1.2 of this safety evaluation. For each configuration, the dose from the side of the HI-TRAC was evaluated. The analysis indicates that there is significant margin between doses from an accident and the 10 CFR 72.106(b) limit, assuming a 500-hour occupancy time at the controlled area boundary. The staff notes that even with the occupancy assumption typically used in dry storage system applications of 720 hours, the dose to an offsite member of the public is well within the regulatory limits.

Thus, based upon the applicant's analysis presented in the SAR and staff's independent evaluation, the staff finds reasonable assurance that the wet transfer operations will meet the regulatory dose limits. As noted above, the dose evaluations include the contribution from the spent fuel dry storage operations for the ISFSI, which is based on the evaluation in the applicant's current 10 CFR 72.212 report, and other facility operations. Operations for the dry storage of spent fuel and other 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 applicant-initiated changes will require evaluation under 10 CFR 72.48 or 10 CFR 50.59 and are subject to NRC inspection.

To ensure that the transfer operations are performed within the assumptions made in the applicant's evaluation, the proposed TS include dose rate limits for the STC and HI-TRAC. These dose rate limits are based on the analyses in the SAR and have several purposes. One important purpose of these limits is to ensure that dose limits will not be exceeded. In the event that the measured dose rates exceed the limits in the TS (see TS, Appendix C, Part II, Section 5.4.2), the TS require the applicant to confirm correct loading of the cask, and to perform an evaluation to determine whether or not continuation of the operations will cause the limits in 10 CFR 72.104 or 10 CFR 20.1301 to be exceeded. It is expected that this 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. The transfer operations will proceed only if the evaluation indicates the limits will not be exceeded.

3.7.4.7 ALARA

The NRC staff reviewed the STC design and the operations descriptions in Chapter 10 of the SAR with respect to considerations of ALARA principles. The shielding analyses initially indicated STC dose rates significantly exceeded those evaluated for transfer casks used in approved dry storage systems leading the staff to question how adequately ALARA was considered in the STC design. The applicant then revised its analysis; as revised, the analysis now concludes that the STC dose rates are within the range of those evaluated for dry storage transfer casks, which have been previously reviewed and found to be acceptable. Additionally, other features were added to the STC to enhance the design with respect to ALARA. These features include, among other things, the shield blocks on the STC vent and drain ports. Another ALARA feature is the shield ring on the STC lid for reducing streaming through the gap between the STC lid and top flange that is present during STC handling.

The operations descriptions include remote handling of the STC. The applicant plans to have redundant means available to allow remote handling of the STC. Remote handling of the STC enhances protection of personnel from the high dose rate areas on the STC that exist where the steel ribs penetrate the lead shielding. The SAR operations descriptions also include performance of activities in low dose rate areas and configurations, such as the STC shielded inside the HI-TRAC, when possible. The descriptions also include cautions with respect to activities with potential for increased dose rates (e.g., potential for activated particles in water being drained from the STC cavity) and suggestions for reducing dose rates and limiting contamination (e.g., wash down of equipment entering and exiting the SFP water). Based on the foregoing, the staff finds that ALARA principles have been adequately incorporated in 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.

3.7.4.8 Additional Radiation Protection Considerations

As described above, the proposed TS include dose rate limits and dose rate measurements for the STC and the HI-TRAC (see TS 5.4). The proposed limits are for the STC lid and the HI-TRAC side. This approach is acceptable because the proposed limits fulfill the purpose of dose rate limits since they are based on personnel proximity to the structural locations on the STC and HI-TRAC that have the highest dose rates while taking into consideration ALARA. The measurements associated with these limits are taken at a time when personnel are near these areas under normal conditions. Under normal conditions, personnel are near the loaded STC only when the STC is in the HI-TRAC or while all but the STC lid is submerged in the spent fuel pool. At the same time, having a limit that is on the STC lid enables identification of a problem at a stage in the operations that allows for quicker performance of corrective actions, if any are needed, and at an earlier stage than measurement against a limit on the HI-TRAC lid would afford. Additionally, personnel perform operations on the STC lid as part of normal operations. The limit values are derived from the bounding dose rates in the applicant's shielding and radiation protection evaluation, with some margin to allow for measurement uncertainty. Based on these considerations, the staff finds the limits to be acceptable.

The staff also reviewed the proposed dose rate measurement locations. The STC lid measurements account for differences in the loading configurations resulting in different locations of peak dose rates. The specified locations ensure that measurements are performed over loaded STC basket cells. The measurements on the HI-TRAC are at representative points around the circumference of the HI-TRAC. Each measurement is compared with its respective limit. Based on these considerations, the staff finds the measurements to be acceptable for confirming compliance with the dose rate limits. The staff finds that the proposed TS (TS 5.4) provides confidence that the dose limits for members of the public will not be exceeded. The operations descriptions in Chapter 10 of the SAR include steps to measure the STC and HI-TRAC dose rates and confirm compliance with the TS dose rate limits. The staff reviewed these descriptions and finds they are performed at a time in the operations sequence such that the configuration is consistent with the configuration that is the basis for the limits. This configuration is also specified in the TS.

The proposed TS (see Appendix C, Part I, Section 1.0) include minimum dimensions of the STC radial shielding. The TS also include similar information for the HI-TRAC. This information and the dose rate limits in the TS help to assure that the shielding capabilities of the STC and the HI-TRAC are within the assumptions in the applicant's analysis. The acceptance tests and maintenance program in Chapter 8 of the SAR include the means for verifying the shielding effectiveness of the STC and aspects of the HI-TRAC relied on for maintaining shielding effectiveness that are unique to the wet transfer operations. The acceptance tests and maintenance program for the HI-STORM 100 dry storage system address aspects of the HI-TRAC that are also relied on for shielding in dry storage operations.

The staff considered the need for a surface contamination TS LCO. This consideration stemmed from the similarity of the operations to loading operations for dry storage. However, the staff has concluded that a surface contamination TS LCO is not needed. As the applicant notes, the STC outside surface is not exposed to the environment outside of the FSBs, unlike a dry storage canister. The STC is transferred between the SFPs and the HI-TRAC, which are all within the IP2 and IP3 FSBs, a restricted area; thus, the contamination control measures for activities within these areas will be applied. These measures include performance of large area swipes of the STC and HI-TRAC surfaces to reduce and detect loose contamination and particles, to allow timely and appropriate personnel protection actions. For transfer between the spent fuel buildings, a solid lid is installed on the HI-TRAC. In this configuration, the HI-TRAC cavity is completely enclosed and the STC is isolated from the external environment. Therefore, any potential contamination is contained within the HI-TRAC.

During its review, the NRC staff considered the adequacy of the proposed TS to ensure adequate radiation protection for personnel and the public, based on the shielding design performance of the STC as indicated by the surface dose rates. As discussed above, the surface dose rates were calculated at one point in the review to be about 6.7 rem/hr (surface-average). The applicant subsequently changed the proposed contents and modified its analyses to more accurately reflect the proposed contents and STC shielding design. These modified analyses indicate that the STC surface dose rates are closer to 3.2 rem/hr (surface-average), which is within the range of surface dose rates for transfer casks of approved dry storage systems. The proposed TS are similar to those for approved dry storage systems with similar surface dose rates on the transfer casks. Thus, the staff finds that the currently proposed TS are sufficient with respect to shielding and radiation protection.

3.7.5 Evaluation Findings

The NRC staff reviewed the applicant's shielding and radiation protection evaluation for the wet transfer of IP3 spent fuel from the IP3 SFP to the IP2 SFP. For these operations, the applicant intends to use an STC that has been specifically designed for this activity together with the HI-TRAC transfer cask from the HI-STORM 100 dry storage system. Additional features that were specifically identified for this activity are used with the HI-TRAC as well. Based on its review of the amendment request, the NRC staff finds that:

- The SAR sufficiently describes the shielding and radiation protection design features and design criteria for the STC and HI-TRAC and the fuel transfer operations.

- The shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 100, and the intent of 10 CFR 72.104 and 10 CFR 72.106.

- The SAR sufficiently describes the methods for controlling and limiting occupational exposures for the proposed activity within the dose and ALARA requirements of 10 CFR Part 20 and these methods are acceptable. Additionally, the designs of the STC and HI-TRAC assist in meeting these requirements, and have been appropriately considered in the SAR shielding and radiation protection evaluations.

Thus, the NRC staff finds there is reasonable assurance that the wet spent fuel transfer operations meet the shielding and radiation protection requirements of 10 CFR Parts 20 and 50, and the intent of 10 CFR Part 72. This finding is based on the staff's review of the regulations, applicable codes and standards, the applicant's proposed operations and analyses, and the staff's confirmatory evaluations.

3.8 Materials Evaluation, Acceptance Tests, and Maintenance Program

3.8.1 Materials Evaluation

3.8.1.1 Structural and Confinement Materials

The NRC staff conducted a materials evaluation to ensure that the materials used in the structures, systems, and components (SSCs) important to safety conform to quality standards commensurate with their safety function. This evaluation also provided reasonable assurance that the material properties will be maintained during all conditions of operation, over the 40 year service life of the STC and HI-TRAC, that the materials are compatible with wet spent fuel loading, transfer and unloading operations, and that the materials used for criticality control and shielding are adequately designed and specified to perform their intended function. The regulatory bases for this part of the evaluation are the GDCs discussed in Section 2.0 of this safety evaluation, with guidance derived from 10 CFR 72.122(a), 72.124(b), 72.126, and 72.236(g)-(i).

The materials used in the STC design are the same as those used in the HI-TRAC 100D and the materials used in the STC fuel basket are the same as those used in the multi-purpose canister (MPC) basket of the HI-STORM 100 system (NRC Docket No. 72-1014). Those structures have been certified for use previously under 10 CFR Part 72.

The STC body is a thick-walled carbon steel (ASME SA-516, Grade 70, SA-515, Grade 70, SA-350, LF2, or SA-36), lead, carbon steel layered cylindrical vessel with a welded base plate and a bolted top lid, fabricated with a stainless steel weld overlay on the inner surfaces and Carboguard 890 coating on the outer surfaces. The NRC staff assessed the STC body materials based on the temperature limit of 150 °C (302 °F) under normal operations and 200 °C (392 °F) under off-normal or accident conditions (SAR Table 3.1.1), in a borated water environment. No chemical or galvanic reactions are expected.

The pressure boundary of the STC meets the stress limits of the ASME Code, Section III, Subsection ND (SAR Sections 1.3.1 and 3.1.4.1). Spent fuel canisters are normally constructed to ASME Code, Subsections NB or NC. The applicant identified and adequately reconciled the discrepancies between the rules for construction of ASME Code Subsection ND versus Subsection NC for the construction of the STC to ensure that the STC was constructed to acceptable quality standards. ASME Code Subsection NC requires full radiography of the weld joints; however, select radiography is required by ASME Code Subsection ND. The staff evaluated the difference in requirements for radiographic testing (RT) and found the use of ASME Code Subsection ND to be acceptable based primarily on the following considerations: 1) each STC transfer operation is expected to take less than 3 days, or a maximum of 30 days (SAR Table 8.2.1); 2) the applicant performed RT on approximately 80% of the STC welds; 3) the STC body has a large margin of safety (SAR Section 6.2.1.1); 4) the applicant provided a linear elastic fracture mechanics based analytical assessment of theoretical flaws in the welds that were not subject to RT at a reference temperature of -40 °C (-40 °F), demonstrating that no crack propagation would ensue; and 5) the applicant will leak-test the entire STC confinement boundary per ANSI N14.5 to "leak-tight" criteria.

The spent fuel basket has an open-ended honeycomb configuration of austenitic stainless steel plates (ASME Type 316, 316LN, 304, or 304LN) with panels of Metamic neutron absorber affixed to the plates under stainless steel sheathing. The edges of the plates are welded together. The arrangement of stainless steel and Metamic plates in the STC fuel basket is based on the NRC-certified MPC-32 canister (NRC Docket No. 72-1014). The material procurement, design, fabrication, and inspection of the STC basket are per ASME Code, Section III, Subsection NG. The austenitic stainless steel fuel basket was assessed based on the temperature limits of 385 °C (725 °F) under normal operations and 510 °C (950 °F) under off-normal or accident conditions. Negligible chemical or galvanic reactions involving the stainless steel are expected due to the low levels of halogens and sulfates contained in the SFP water. The Metamic neutron absorber was evaluated based on the temperature limit of 427 °C (800 °F) under normal operations and 538 °C (1000 °F) under off-normal or accident conditions.

The elastomeric seals used in the STC top lid are required to meet specific characteristics as stated in Table 8.2.2 of the SAR. The Parker O-Ring EPDM Rubber E0740-75 has been prequalified to satisfactorily meet these requirements. Proper selection and installation of adequate seals is essential to ensuring that the STC retains its contents. The STC seals were chosen based on their: compression and decompression characteristics over the temperature range of interest, springback adequacy, ability to withstand a borated water environment, radiation resistance, and suitability for the joints required to withstand impact loads. The STC seals have a temperature limit of 120 °C (248 °F) under normal operations and 127 °C (260°F) under off-normal and accident conditions. These seals have a limit to their useful life and will be tested and replaced as discussed in Section 3.8.3, below.

Intact fuel with a burn-up less than 55,000 MWd/MTU and a minimum cooling time of 10 years is eligible for transfer, with additional limitations as stated in TS 3.1.2. The fuel cladding temperature limit is 400 °C (752 °F) under normal operations and 570 °C (1058 °F) under off-normal and accident conditions, consistent with SFST-ISG-11, Rev. 3.

The HI-TRAC is an existing structure that is part of an approved (certified) dry storage system (NRC Docket 72-1014) and has a carbon steel, lead, and carbon steel layered cylindrical shell for gamma radiation shielding and an outer annulus which is filled with water for neutron shielding. The interior surfaces are coated with Thermaline 450 and the exterior surfaces are coated with Carboguard 890. The HI-TRAC temperature limit is 150 °C (302 °F) under normal operations and 200 °C (392 °F) under off-normal or accident conditions. In the LAR, the standard HI-TRAC lid is replaced with a solid circular lid, fabricated from ASME SA-516, Grade 70.

Proper selection and installation of the elastomeric seals used in the HI-TRAC top lid and bottom pool lid is essential to ensuring that the HI-TRAC retains the annulus water and acts as a secondary boundary for the STC contents. The HI-TRAC seals were chosen based on their compression and decompression characteristics over the temperature range of interest, spring-back adequacy, ability to withstand a borated water environment, and radiation resistance. The HI-TRAC seals have a temperature limit of 120 °C (248 °F) under normal operations, as well as off-normal and accident conditions. These seals have a limit to their useful life and will be tested and replaced as discussed in Section 3.8.3, below.

3.8.1.2 Neutron Absorber Panels (Metamic)

This portion of the NRC staff's review addresses the Metamic neutron absorber material in the STC. GDC 62, "Prevention of criticality in fuel storage and handling," states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Accordingly, the applicant must limit the potential for criticality in the FSB and the proposed fuel handling and storage system by physical systems or processes.

The STC is designed and shall be maintained with the Metamic neutron absorber panels with a minimum thickness of 0.102 inches. The B_4C in the Metamic neutron absorber will be between 31.5 wt% and 33.0 wt%, and the B_4C in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%.

In order to ensure that these specifications are maintained for the service life of the STC, a coupon surveillance program will be implemented to maintain surveillance of the Metamic neutron absorber material under the radiation, chemical, and thermal environment of the STC. The surveillance program will be implemented to monitor the performance of Metamic by installing a minimum of four bare coupons near the maximum gamma flux elevation (mid height) at no less than four circumferential downcomer areas around the STC fuel basket. At any time during its use the STC must have a minimum of one coupon installed in each quadrant. The Metamic coupons used for testing must have been installed during the entire operational history of the STC.

TS 5.2, "Metamic Coupon Sampling Program," which will be part of the new Technical Specification Appendix C, "Inter-Unit Fuel Transfer Technical Specifications," will include the following specifications for the coupon surveillance program:

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

(ii) Pre-characterization testing: Before installation, each coupon will be measured and weighed. The measurements shall be taken at locations pre-specified in the test program. Each coupon shall be tested by neutron attenuation before installation in the STC. The weight, length, width, thickness, and results of the neutron attenuation testing shall be documented and retained.

(iii) Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained.

(iv) The coupons shall be examined for any indication of swelling, delamination, edge degradation, or general corrosion. The results of the examination shall be documented and retained.

(v) The coupons shall be tested by neutron attenuation and the results compared with the pre-characterization testing data. The results of the testing shall be documented and retained. Results are acceptable if the measured value is within +/-2.5% of the value measured for the same coupon at manufacturing.

(vi) The coupons shall be returned to their locations in the STC unless anomalous material behavior is found. If the results indicate anomalous material behavior, evaluation and corrective actions shall be pursued.

The NRC staff reviewed the proposed coupon surveillance program to determine if the parameters measured, acceptance criteria, testing interval, and coupon placement in the STC will adequately represent the Metamic material in the STC racks. The staff determined that the surveillance program which includes visual, physical and confirmatory tests is capable of detecting potential degradation of the Metamic material in the rack that could impair its neutron absorption capability. Based on the use of the surveillance program, the original material qualification tests for Metamic, and operating experience with Metamic at other nuclear power plants, the staff concludes that the Metamic neutron absorbing material is acceptable for use in the STC at IP2 and IP3.

3.8.1.3 Conclusion for Materials

The NRC staff finds that the STC and HI-TRAC materials conform to quality standards commensurate with their safety function. The STC and HI-TRAC employ materials that are compatible with wet loading of spent nuclear fuel, transfer, and unloading operations and facilities. There is reasonable assurance that the materials will not degrade over the license period or react with one another during transfer. The material properties will be maintained during all conditions of operation so that the spent fuel can be safely transferred over the 40-year service life of the STC. The materials used for criticality control and shielding are adequately designed and specified to perform their intended function.

3.8.2 Acceptance Test Evaluation

The staff performed an evaluation of the acceptance test to ensure that the STC and HI-TRAC SSCs have acceptance standards commensurate with the safety function they are intended to perform. The applicant will examine and test the STC to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. The regulatory bases for this part of the evaluation are the GDCs discussed in Section 2.0 of this safety evaluation, 10 CFR Part 50, Appendix B, Criterion XI – Test Control, and guidance derived from 10 CFR 72.82(d), 72.122(a) and (f), 72.124(b), 72.162, 72.236(j), and 72.236(l).

In addition to the inspections and acceptance tests detailed for the HI-TRAC in the HI-STORM 100 FSAR, visual inspections, weld examination, structural tests, leakage tests, as well as component and material tests will be performed to ensure that the transfer system will perform as expected.

Visual inspections and measurements will be performed to ensure that the STC dimensions conform to the TS, including verification of: minimum thicknesses of safety related components, appropriate installation of neutron absorber panels, full coverage of corrosion inhibiting coatings and overlays, and proper cleanliness preparations. STC weld examinations will be performed in accordance with the licensing drawings in Section 1.5 of the SAR and applicable codes and standards.

The STC and HI-TRAC are required to undergo a hydrostatic pressure test to check structural integrity in accordance with the ASME Code, Section III, Subsection ND, Article ND-6000 at a minimum test pressure of 125% of the design pressure. The NRC staff reviewed the SAR to determine if the Code required pressure test should be 125% of the summation of the vessel internal pressure and the stress attributable to the weight of the contents since both vessels are being lifted by connections to their upper region. Section 8.4.3 of the SAR clarifies that the pressure test of the HI-TRAC would account for the weight of the contents (i.e., the loaded STC and water) by either inserting an appropriate deadweight or increasing the test pressure. The applicant explained that the STC's lid is not firmly attached until after it is placed in the HI-TRAC, therefore the internal pressure could not increase until the STC is in the HI-TRAC, and the current design pressure of the STC bounds the stress attributable to the weight of the contents. Once placed in the HI-TRAC, the STC rests on the bottom of the HI-TRAC and consequently its content is supported by the HI-TRAC during transfer. When the STC pressure test is performed, the STC is supported by either the top lid or the lifting trunnions while the hydrostatic pressure is at 125% of the design pressure, to maximize the loads on the confinement boundary. The staff finds this arrangement acceptable to ensure that ASME Code requirements are satisfied.

As part of its review, the NRC staff witnessed a one-time STC hydrostatic pressure test, which was performed in accordance with ASME Code, Section III, Subsection ND, Article ND-6000 at a minimum test pressure of 125% of the design pressure, a test pressure greater than or equal to 62.5 psig, while the STC was supported by the lifting trunnions. The pressure test was completed satisfactorily. The HI-TRAC will also be subjected to a one-time hydrostatic pressure test with a minimum test pressure of 125% of the design pressure, a test pressure of 37.5 psig, while being supported by its trunnions with the equivalent weight of the loaded STC and water inside the HI-TRAC, prior to the first fuel loading. The STC lifting trunnions and lifting attachment will be load tested in accordance with the requirements of ANSI N14.6, at 300% of the maximum design lifting load of 40 tons.

For leakage testing of the confinement boundary of the STC, the NRC staff considers it appropriate that the leak testing be similar to cask transportation requirements rather than storage requirements, due to the repeated use of the STC to support fuel transfer from IP3 to IP2. In this regard, the staff questioned the initial assumptions in Section 7.4.5, "Effluent Dose Evaluation," of the SAR. The first off-normal condition considered in the SAR was a breakdown of the cask transporter without HI-TRAC recoverability for 30 days, in which the applicant only assumed 1% fuel rod breach for the dose calculations. This assumption was in conflict with Table 5-2 of NUREG-1536 Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," which identified the percent of spent fuel postulated to fail for off-normal conditions as 10%. In lieu of providing the dose impact of this typical 10% postulated failed fuel assumption, the applicant decided to make the STC confinement boundary leaktight as defined in ANSI N14.5-1997 (i.e. 1×10^{-7} ref-cm^3/ sec). Accordingly, the entire STC confinement boundary will be initially tested to the leaktight criteria with no fuel in the STC after fabrication. Within 12 months prior to each transfer, the STC seals (lid and lid cover plates) will be tested to the leaktight criteria. After loading spent fuel, the applicant will test the STC seals to ensure that they are properly seated by ensuring no detected leakage when tested to a sensitivity of 1×10^{-3} ref-cm^3/ sec. The staff finds these proposed leaktight criteria acceptable for compliance with GDC 61. There will be no effluent contribution to the site boundary dose calculation, which negates the need to perform effluent dose calculations.

During the course of the staff's review, the applicant made responsive changes to the design of the STC that ensures the integrity of the STC confinement boundary. Relief valves were removed from the STC to eliminate a leakage pathway. Also, the vent and drain valves were recessed into the lid of the STC to prevent them from being damaged during loading operations and possibly compromising the confinement boundary. With these design changes, the staff finds reasonable assurance that the STC confinement boundary will maintain its design basis integrity.

The HI-TRAC pressure boundary will be tested to ensure that it is water tight to prevent the loss of water from the annular region during fuel transfer operations. The HI-TRAC top lid seals will be leak tested using the soap bubble test method on the main seal and the gas pressure rise method on the vent port cover per ANSI N14.5, or other leak test methods permitted by ANSI N14.5.

Several component and material acceptance tests will be performed to ensure adequate performance. Brittle fracture testing will be performed per ASME Code, Section III, Subsection ND, as required, for the ferritic steels used in the STC.

The NRC staff also reviewed the acceptance testing for the lead shielding components of the transfer system. This review focused mainly on the lead shielding for the STC. The acceptance testing for the lead shielding in the HI-TRAC is stated in the FSAR for the HI-STORM 100 dry storage system and has been reviewed and found acceptable as part of the HI-STORM 100 Certificate of Compliance (CoC) and its amendments. Nothing in the wet transfer operations proposed in the license amendment challenges that finding. Thus that CoC finding is acceptable for this review.

The STC lead shielding is fabricated by a layering of lead sheets to achieve the minimum total lead thickness specified in the technical drawings and proposed TS. Given the design of the STC, there are four areas that are separated by steel ribs where the lead shielding is placed in the STC wall. Each lead layer may consist of a single sheet or multiple sheets. If multiple sheets are used, the layers are formed such that the edges of the lead sheets in adjacent layers do not coincide with each other. This pattern prevents the formation of streaming paths. The SAR also states that the lead sheets are trimmed so that they fit in the space between the steel ribs. Any gaps between the lead sheets and the ribs are filled with compressed lead wool.

The NRC staff questioned the use of lead wool and the acceptance tests used to ensure that the level of shielding provided in these areas would meet the design criteria. The applicant indicated that the gaps where lead wool was used were small. The applicant also proposed using a gamma scan to ensure the as-fabricated shielding was adequate. The scan would include comparison of measurements on the STC with those from a test block with the same shielding configuration and the materials at the minimum thicknesses. For scan measurements adjacent to a steel rib, the rib itself would be the standard. Given the small size of the gaps and the results of the measurements on the as-fabricated STC (as stated in the applicant's responses to staff's requests for additional information) as well as accounting for how normal operations will be conducted (i.e., personnel will be located at significant distances from a loaded STC during STC handling outside the pool and HI-TRAC), the staff finds the acceptance tests for the STC lead shielding to be acceptable.

In summary, the NRC staff finds the STC and HI-TRAC SSCs employ acceptance standards that are commensurate with the safety function they are intended to perform. The applicant will examine and/or test the STC to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness.

3.8.3 Maintenance Program Evaluation

The staff evaluated the maintenance program to ensure that the STC and HI-TRAC SSCs employ maintenance standards commensurate with the safety function they are intended to perform. The regulatory bases for this part of the evaluation are the GDCs discussed in Section 2.0 of this safety evaluation, with guidance derived from 10 CFR 72.82(d), 72.122(a) and (f), 72.236(g), (j), and (l).

The STC and HI-TRAC are designed for a 40-year service life and will be subject to an ongoing maintenance program over this period to ensure continued effective performance.

As part of the maintenance program, accessible surfaces will be visually inspected for surface and component damage. This visual inspection will include the accessible surfaces of the STC and HI-TRAC, closure bolts, lifting devices, and closure seals, prior to each fuel transfer. It will also include inspections of seal seating surfaces prior to installation or replacement. Dimensional inspections of the lifting devices and trunnions will be performed once every 12 months or prior to each loading campaign if the transfer program is suspended. STC closure bolts will be replaced every 240 bolting cycles and HI-TRAC top lid bolts will be replaced every 1000 bolting cycles. A bolting cycle is one tightening and loosening of a set of bolts. A safety evaluation will be performed for all damage that is found, and components will be repaired or replaced as appropriate. If repair or replacement affecting the pressure retaining function is necessary, the structural pressure tests described in Section 3.8.2 above, and Section 8.4.3 of the SAR, will be repeated.

Several maintenance leakage tests will be performed to ensure adequate sealing functionality. Leakage tests of the STC seals to "leak-tight" per ANSI N14.5 will be performed within 12 months of each fuel transfer and following seal replacement. Following each fuel loading, prior to fuel transfer, a leakage test will be performed to a sensitivity of 1.0×10^{-3} ref-cc/s. The STC lid seal will be replaced every 6 fuel transfers or as necessary based on visual inspection results or failure to seal. Additionally, leakage tests of the HI-TRAC top lid seals to 1.0×10^{-3} ref-cc/s will be performed following each fuel loading, prior to fuel transfer. Leakage tests of the HI-TRAC bottom pool lid seal and drain plug will be performed prior to each loading campaign, not to exceed 12 months. The HI-TRAC top lid and bottom pool lid seals will be replaced prior to each fuel transfer campaign or as necessary based on visual inspection results or failure to seal.

A surveillance program will be implemented to monitor the performance of Metamic by installing a minimum of four bare coupons near the maximum gamma flux elevation. Test coupons must be installed during the entire fuel loading history of the STC. Four coupons will be tested at the end of each fuel transfer campaign. A fuel transfer campaign shall not last longer than two years as specified in TS 5.2 (iii).

The NRC staff finds that the STC and HI-TRAC SSCs employ maintenance standards commensurate with the safety function they are intended to perform.

3.8.4 Conclusions

The NRC staff concludes that the material properties of the SSCs of the STC and HI-TRAC are in compliance with 10 CFR Part 50, and that the applicable design and acceptance criteria have been satisfied. The STC and HI-TRAC employ materials that are compatible with wet loading of spent nuclear fuel, transfer, and unloading operations and facilities. There is reasonable assurance that the materials will not degrade over the license period or react with one another during transfer. The materials used for criticality control and shielding are adequately designed and specified to perform their intended function. The evaluation of the material properties provides reasonable assurance that the STC and HI-TRAC will allow safe transfer of spent fuel over a service life of 40 years. This finding is reached on the basis of a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

The NRC staff concludes that the acceptance tests and maintenance program for the STC and HI-TRAC are in compliance with 10 CFR Part 50 and that the applicable acceptance criteria and maintenance program criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the STC and the HI-TRAC will allow safe transfer of IP3 spent fuel to the IP2 SFP for the service life of 40 years. This finding is reached on the basis of a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

Further, the NRC staff finds that the STC confinement boundary will maintain its ability to retain radioactive materials and prevent effluents and comply with regulations in GDC 61, "Fuel Storage and Handling and Radioactivity Control," and 10 CFR Part 50, Appendix B, Criterion XI – Test Control.

In summary, the NRC staff concludes that there is reasonable assurance with regard to the materials, acceptance tests, and maintenance program that the proposed fuel transfer process will not endanger the public health and safety.

3.9 Environmental Considerations

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared by the NRC staff (ADAMS Accession No. ML120740550). Based upon the environmental assessment, the NRC has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

3.10 Operating Procedures

3.10.1 Objective

The objective of this portion of the staff's evaluation is to assess the operations procedures, including the preoperational testing and training exercise procedures, provided by the applicant in the SAR to ensure the safe transfer of IP3 spent fuel to the IP2 SFP. Applicable sections of NUREG-1567, Chapter 10, "Conduct of Operations," were used by the NRC staff as guidance in performing this evaluation.

3.10.1 Operating Procedures Evaluation

The applicant provided the following requirement in the proposed TS: "Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, maintenance, and recovery from off normal conditions such as crane hang-up. The written operating procedures shall be consistent with the technical basis described in Chapter 10 of the Licensing Report (Holtec International Report HI-2094289) [SAR]." The applicant also stated that "...Entergy confirms that the procedures that govern the inter-unit transfer of fuel have been written in accordance with existing Entergy standards for procedure development and are therefore appropriately human factored." The NRC staff has evaluated Chapter 10 of the SAR and has determined that the procedure guidelines are thorough and are acceptable.

3.10.2 Pre-Operational Testing and Training Exercise Evaluation

The applicant has provided the following requirements in its proposed TS:

> A training exercise of the loading, closure, handling/transfer, and unloading, of the equipment shall be conducted prior to the first transfer. The training exercise shall not be conducted with irradiated fuel. The training exercise may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The training exercise shall include, but is not limited to the following:
>
> a) Moving the STC into the IP3 spent fuel pool.
> b) Preparation of the HI-TRAC for STC loading.
> c) Selection and verification of specific fuel assemblies and non-fuel hardware to ensure type conformance.
> d) Loading specific assemblies and placing assemblies into the STC (using a single dummy fuel assembly), including appropriate independent verification.
> e) Remote installation of the STC lid and removal of the STC from the spent fuel pool.
> f) Placement of the STC into the HI-TRAC with the STC centering assembly.
> g) STC closure, establishment of STC water level with steam, verification of STC water level, STC leakage testing, and operational steps required prior to transfer, as applicable.
> h) Establishment and verification of HI-TRAC water level.
> i) Installation of the HI-TRAC top lid.
> j) HI-TRAC closure, leakage testing, and operational steps required prior to transfer, as applicable.
> k) Movement of the HI-TRAC with STC from the IP3 fuel handling building to the IP2 fuel handling building along the haul route with designated devices.
> l) Moving the STC into the IP2 spent fuel pool.
> m) Manual crane operations for bare STC movements including demonstration of recovery from a crane hang-up with the STC suspended from the crane.

The NRC staff has reviewed the TS requirements for conduct of a training exercise and has determined that they are comprehensive and cover the major aspects required by the IP3 on-site spent fuel transfer evolution. The TS are consistent with those the staff requires for similar 10 CFR Part 72 Site Specific and General License spent fuel handling and storage evolutions, as well as the guidelines of NUREG–1567, Chapter 10. The staff therefore finds these proposed requirements acceptable.

3.10.3 Conclusions

Based upon its review of the LAR, as discussed above, the NRC staff concludes as follows:

- The SAR includes an acceptable plan for the conduct of operations which conforms to 10 CFR 72.24(h), and is acceptable.

- The TS include an acceptable description of the program covering preoperational testing and initial operations, which conforms to 10 CFR 72.24(p), and is acceptable.

- The application provides reasonable assurance with regard to the management, organization, and planning for preoperational testing and initial operations, that the activities authorized by the license amendment can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 50.92.

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. The State official provided detailed comments in a letter dated February 17, 2012, ADAMS Accession No. ML120650735. The following discussion addresses the State's comments, most of which are quoted from the summary section of the State's letter:

State Comment 1:

Although the NRC and Entergy have addressed and resolved numerous issues that were of concern to the State, we remain primarily concerned with the multiple interactions needed for this operation given the potential for accidents and the fact that accidents have occurred in the past. Paramount to these concerns is the multiple handling of the spent fuel. This is the only plant in this country, and to the best of our research, in the world, that will have to move fuel in the proposed manner. Normally, fuel would be handled only once as it is moved from the spent fuel pool directly into dry cask storage and transported to an Independent Spent Fuel Storage Installation (ISFSI). However, because of the inability to move fuel directly to dry cask storage, the proposed method would cause each spent fuel assembly to be handled three times before it reaches the ISFSI. This increase in handling likewise increases the risk of an accident.

Response 1:

The NRC staff agrees with the general principle that efforts should be made to minimize the handling of spent fuel as much as possible. However, spent fuel assemblies typically are exposed to more handling than the State is aware of. For example, each fuel assembly is typically used for three core cycles, and many applicants completely unload the reactor vessel to the SFP during refueling outages. This results in a minimum of five handlings (as spent fuel) prior to handling for dry cask storage. Nor is Indian Point the only facility to transfer spent fuel between its SFPs. The NRC has previously given approval to several applicants to transfer spent fuel from one SFP to another SFP, including Carolina Power and Light, Southern California Edison Co., Duke Power, and Florida Power and Light. The unique aspect of this application is the use of a water-filled transfer cask licensed under 10 CFR Part 50, rather than a dry cask licensed under 10 CFR Part 71. The NRC staff finds that the amount of fuel handling involved is not excessive when compared to previous approvals, and there is reasonable assurance that the proposed transfer operations will not endanger public health and safety.

State Comment 2:

NRC should develop a supplemental license amendment that will address movement of damaged fuel.

Response 2:

The NRC staff acknowledges that Entergy cannot move non-intact fuel out of the IP3 SFP under this amendment, and that eventually some means of moving non-intact fuel must be developed. However, Entergy may decide to implement another method for moving non-intact fuel, such as a different transfer cask. In that event, this issue will need to be addressed in the future, as appropriate.

State Comment 3:

Pre-move inspection of the assemblies planned to be moved into each 12 assembly (max) load should be performed before any assembly is moved.

Response 3:

Fuel assemblies in the IP3 SFP have been inspected to characterize them as intact or not. Under the applicant's procedures, prior to any fuel movement, fuel move sheets will be written up by a reactor engineer and independently verified. Prior to moving a fuel assembly from its SFP storage rack, fuel handling personnel have a first and second checker verify that the fuel assembly serial number matches that on the approved fuel move sheet. After the fuel assembly is placed in the STC, a peer checker verifies it is in the correct cell location as specified on the approved fuel move sheet. After the STC is loaded, it undergoes a 24 hour heatup test as an independent verification that the fuel assemblies that were loaded do not exceed the thermal design limits of the STC. When the STC reaches the IP2 SFP, the applicant will again utilize a fuel move sheet and first and second checkers prior to placing a fuel assembly in the IP2 SFP. The NRC staff has approved this quality control process in the loading of the STC and the transfer of the fuel assemblies.

State Comment 4:

Before removal from IP3 SFP, an independent verification by visual inspection, with peer check, must be performed to assure that none of the assemblies being removed are fresh fuel assemblies.

Response 4:

Refer to Response 3 for details on selecting fuel assemblies. Also, because the STC has not been approved to transport fresh fuel assemblies, the NRC staff has required that the TS for the IP3 license (Appendix C, Inter-Unit Fuel Transfer Technical Specifications) state that loading of the STC is only permitted when there are no fresh fuel assemblies in the SFP.

State Comment 5:

All workers associated with movement of spent fuel, from the preparatory stages at Unit 3 to the final placement in the ISFSI, should be subject to the NRC's rules for covered workers.

Response 5:

The NRC's Fitness For Duty (FFD) rules are specified in 10 CFR Part 26. All workers who enter the FSB unescorted are required to be enrolled in the FFD program, and must pass a background investigation, random drug and alcohol tests, and behavioral observations. In addition, there must be at least one supervisor on site who is directly licensed by the NRC as a senior operator (or a senior operator limited to fuel handling) when fuel is being transferred. All workers who enter the FSB unescorted are also covered by part or all of the NRC's rules for managing fatigue. Although some of these workers, depending on the safety significance of their job, may not be subject to the NRC limits on hours worked per week and breaks between shifts, they will all be subject to fatigue assessments if they are observed to show impaired alertness.

State Comment 6:

Strict procedure compliance and peer checking is required for each assembly's pre-selected location and orientation before and after placement in the STC and when spent fuel is removed from the STC and placed into the Unit 2 pool.

Response 6:

Refer to Response 3 for the details on how peer checking is performed for fuel movement. The NRC staff finds that the applicant's procedures are in accord with the recommendation in State Comment 6.

State Comment 7:

Fuel movement should be accomplished only during daylight hours.

Response 7:

Nuclear power plant operations are conducted around the clock. For the nuclear power plants the NRC staff finds that the environment (lighting, ventilation, heating and air conditioning, etc), and fatigue considerations are sufficiently controlled that it is not necessary to restrict fuel movement to daylight hours.

State Comment 8:

Briefings of all workers involved in fuel movements should be conducted daily at the start of each shift.

Response 8:

The NRC staff acknowledges that briefings are generally helpful in ensuring that workers know their duties. Entergy does conduct pre-briefings for complex and risk-significant procedures and evolutions. Since the actions to be taken during fuel transfer will be proceduralized, with appropriate oversight from a licensed operator, the NRC staff finds that it is not necessary to impose a requirement for additional briefings. The NRC staff will conduct inspections of the fuel transfer activity, and if deficiencies are noted appropriate action will be taken.

State Comment 9:

In addition to a straight time factor, a frequency of use factor should also be applied to inspection of all fuel handling and transport equipment at both Units.

Response 9:

The NRC staff acknowledges the State's recommendations, and believes that the applicant's proposal meets the State's suggestion. In this regard, the staff has approved the following inspection criteria, which include both straight-time factors and frequency-of-use factors:

a. Prior to each fuel transfer, the external and internal surfaces of the STC and the HI-TRAC shall be visually inspected for damage, along with their closure bolts and lifting devices. Any evidence of deformation, distortion, or cracking will require repair of the equipment.
b. Prior to each fuel transfer, closure seals and sealing surfaces will be inspected for conditions that affect sealing capability. Damaged seals shall be replaced.
c. A leakage test of the STC seals to "leaktight" criteria shall be conducted within the 12 months prior to each fuel transfer. This leakage test of the STC seals shall also be conducted if the seals are replaced.
d. A leakage test on the STC and HI-TRAC seals to demonstrate correct seating shall be conducted just prior to fuel transfer.
e. STC and HI-TRAC lifting trunnions and lifting attachments shall be inspected annually in accordance with ANSI N14.6.
f. The Metamic coupons (neutron absorber) shall be tested at the end of each fuel transfer campaign, not to exceed two years.
g. The VCT and FSB cranes will be periodically inspected and tested in accordance with their design basis documents.

State Comment 10:

Planning for fuel movement campaigns should include sufficient time to allow for possible weather delays.

Response 10:

The NRC staff agrees that weather should be a consideration. The IP2 and IP3 licenses have been modified to state that fuel transfer operations can only be conducted when outside air temperatures are ≥ 0 °F and ≤ 100 °F. The applicant has procedures that restrict risk-significant activities during high winds, hurricanes, or other adverse weather conditions.

State Comment 11:

A shielded transfer cask should not be used as a storage cask.

Response 11:

The NRC staff agrees with this comment. The STC is not intended or approved for use as a storage cask. In the case of a VCT breakdown, the STC has been analyzed to safely hold the fuel for up to 30 days, but the applicant is expected to fix the VCT or acquire another and complete the fuel transfer prior to expiration of the 30 day period.

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 and changes surveillance requirements. Pursuant to 10 CFR Part 51, an environmental assessment was issued in the Federal Register on July 13, 2012 (77 FR 41454), in which the NRC staff issued a finding of no significant impact, and concluded that the environmental impacts are small and that there was no need to issue an environmental impact statement.

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) 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.

7.0 REFERENCES

1. J. E. Pollock, Entergy Nuclear Operations, Inc., letter to USNRC document control desk, re: "Application for Unit 2 Operating License Condition Change and Units 2 and 3 Technical Specification Changes to Add Inter-Unit Spent Fuel Transfer Requirements," Indian Point Units 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, July 8, 2009. (ADAMS Accession No. ML091940177)

2. J. E. Pollock, Entergy Nuclear Operations, Inc., letter to USNRC document control desk, re: "Response to Request for Additional Information Regarding the Inter-Unit Spent Fuel Transfer License Amendment Request (TAC Nos. ME1671, ME1672, and L24299)," Indian Point Units 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, December 15, 2011. (ADAMS Accession No. ML12013A259)

3. John A. Ventosa, Entergy Nuclear Operations, Inc., letter to USNRC document control desk, re: "Response to Request for Additional Information Regarding the Inter-Unit Spent Fuel Transfer License Amendment Request (TAC Nos. ME1671, ME1672, and L24299)," Indian Point Units 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, March 2, 2012. (ADAMS Accession No. ML12074A027)

4. L. Kopp, NRC, memorandum to T. Collins, NRC, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. (ADAMS Accession No. ML003728001)

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

6. Information Notice 2011-03, "Nonconservative Criticality Safety Analyses For Fuel Storage," February 16, 2011. (ADAMS Accession No. ML103090055)

7. Holtec International, "Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at the Indian Point Energy Center," HI-2094289, Revision 6, April 23, 2012. The proprietary version is in ADAMS at ML121300164, the public version at ML12137A201.

8. NRC, "HI-STORM 100 Cask System Safety Evaluation Report (Issued May 2000 with CoC 1014)," May 4, 2000. (ADAMS Accession No. ML003736794)

Principal Contributors: J. Solis, S. Jones, M. Call, K. Wood, Z. Li, N. Day, S. DePaula, M. Yoder, D. Hoang, H. Lindsay, J. Goshen, J. Boska

Date: July 13, 2012

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

Sincerely,

/ra/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. Amendment No. 268 to DPR-26
2. Amendment No. 246 to DPR-64
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

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*Concurrence via email; **Safety evaluation input provided by memo

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