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NL-09-076

July 08, 2009

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station O-P1-17
Washington, DC 20555-0001

Subject: **Indian Point Nuclear Power Plant Units 2 and 3**
Application for Unit 2 Operating License Condition Change and Units 2 and 3 Technical Specification Changes to Add Inter-Unit Spent Fuel Transfer Requirements
Indian Point Units 2 & 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby requests amendments to an Operating License (OL) condition and the Technical Specifications (TS) for Indian Point Unit 2 and an amendment to the TS for Indian Point Unit 3. Specifically, this request proposes to:

- Revise existing Unit 2 OL Condition 2.B.(5);
- Add new Unit 2 OL Condition 2.P;
- Add new Unit 2 LCO 3.7.15;
- Add new Unit 2 TS 4.4;
- Add a new definition of INTACT FUEL ASSEMBLIES to the Unit 3 TS
- Add new Unit 3 LCO 3.7.18; and
- Add new Unit 3 TS 4.4

These changes provide the necessary controls and permission required for Entergy to move spent fuel from the Unit 3 spent fuel pit to the Unit 2 spent fuel pit. Once in the Unit 2 spent fuel pit, the Unit 3 spent fuel will subsequently be moved into dry storage at the existing on-site Independent Spent Fuel Storage Installation (ISFSI). Entergy operates the ISFSI under the general license provisions of 10 CFR 72, Subpart K (Docket 72-051) and is authorized to store spent fuel from Indian Point Units 1, 2, and 3 at the ISFSI under the Part 72 general license. Entergy uses the NRC-certified HI-STORM 100 System (Docket 72-1014) for dry spent fuel storage at the ISFSI. Placing Unit 3 spent fuel directly into dry storage casks from the Unit 3 spent fuel pit is not possible due to the 40-ton cask handling crane capacity. A cask handling crane capacity of at least 100 tons is required to lift and handle the loaded HI-TRAC transfer cask licensed as part of the HI-STORM 100 System. An upgrade to the Unit 3 crane capacity to 100 tons or more was evaluated and found to be not feasible and as such results in the need for inter-unit fuel transfer.

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Approval of these changes is needed to restore and maintain full core offload capability in the Unit 3 spent fuel pit for the remainder of its service life and to allow for periodic receipt of new fuel in support of future refueling outages.

This submittal consists of two License Amendment Requests (LARs), one each for Unit 2 and Unit 3. The inter-unit transfer of spent fuel from Unit 3 to Unit 2 will be conducted using a Shielded Transfer Canister (STC), the design of which is being licensed under these proposed amendments and a HI-TRAC transfer cask that has been certified for dry spent fuel storage cask loading operations under 10 CFR 72 and is slightly modified for this application by the addition of a specially designed closure lid. The nature of those modifications is described in Attachment 1 and Enclosure 1. The STC design is supported by appropriate structural, shielding, criticality, thermal, and materials evaluations summarized in the Attachment 1 and Enclosure 1. The inter-unit transfer operation is governed entirely by 10 CFR 50 regulations and does not require licensing action under 10 CFR 72. However, in some cases where design guidance and/or acceptance criteria do not exist in Part 50, appropriate guidance/criteria from 10 CFR 72 (and 10 CFR 71) are used and the source and justification for their use is provided.

This submittal includes information deemed proprietary by an entity that is providing support to Entergy on this project. As such, a 10 CFR 2.390 affidavit has been executed by the owner of the information and a non-proprietary version of the proprietary document is also included herein.

This submittal is organized as follows:

- Attachment 1: Analysis of proposed operating license condition and technical specification changes pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Attachment 2: Marked-up IP2 operating license conditions, technical specifications and technical specification bases pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Attachment 3: Marked-up IP3 technical specifications and technical specification bases pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Attachment 4: No significant hazards consideration for Indian Point Unit 2 pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Attachment 5: No significant hazards consideration for Indian Point Unit 3 pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Attachment 6: New commitments pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
- Enclosure 1: Holtec International Licensing Report HI-2094289, Revision 1 (Holtec-Proprietary)
- Enclosure 2: Non-proprietary version of the proprietary document in Enclosure 1.
- Enclosure 3: Affidavit executed pursuant to 10 CFR 2.390 governing the proprietary information included in the Holtec Licensing Report.

Entergy has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) and has determined that these proposed changes involve no significant hazards considerations, as described in Attachments 4 and 5 for IP2 and IP3, respectively.

Entergy requests approval of the proposed license amendments by August 31, 2010, to support inter-unit fuel transfer operations in the fourth quarter of 2010. Inter-unit fuel transfer of 96 fuel assemblies is required to permit full core offload in the next IP3 refueling outage scheduled to begin in March 2011.

In accordance with 10 CFR 50.91, a copy of this application, with attachments and enclosures is being provided to the designated New York State official.

New regulatory commitments made in this submittal are identified in Attachment 6 to this letter.

If you have any questions or require additional information, please contact Mr. Robert Walpole, Licensing Manager at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge. Executed on July 8, 2009.

Sincerely,

JE Pollock

JEP/rw

Attachments:
1. Analysis of proposed operating license condition and technical specification changes pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
2. Marked-up IP2 operating license conditions, technical specifications and technical specification bases pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
3. Marked-up IP3 technical specifications and technical specification bases pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
4. No significant hazards consideration for Indian Point Unit 2 pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
5. No significant hazards consideration for Indian Point Unit 3 pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2
6. New commitments pertaining to the inter-unit transfer of spent fuel from Indian Point Unit 3 to Indian Point Unit 2

Enclosures:
1. Holtec International Licensing Report HI-2094289, Revision 1 (Holtec-Proprietary)
2. Non-proprietary version of the proprietary document in Enclosure 1.
3. Affidavit executed pursuant to 10 CFR 2.390 governing the proprietary information included in the Holtec Licensing Report.

cc: NRC Resident Inspector's Office
Mr. John Boska, Senior Project Manager, NRC NRR DORL
Mr. Theodore Smith, Project Manager, NRC FSME DWMEP DURLD
Mr. Samuel Collins, Regional Administrator, NRC Region 1
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA (w/o proprietary information)
Mr. Paul Eddy, New York State Dept. of Public Service (w/o proprietary information)
Mr. John White, Branch Chief, NRC Region 1
Mr. Tim Rice, New York State DEC (w/o proprietary information)

ATTACHMENT 1 TO NL-09-076

ANALYSIS OF PROPOSED OPERATING LICENSE CONDITION AND TECHNICAL SPECIFICATION CHANGES PERTAINING TO THE INTER-UNIT TRANSFER OF SPENT FUEL FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

Entergy Nuclear Operations, Inc.
Indian Point Units 2 and 3
Docket Nos. 50-247 and 50-286

1.0 DESCRIPTION

This letter is a request to amend Operating License Nos. DPR-26 for Indian Point Unit 2 (IP2) and DPR-64 for Indian Point Unit 3 (IP3).

The inter-unit spent fuel transfer operation was deemed as the most effective solution for relocating the IP3 fuel from the spent fuel pool into dry storage. To load fuel directly into a currently approved dry storage cask system (Part 72) from the IP3 pool would require upgrading the IP3 cask handling crane to at least 100 tons. This option was evaluated and found to be not feasible. Using a currently approved dry transportation cask which is less than 40 tons (Part 71) would significantly increase the number of fuel transfer evolutions and would place undue stress on the fuel as a result of an extra drying and subsequent rewetting of the fuel prior to dry storage. Additionally, more than half of the fuel currently in the IP3 pool does not meet the burnup and enrichment limitations of an approved 40 ton transportation cask. Therefore, in consideration of the above, the inter-unit fuel transfer is the most viable option.

The changes proposed in this LAR are necessary to permit inter-unit fuel transfer summarized in the following major activities:

- Loading up to twelve IP3 spent fuel assemblies into a specially designed, lightweight Shielded Transfer Canister (STC) in the spent fuel pit (the STC is a thick walled cylindrical container that is compatible with the HI-TRAC transfer cask and serves as the enclosure for wet transfer of the IP3 fuel to IP2 pit):

- Moving the fuel-loaded STC into a dry storage transfer cask (HI-TRAC 100D) in the IP3 Fuel Storage Building (FSB) truck bay using an upgraded, single-failure-proof IP3 40-ton cask handling crane (the annular space between the outside surface of the STC and the inside surface of the HI-TRAC 100D will also be filled with water),

- Moving the HI-TRAC/STC assemblage out of the IP3 FSB truck bay on air pads,

- Moving the HI-TRAC/STC assemblage to the IP2 FSB truck bay entrance with a vertical cask transporter (VCT),

- Moving the HI-TRAC/STC assemblage into the IP2 truck bay on a low profile transporter (LPT),

- Removing the STC from the HI-TRAC and placing it in the IP2 spent fuel pit using the single-failure-proof IP2 110-ton capacity cask handling crane,

- Moving the IP3 fuel assemblies from the STC into the IP2 spent fuel pit storage racks.

This request proposes to amend the IP2 operating license and Technical Specifications (TS) and the IP3 TS to permit the transfer of spent fuel from IP3 to IP2 on an as-needed basis as determined by Entergy's fuel management needs.

2.0 **PROPOSED CHANGES**

Attachment 2 to this submittal contains the marked-up pages showing the specific proposed changes to the IP2 Operating License, Technical Specifications, and Technical Specification Bases. Attachment 3 to this submittal contains the marked-up pages showing the specific proposed changes to the IP3 Technical Specifications and Technical Specification Bases. The changes are summarized below.

2.1 IP2 Operating License Changes

The conditions of IP2 Operating License (OL) are proposed to be changed in two ways.

1. OL Condition 2.B.(5) presently states:

   > 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.

   This request proposes to modify OL Condition 2.B.(5) to state (new words in italics):

   > 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).*

2. New OL Condition 2.P is proposed, which states:

   > *P. Indian Point Unit 3 spent fuel may be transferred to the Indian Point Unit 2 spent fuel pit, as needed.*

The proposed change to OL Condition 2.B.(5) clarifies that fuel burned in the IP3 reactor is permitted to be transferred to, and received and possessed by IP2 under the IP2 operating license. Proposed new OL Condition 2.P clarifies that ENO may perform inter-unit spent fuel transfers solely from IP3 to IP2 as determined by Entergy's fuel management needs.

The specific requirements and limits applicable to fuel selection, loading and unloading the STC are proposed in the changes to the respective plants' TS. Entergy is required to comply with the plants' TS by Condition 2.C.(2) in each plant's Operating License.

2.2 IP2 Technical Specification Changes

The following changes are proposed to the IP2 TS:

1. A new TS Limiting Condition for Operation (LCO) is proposed, LCO 3.7.15 *Shielded Transfer Canister (STC) Unloading*, to provide controls on the STC and the IP3 fuel while in the IP2 spent fuel pit. This LCO ensures that the IP3 fuel in the IP2 spent fuel pit is always in an authorized location. Associated with this LCO is a new proposed Bases B 3.7.15.

2. A new Surveillance Requirement (SR), SR 3.7.15.1 is proposed to ensure any IP3 fuel assembly removed from the STC in the IP2 spent fuel pit that needs to be returned to the STC before being inserted into the IP2 spent fuel racks get placed in the same STC cell from which it was removed.

3. A new TS Design Feature 4.4 *Shielded Transfer Canister* is proposed to provide a limit for the maximum enrichment of any single fuel assembly to be loaded into the STC, the acceptance criterion for the maximum analyzed reactivity of the STC, a nominal center-to-center fuel assembly spacing dimension in the STC, a limit on the fuel assembly capacity of the STC, and certain requirements for drainage of the STC.

In summary, the IP2 TS are proposed to be changed to provide the appropriate limits and controls on the STC and its contents while in the IP2 spent fuel pit. The proposed IP2 TS changes do not permit the loading of an IP3 fuel assembly into the STC once the assembly has been seated in a spent fuel rack location and released from the spent fuel bridge crane. This approach has the intended effect of prohibiting transfer of IP3 fuel, once in the IP2 spent fuel racks, back to the IP3 spent fuel pit. However, the TS do provide the necessary operational flexibility if, for some unanticipated reason, an IP3 fuel assembly cannot be placed in the fuel rack location specified on the fuel move sheet, to return the assembly to a safe, analyzed location in the STC. This flexibility will permit Entergy's staff to determine a new acceptable location in the IP2 spent fuel pit for the affected fuel assembly in a controlled, deliberate manner.

2.3 IP3 Technical Specification Changes

The following changes are proposed to the IP3 TS:

1. A new definition for INTACT FUEL ASSEMBLIES is proposed to be added to TS Section 1.1 to provide controls on the physical condition of the fuel assemblies loaded into the STC and transferred to IP2. Damaged or failed fuel is not permitted for loading into the STC for transfer to IP2.

2. A new TS Limiting Condition for Operation (LCO) is proposed, LCO 3.7.18 *Spent Fuel Assembly Transfer*, to provide the limits for fuel assembly loading into the STC in the IP3 spent fuel pit. These limits include a maximum burnup (for shielding), minimum burnup as a function of initial enrichment (for criticality control), and post-irradiation cooling time and decay heat (for thermal). This LCO ensures that the IP3 fuel assemblies selected for loading into the STC and the arrangement of those assemblies in the fuel basket is always bounded by the criticality, thermal, shielding, and accident analyses supporting the design. These limits also ensure that the fuel assemblies being transferred in the STC can be stored in the IP2 spent fuel storage rack locations per IP2 TS 3.7.13 *Spent Fuel Pit Storage* as designated on the fuel move sheets. Associated with this LCO is a new proposed Bases B 3.7.18.

3. A new Surveillance Requirement (SR), SR 3.7.18.1 is proposed to require an administrative verification that each fuel assembly chosen for loading meets the requirements of the LCO for the STC fuel basket cell designated for that assembly.

4. A new TS Design Feature 4.4 *Shielded Transfer Canister* is proposed to provide a limit for the maximum enrichment of any single fuel assembly to be loaded into the STC, the

acceptance criterion for the maximum analyzed reactivity of the STC, a nominal center-to-center fuel assembly spacing dimension in the STC, a limit on the fuel assembly capacity of the STC, and certain requirements for drainage of the STC. The TS is exactly the same as the proposed new TS 4.4 for IP2.

In summary, these TS ensure that the fuel to be loaded into the STC meets the design basis for the STC and has an acceptable rack location in the IP2 spent fuel pit before the STC is loaded with fuel.

3.0 BACKGROUND

3.1 Original Plant Design

Indian Point Units 2 and 3 were originally constructed with cask handling cranes to lift and handle low-capacity, relatively lightweight shipping casks on the order of 40 tons. The cask cranes were to be used to lift and place the empty shipping cask in the spent fuel pit and subsequently lift and move the fuel-loaded shipping cask to a preparation area and then to a truck located in the truck bay for ultimate removal of the fuel to an offsite location.

Without an approved off-site high-level repository, IP2 and IP3 have continued to store spent fuel in their respective spent fuel pits. In the late 1990's it became apparent that another solution to fuel storage would be required before the plants ran out of wet storage space.

It was decided, that dry cask storage (DCS) of spent fuel at an onsite Independent Spent Fuel Storage Installation (ISFSI) was the best option for both units to ensure continued reliable plant operation and electric service to our customers. Thus, the DCS project was initiated. A major project expectation is to maintain full core offload capability for both plants at all times and if it is lost, to limit the duration of the loss of full core offload capability.

3.2 Dry Cask Storage Operations at IP2 and IP3

Entergy selected the Holtec HI-STORM 100 System as the dry storage cask technology for use at the ISFSI under the general license provisions of 10 CFR Part 72. The HI-STORM 100 System, like all contemporaneous DCS technologies, accommodates many more fuel assemblies than the 40-ton shipping cask assumed in original plant design and licensing. The 32-assembly multi-purpose canister (MPC) was chosen as the best, most cost-effective option for use at both units. Entergy understood at that time that the cask handling cranes in Units 2 and 3 were of insufficient capacity to support DCS operations and the crane design and limitation issue would need to be addressed as part of the DCS project for each plant.

Current HI-STORM DCS operations involve the handling of heavy loads inside the IP2 FSB, in the same areas originally designed for the 40-ton shipping cask, to facilitate movement of spent fuel from the spent fuel pit to the ISFSI. The DCS canister and transfer cask assemblage weighs about 100 tons fully loaded and must be moved within the Fuel Storage Building (FSB) as follows:

- From a staging area into the spent fuel pit for fuel loading
- From the spent fuel pit to the west end of the truck bay floor for canister closure operations after fuel loading

- From the west end of the truck bay floor to the east end of the truck bay floor where the transfer cask is stacked atop the storage overpack and the canister inside is transferred to the storage overpack

The transfer cask is removed from atop the overpack at the stack-up location and moved back to the west end of the truck bay floor. The loaded overpack is moved to the ISFSI using a suitably designed tracked vertical cask transporter.

3.3 Crane Solutions

Neither the IP2 nor the IP3 FSB structure could withstand the significant increase in loads that would result from increasing the capacity of the existing overhead bridge-and-trolley cask handling cranes from 40 tons to 110 tons. The under-capacity of the IP2 and IP3 cask handling cranes are addressed in different ways because the two plants are not identical with respect to the immediate site topography adjacent to the FSBs.

At IP2, a floor-mounted gantry crane was designed, fabricated, licensed, and installed to facilitate the handling of the 100-ton maximum lifted DCS load comprised of the transfer cask and the MPC, which is filled with 32 fuel assemblies and spent fuel pit water.

At IP3, a similar crane upgrade was determined by engineering evaluation to not be feasible. The distance between the top of the spent fuel pit wall and the truck bay is approximately 23 feet longer at IP3 than at IP2. This increased distance significantly increases the amplification of seismic loads on the crane. These loads would require significant increases in crane structural member sizes in the FSB, which was only designed for the 40-ton capacity cask crane. Due to the limited space in the IP3 truck bay area, accommodating these significantly larger crane structural members would have been an extraordinary engineering task with uncertain success. Moreover, there are numerous plant equipment interferences that would have taken significant design and construction effort to re-locate. These issues made the feasibility and cost of a crane upgrade at IP3, even if physically possible, untenable.

3.4 The Chosen Solution

The Entergy DCS project team studied the IP3 crane issues with their DCS technology supplier, Holtec International, and determined that the most cost-effective way to move IP3 spent fuel into dry storage was to first move the fuel into the IP2 spent fuel pit and subsequently load the IP3 dry storage casks using the infrastructure constructed for IP2 DCS operations.

The inter-unit fuel transfer solution involves upgrading the existing IP3 cask handling crane to a single-failure-proof design while maintaining the 40-ton capacity. The 40-ton crane capacity limit represented an unalterable design criterion around which other parts of the project were optimized.

The STC was designed such that the weight of the canister, canister lid, fuel, water, and rigging would be no more than 40 tons on the IP3 cask crane hook. The optimized capacity of the STC within these constraints was determined to be twelve IP3 fuel assemblies. The reason the capacity of the STC is so much greater than that for a similar weight shipping cask is that the STC structural design and shielding requirements have been specifically established for an on-site transfer canister as opposed to those required of a 10 CFR 71-licensed shipping cask destined for use on the public roadways. This permitted optimization of the shielding and capacity to minimize the number of shipments needed to move batches of fuel from IP3 to IP2 while keeping the estimated occupational radiation exposure for inter-unit transfer operations as low as reasonably achievable (ALARA).

The bolted-lid STC will be moved from IP3 to IP2 inside the Holtec Part 72-licensed HI-TRAC 100D transfer cask, which provides additional shielding and structural protection of the canister during the portion of the transfer occurring outdoors. The HI-TRAC is already compatible with the vertical cask transporter used for ISFSI operations on site. The STC will remain filled with water during the inter-unit transfer using the VCT, which is expected to be accomplished within an 8 hour shift. The first fuel transfer campaign will move 96 assemblies in eight, 12-assembly moves. In a separate campaign to follow, the 96 IP3 assemblies would be removed from the IP2 spent fuel pit and placed into dry storage using three HI-STORM cask systems under the site's Part 72 general license.

4.0 <u>TECHNICAL ANALYSIS</u>

4.1 <u>Introduction</u>

The Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at the Indian Point Energy Center (Reference 1 and Enclosure 1) provides full details of the system design, design calculations and evaluations, material selection, operation, and maintenance requirements. Provided below is a summary of the key analyses and evaluations that support the determination of No Significant Hazards for both IP2 and IP3 provided in Attachments 4 and 5, respectively.

4.2 <u>Fuel Acceptance Criteria and Engineered Measures for Safety</u>

In support of this amendment request a number of analyses have been performed that demonstrate that the fuel is protected throughout the inter-unit transfer, that occupational doses are ALARA, and that the health and safety of the public remains protected. These analyses include criticality, thermal-hydraulic, structural, shielding design and ALARA, and materials evaluations for normal, off-normal and accident conditions. These analyses assume that the STC is loaded in the IP3 spent fuel pit with spent fuel assemblies that meet the following characteristics:

a) The fuel must be intact as defined in Holtec's Dry Storage Certificate of Compliance (Docket 72-1014) and in the proposed addition to the IP3 TS Definitions section
b) Post-irradiation cooling time ≥ 5 years
c) Average burnup per fuel assembly ≤ 55,000 MWD/MTU
d) Decay heat ≤ 650 Watts (any cell)
e) Decay heat ≤ 1105 Watts (interior cells only)
f) Average burnup per fuel assembly ≥ 40,000 MWD/MTU or have a minimum burnup as a function of initial enrichment as specified in proposed TS 3.7.18 to be placed in any location in the STC. Fuel not meeting these burnup requirements may only be loaded in the peripheral cells of the STC and the inner cells must remain empty
g) Initial average assembly enrichment must be less than 5 wt% U-235
h) The fuel may or may not contain non-fuel hardware

Provided the STC is loaded with fuel with these characteristics the acceptance criteria for each analysis is met for normal, off-normal and accident conditions.

A more detailed discussion of the fuel acceptance criteria and safety and protective measures presented below is provided in Chapter 3 of Enclosure 1.

4.2.1 Safety and Protective Measures

4.2.1.1 Criticality Safety through Physical Design

The fuel acceptance criteria ensure that the reactivity criteria will be met by the Spent Nuclear Fuel (SNF) that will be loaded in the STC. The above statement is based on comparing the design data that directly affects reactivity of the spent fuel storage devices in the IP2 SFP (TS Section 4.3) and the IP3 SFP (TS Section 4.3) and the STC. The following observations provide the basis for concluding that criticality safety is assured.

i. The areal B-10 density in the STC fuel basket is substantially greater than that in IP2 and IP3 Region 2 racks. A greater B-10 loading corresponds to reduced reactivity.

ii. The thickness of the stainless steel walls in the STC fuel basket is considerably greater than that in the fuel racks in either pool. An increased mass of stainless steel reduces reactivity.

4.2.1.2 Criticality Safety through Assured Boron Concentration

The fuel transferred to the STC is surrounded by its native environment in the pool, which is the pool's borated water. After the STC is raised from the pool, the boron concentration in the STC cavity will be the same as the pool and cannot be reduced by dilution or any other means. Thus the STC will, by virtue of it sealed configuration, maintain the boron concentration throughout the transfer process. The assured presence of soluble boron in the STC cavity adds another layer of safety against violation of the postulated reactivity limit.

4.2.1.3 Release Protection by Multiple Barriers

The proposed inter-unit transfer operation incorporates three *independent* barriers against release of radioactivity to the environment, namely:

i. The fuel cladding (only intact fuel is permitted to be transferred)
ii. The pressure tested STC is qualified to withstand a normal internal pressure of 50 psig
iii. The HI-TRAC 100D Transfer Cask is qualified to withstand a normal internal pressure of 30 psig

4.2.1.4 Protection by a Favorable Thermal-Hydraulic Environment

The thermal-hydraulic environment around the spent nuclear fuel in the STC basket is considerably more benign than that in the reactor vessel. The thermal environment to which the fuel is subjected to in the reactor is significantly more aggressive than in an STC. Therefore, the risk of degradation of the fuel cladding during the transfer operation is negligible.

4.2.1.5 Protection by the Selection of Low Dose Emitting Fuel

The specific fuel selected for the inter-unit transfer shall have achieved a sufficient decay time so as to meet the fuel assembly heat load acceptance criteria. As the radiation emitted by the fuel decreases exponentially with the passage of time, the batch selected for transfer will have a correspondingly low dose accretion rate.

4.2.1.6 Protection by Use of Proven Equipment

The HI-TRAC 100D transfer cask which serves as a principal radiation barrier in the inter-unit transfer operations is a proven piece of equipment through multiple uses in the IP2 dry storage campaign. It is part of the HI-STORM 100 System approved by the USNRC and demonstrated by measurement. Therefore, the safety of the inter-unit transfer operation is ensured to be ALARA.

4.2.1.7 Protection by Material Selection

The STC and HI-TRAC 100D are two principal components whose materials of construction must be assured from an adverse performance. The materials used in the manufacture of the STC are of the same genre as used in the fabrication of casks and fuel baskets in Holtec's dry storage program. The suitability of these materials, including surface preservatives, has been endorsed by the USNRC on several active Holtec dockets. Therefore, the risk of an anomalous performance by an STC material is unlikely.

The HI-TRAC 100D Transfer Cask is proven hardware having been used in a virtually identical environment in the IP2 dry storage campaign. Therefore, the risk of a material malfunction during the inter-unit transfer campaigns is unlikely.

4.2.1.8 Reliability through Increased Structural Margins

The systems, structures and components proposed for use in the inter-unit transfer program have been engineered with significantly larger structural margins of safety than required to meet the applicable design criteria.

Specifically:

i. The STC is engineered to maintain the stress levels in its pressure boundary to well below the ASME B&PV Code Section III, subsection ND allowables.

ii. The tensile strength of the flange bolts in the STC is considerably larger than that required to maintain the joint seals.

iii. The HI-TRAC 100D will maintain its stress levels when subject to the Design Pressure values that are considerably lower than the respective Code allowables.

iv. Special lifting devices such as the lift yoke, the lift cleat and other lifting appurtenances are designed to meet the stress limits of ANSI N14.6 and NUREG-0612, Section 5.1.6.(1)(a) with ample margins. These special lifting devices will be tested and inspected in accordance with ANSI N14.6 prior to use. Other lifting interfaces such as the trunnions, the STC and HI-TRAC lid lifting points are designed per guidance from NUREG-0612, Section 5.1.6 (3).

v. The IP2 cask and IP3 canister handling cranes will be single-failure-proof and comply with NUREG-0554 and NOG-1-2004. The vertical cask transporter also has a redundant drop protection feature.

The above attributes of the STC, HI-TRAC 100D and the lifting equipment ensure that the inter-unit transfer operation shall not suffer from a structural malfunction or failure.

4.2.1.9 <u>Protection by Design to Prevent Inadvertent Water Loss</u>

The STC is welded cylindrical layered (steel-lead-steel) canister with a welded steel base plate. The top lid is bolted and sealed using an elastomeric seal which will be pressure tested to ensure no leakage of water can occur. There are no penetrations in the STC which allow a release of water after it is sealed. During operations, prior to the lid being sealed, water levels will be monitored and maintained at the required levels. There is no malfunction or accident which would cause a loss of water from the STC cavity. Similarly, water loss from the HI-TRAC-100D is prevented by use of top and bottom bolted covers that are sealed using elastomeric seals which will also be pressure tested to ensure no leakage can occur.

4.2.1.10 <u>Protection by Procedural Controls</u>

The following robust administrative controls will be in effect to ensure that a fuel assembly misloading accident will not occur:

i. Prior to development of fuel move sheets for loading the STC, the fuel will be characterized to ensure compliance with proposed IP3 Technical Specification 3.7.18 and existing IP2 Technical Specification 3.7.13. This characterization will be performed in accordance with approved Reactor Engineering procedures.

ii. Fuel move sheets will be developed using the results of the fuel characterization process. The moves sheets will be independently checked by a qualified Reactor Engineer as required by Reactor Engineering procedures.

   It should be noted that the fuel assembly ID number, insert ID number, and location of the proposed fuel assembly candidates are verified during visual fuel inspections performed as part of the fuel selection process for dry cask storage and this practice will be continued for STC loading and unloading. This verification is performed independently by fuel handing and Reactor Engineering personnel.

iii. Prior to removal of a fuel assembly from its SFP storage rack location, fuel handling personnel will verify and peer check that the fuel assembly physical ID number correctly corresponds to the fuel assembly ID number specified on the approved move sheets.

iv. Peer checking will be performed by fuel handling personnel to ensure that the fuel assembly is subsequently placed into the proper STC cell location as designated on the approved fuel move sheets.

v. A video recording of the fully loaded STC will be performed and peer checked to ensure that the STC has been properly loaded in accordance with the approved load plan.

vi. Prior to transfer of the loaded STC to IP2, an independent check of the video recording will be performed by a qualified Reactor Engineer to demonstrate that the STC has been properly loaded in accordance with the approved load plan.

vii. To prevent a misloading event in the center STC cells, a cell blocker will be used.

viii. To detect a fuel assembly misloading event (i.e. inadvertent substitution of a fuel assembly with a significantly higher decay heat load than expected), water temperature readings will be taken at the top of each fuel assembly using a remote indicating temperature measuring device prior to STC lid closure.

A more detailed discussion of procedural controls is provided in Chapter 10 of Enclosure 1.

4.3 Applicable Design, Normal, and Postulated Accidents

4.3.1 Design Basis and Normal Load Conditions

The STC and HI-TRAC 100D are required to meet the applicable stress limits under the provisions of ASME Section III Subsection ND, Class 3. The analysis of the normal and various accident condition loads must show that the internal pressure and temperature remain below their respective design limits. The operating pressure and temperature under normal operations are bounded by the design pressure and temperature and, therefore, do not require a separate analysis.

4.3.2 Accident Conditions

Accident conditions for inter-unit operations belong to two categories, namely:

i. Credible accidents inside Part 50 structures
ii. Credible and non-credible accidents during the transfer of loaded HI-TRAC 100D from the IP3 FSB to the IP2 FSB

4.3.2.1 Accident and Natural Phenomena Events Considered during STC Loading and Unloading Inside Part 50 Structures

The accidents postulated inside Part 50 structures are summarized in Table 4.1 together with the pertinent analysis considerations and analysis or evaluation objective.

4.3.2.2 Accident and Natural Phenomena Events Considered during Inter-Unit Transfer Outside Part 50 Structures

The accidents postulated outside Part 50 structures are summarized in Table 4.2 together with the pertinent analysis consideration and analysis or evaluation objective.

Table 4.1

Accident Scenarios Inside Part 50 Structures

Accident	Event	Consideration	Objective
Accidental drop of a fuel assembly	A dropped fuel assembly plus handling tool impacts the top of the STC basket	Structural	Demonstrate no impairment of safety function
		Dose	Demonstrate that any radioactive release is bounded by the existing FHA
Misloading of a fuel assembly in the STC basket	Misloading of a fresh fuel assembly into the STC	Criticality	Demonstrate that the acceptance criteria of 10 CFR 50.68 continue to be met
	Misloading of a high decay heat assembly into the STC	Thermal	Demonstrate that protective measures will prevent the inter-unit transfer of a high decay heat fuel assembly
Earthquake	Unloaded and loaded STC and HI-TRAC subjected to Design Basis Earthquake (DBE)	Structural	Demonstrate that the unloaded and loaded STC and HI-TRAC will remain stable under DBE conditions
Boron dilution in the spent fuel pits	Inadvertent boron dilution	Criticality	Demonstrate that this amendment has no impact on event
Uncontrolled increase in spent fuel pit temperature	Loss of spent fuel pit cooling	Thermal	Demonstrate that this amendment has no impact on event

Table 4.2

Accident Scenarios Outside Part 50 Structures

Accident	Event	Consideration	Objective
Accidental drop of loaded HI-TRAC 100D	A HI-TRAC drops the maximum lift height onto the ground	Structural	Demonstrate no impairment of safety function
		Dose	Demonstrate drop will not result in any unacceptable consequence for the transfer cask and its contents

Fire	A fire engulfs the HI-TRAC due to the spillage and ignition of 50 gallons of combustible fuel	Thermal	Demonstrate that the fuel cladding temperature is below SFST-ISF-11 limits, materials within design temperatures and STC within design pressure
Lightning	Lightning strike on the HI-TRAC	Various	Demonstrate that lightning will not impair the safety function of the cask
Earthquake	Loaded VCT and loaded HI-TRAC standing alone subjected to Design Basis Earthquake (DBE)	Structural	Demonstrate that the loaded VCT and loaded HI-TRAC will remain stable under DBE conditions
Flood	Loaded VCT becomes flooded	Various	Demonstrate that the loaded VCT will not be externally flooded
Environmental Loadings	HI-TRAC subjected to high winds, tornado, and tornado-borne missiles	Structural	Demonstrate that environmental loadings will not impair the safety function of the cask
Loss of Water in the Water Jacket	HI-TRAC subjected to loss of water jacket water	Thermal	Demonstrate that temperature and pressure limits will not be exceeded
		Dose	Demonstrate compliance with annual site boundary dose limit
Extended time of STC residence in the HI-TRAC	HI-TRAC subjected to prolonged VCT breakdown	Thermal	Demonstrate that temperature and pressure limits are not exceeded
		Dose	Demonstrate that regulatory limits are met
Collapse of the roadway during transfer resulting in a cask rollover	Loaded VCT subjected to roadway collapse	Structural	Demonstrate that the roadway will not collapse and it is therefore a non credible event. Demonstrate significant margin to cask rollover.
Large radioactive release from the cask	STC subjected to breach resulting in instantaneous release of all 12 assemblies gases into the STC cavity and subsequently into the environment	Dose	Demonstrate that any radioactive release is bounded by the existing FHA for this non credible event

4.4 Criticality Evaluation

4.4.1 Introduction

The purpose of the criticality evaluation is to provide the necessary assurance that there will be no unacceptable risk due to criticality events. To that end, the criticality analyses have been performed under the following normal and accident scenarios:

i. Evaluation of two normal loading configurations in accordance with the proposed TS:

(a) Configuration 1 is analyzed to accommodate fuel with a specified minimum burnup as a function of the initial enrichment (see Table 4.3) in every basket location. A fuel assembly with a burnup less than the minimum burnup specified in Table 4.3 is classified as Type 1 fuel. Similarly, a fuel assembly with burnup greater than the minimum is classified as Type 2 fuel. These limits are reflected in the proposed new IP3 TS LCO 3.7.18 *Spent Fuel Assembly Transfer*.

(b) Configuration 2 is analyzed to accommodate fresh fuel in the peripheral eight fuel basket locations. There is no intent to load fresh fuel into the STC and the proposed new Technical Specification 3.7.18 *Spent Fuel Assembly Transfer* expressly precludes this event. The criticality analysis assumes fresh fuel in order to bound the reactivity of any fuel assembly that could be loaded into the STC. The interior four locations remain empty.

ii. Evaluation of the following off-normal and accident scenarios:

(a) Accidental misloading of a fresh fuel assembly with maximum enrichment into the STC under Configuration 1, as well as the misloading of any fuel assembly into one of the four center STC cells in Configuration 2

(b) Off-normal spent fuel pit temperature

(c) Fuel assembly dropped onto the top of the STC

(d) Dropped HI-TRAC

(e) Boron dilution in the spent fuel pit

Additionally, the acceptability of storing fuel from IP3 in the pit of IP2 has been evaluated.

A more detailed discussion of the criticality evaluation is provided in Chapter 4 of Enclosure 1.

4.4.2 Methodology

The principal method for the criticality analysis of the high-density fuel basket is the use of the three-dimensional Monte Carlo code MCNP4a. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been used previously and verified for criticality analyses and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data predominantly based on ENDF/B-V and ENDF/B-VI. Additionally, CASMO-4, a two-dimensional transport theory lattice physics code, is used to determine reactivity differences for moderator temperature variation, manufacturing tolerances, depletion uncertainty and to calculate the isotopic inventory of the spent fuel for use in MCNP4a. Both codes are benchmarked by comparison with critical experiments.

4.4.3 Acceptance Criteria

The objective of this evaluation is to show that the effective neutron multiplication factor, k_{eff}, is less than 0.95 with the fuel basket loaded with fuel of the highest anticipated reactivity and the STC flooded with unborated (normal conditions) or borated (accident conditions) water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including manufacturing tolerances and is shown to be less than 0.95 with a 95% probability at a 95% confidence level. Reactivity effects of abnormal and accident conditions have also been evaluated to assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95 under borated conditions. These acceptance criteria are in accordance with 10 CFR 50.68(b).

4.4.4 Design Attributes

The areal B-10 density in the STC fuel basket is substantially greater than that in IP2 and IP3 region 2 racks. A greater B-10 loading corresponds to reduced reactivity. The thickness of the stainless steel walls in the STC fuel basket is considerably greater than that in the fuel racks in either pool. An increased mass of stainless steel reduces reactivity.

To prevent a misloaded assembly in Configuration 2, a cell blocker device will be used to block the openings of the four center cells during the loading of the STC. To prevent a misloaded assembly in Configuration 1, this same device will be used when loading the first eight assemblies and additional visual inspections can be implemented prior to and after loading each of the remaining four assemblies.

4.4.5 Key Assumptions and Inputs

Full three-dimensional calculational models were used in MCNP, explicitly modeling fuel rods and cladding, guide tubes, basket walls, and neutron absorber panels on the basket walls covered by sheathing. The STC around the basket is also modeled, surrounded by a water reflector. The HI-TRAC is not included in the models. Studies were performed showing that this is an appropriate approach. CASMO uses infinite arrays of basket cells in a two-dimensional geometry. Additionally, the analyses use a range of assumptions, in order to simplify the calculations and/or to provide additional conservatism. In summary, those assumptions assure that the true reactivity will always be less than the calculated reactivity. The following is a list of the major assumptions that were employed:

i. A conservative assumption of zero cooling time is used along with setting the xenon concentration to zero for all CASMO-4 calculations in the fuel basket models. No credit is therefore taken for the significant actual cooling time of the fuel assemblies to be loaded (≥ 5 years).
ii. Conservative operating parameters during fuel depletion are assumed.
iii. All assemblies are assumed to contain burnable absorbers during depletion.
iv. Reactivity control devices (RCCAs, WABAs, BPRAs, etc) that may be present in the fuel are not credited in the STC calculations.

v. A burnup record uncertainty of 5% is assumed and incorporated in the analyses.
vi. Calculations use nominal fuel and fuel assembly dimensions. Tolerances are treated as uncertainties.
vii. Conservative axial burnup profiles are used.
viii. Centered and eccentric positioning of fuel assemblies in the basket cells are considered.
ix. Calculations are performed at a water temperature and density within the operating range that maximizes k_{eff}.
x. Only intact fuel assemblies are permitted for loading in the STC. 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 are not considered intact 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). This definition of intact fuel assemblies is proposed to be added to the Definitions section of the IP3 Technical Specifications. The fuel classification and selection procedure(s) for STC loading will include this definition to ensure only intact assemblies meeting this definition and the IP3 LCO for STC loading are selected for transfer to IP2.

4.4.6 Results and Conclusions

4.4.6.1 Normal Loading Configurations

The results of the evaluation of the two normal loading configurations are summarized in Table 4.4 and show that the maximum k_{eff} value is below the 10 CFR 50.68 limit of 0.95.

4.4.6.2 Accidental Misloading of a Fresh Fuel Assembly with Maximum Enrichment into the STC

The results of the evaluation of the accident condition of a misloaded fresh fuel assembly of maximum enrichment are also summarized in Table 4.4. The results show that the maximum k_{eff} value is below the 10 CFR 50.68 limit of 0.95.

4.4.6.3 Off-Normal Spent Fuel Pit Temperature

All calculations for the fuel basket are performed at a water temperature of 39.2 °F (4 °C). The temperature coefficient of reactivity is negative; therefore no additional calculations are required, because a further increase in temperature reduces the reactivity.

4.4.6.4 Fuel Assembly Dropped onto the Top of the STC

For the case in which a fuel assembly is assumed to be dropped on top of the STC, the fuel assembly will come to rest horizontally on top of the STC, with a minimum separation distance from the active fuel region of more than 12 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity. It is also possible to vertically drop an assembly into an empty location or a location occupied by another assembly. Such a vertical impact would at most cause a small compression of the assembly, reducing the water-to-fuel ratio and thereby reducing reactivity. Furthermore, the reactivity effect of a dropped assembly would always be bounded by the misloading condition discussed above, and the soluble boron maintained in the spent fuel pit water in accordance with the plant technical specifications assures that the true reactivity is always less than the limiting value for such dropped fuel accident. The structural analyses have also considered any potential damage to the basket from the fuel assembly drop accident, and concluded that the reactivity control features of the basket remain unaffected.

4.4.6.5 Dropped HI-TRAC

For the case in which a HI-TRAC is dropped a maximum of 6" a structural analysis has demonstrated that the fuel basket geometry continues to preserve the assumptions and inputs used in the STC criticality analysis.

4.4.6.6 Boron Dilution

The STC criticality analysis for normal conditions was performed assuming unborated water in the STC. Thus, even an unexpected complete de-boration of either spent fuel pit will not result in an unanalyzed condition for the STC.

The minimum required boron concentration in the IP2 spent fuel pit for normal spent fuel storage in the wet storage racks is 2,000 ppm per IP2 TS LCO 3.7.12. The minimum required boron concentration in the IP3 spent fuel pit for normal spent fuel storage in the wet storage racks is 1,000 ppm per IP3 LCO 3.7.15. The STC is being loaded in the IP3 pit. If the IP3 spent fuel pit boron concentration was at its TS minimum of 1,000 ppm during STC loading and the STC was transferred to the IP2 spent fuel pit with its boron concentration at its minimum of 2,000 ppm, the boron concentration of the IP2 spent fuel pit would fall slightly below the minimum required, to about 1,990 ppm.

However, the IP3 spent fuel pit boron concentration is administratively controlled by procedure at 2,000 ppm, well above the 1,000 ppm minimum. This existing administrative control will be used to ensure the IP3 spent fuel pit boron concentration is at least 2,000 ppm during STC loading to avoid any dilution of the IP2 spent fuel pit when the STC is unloaded.

4.4.6.7 Acceptability of Storing IP3 Fuel in the IP2 Spent Fuel Pit

From a criticality perspective, the fuel from Indian Point Unit 2 (IP2) and Indian Point Unit 3 (IP3) are essentially the same. The most significant difference between the fuel types in the two plants is that IP2 fuel has more grid straps than the similar model for IP3, which is a detail too fine to affect the criticality model. Every type of fuel used at IP3 (LOPAR, OFA, Vantage5, Vantage+, 15x15 Upgrade) has also been used at IP2 and is resident in the IP2 Spent Fuel Pit (SFP). Otherwise, the pertinent physical characteristics of the fuel types, such as general physical geometry, uranium enrichment and loading are identical. Also, the use of burnable poisons, operating temperature and reactor power is comparably the same. Note that thermal power has increased gradually for both plants since initial operation.

Table 4.3

Minimum Fuel Assembly Burnup versus Enrichment for STC Loading

Nominal Initial Enrichment (wt% U-235)	Minimum Burnup (GWD/MTU)
1.8	0
2	5.50
2.5	12.75
3	18.20
3.5	23.90
4	29.75
4.5	35.00
4.95	40.00

Table 4.4

Criticality Results

	Configuration 1 (Fuel Type 2)		Configuration 2 (Fuel Type 1 or 2)	
	Normal	Accident*	Normal	Accident*
Soluble Boron (ppm)	0	600	0	600
k-calc	0.9118	0.9038	0.9127	0.9189
Uncertainties (delta-k)	0.0172	0.0154	0.0136	0.0139
Biases (delta-k)	0.0089	0.0074	0.0060	0.0045
Maximum k_{eff}	0.9379	0.9266	0.9323	0.9373

* A misloaded fresh fuel assembly with an initial enrichment of 4.95 wt% is assumed

4.5 Thermal Evaluation

4.5.1 Introduction

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk due to fuel cladding temperature, and component and material temperatures and pressures being outside their design limits.

The thermal-hydraulic adequacy of the STC has been evaluated for the following scenarios:

i. Normal onsite transfer of IP3 fuel
ii. Long term VCT breakdown
iii. A postulated accident event resulting in the rupture of the HI-TRAC water jacket
iv. A postulated 50-gallon transporter fuel tank rupture and fire accident
v. Lightning strike

Though not specifically evaluated in Enclosure 1 the following have also been considered:

vi. Misloading of a high decay heat fuel assembly
vii. Loss of spent fuel pool cooling

A more detailed discussion of the thermal evaluation is provided in Chapter 5 of Enclosure 1.

4.5.2 Methodology

The STC interior is a 3-D array of square shaped cells inside an irregularly shaped basket outline confined inside the cylindrical space of the STC cavity. To ensure an adequate representation of these features, a 3-D geometric model of the STC is constructed using the FLUENT Computational Fluid Dynamics (CFD) code pre-processor. Other than representing the composite cell walls (made up of stainless steel panels, neutron absorber panels and stainless steel sheathing) by a homogeneous panel with equivalent orthotropic (thru-thickness and parallel plates direction) thermal conductivities, the 3-D model requires no idealizations of the fuel basket structure. Further, since as it is impractical to model every fuel rod in every stored fuel assembly explicitly, the cross section bounded by the inside of the storage cell, which surrounds the assemblage of fuel rods and the interstitial water, is replaced with an "equivalent" square homogeneous section characterized by an effective thermal conductivity. For thermal-hydraulic simulation, each fuel assembly in its storage cell is represented by an equivalent porous medium.

4.5.3 Acceptance Criteria

The following acceptance criteria apply during normal onsite transfer and accident scenarios:

i. The fuel cladding temperatures must remain below the SFST-ISG-11 temperature limit of 752°F and 1058°F during normal operation and accident scenarios, respectively.

ii. The maximum temperature of the basket structural materials, neutron absorbers, STC pressure boundary materials must remain within their design limits.

iii. The maximum STC, HI-TRAC annulus, and water jacket pressures must be within their design limits.

4.5.4 Design Attributes

The STC is a thick-walled vessel containing a fuel basket. The fuel basket design is similar to the Holtec Multi-Purpose Canisters (MPCs) deployed for storing fuel in the HI-STORM System but the STC wall is substantially thicker than the MPC wall to provide enhanced radiation protection. Up to twelve fuel assemblies can be accommodated in the STC fuel basket. For additional shielding the STC is loaded in a HI-TRAC steel-lead-steel transfer cask having a thick bolted lid prior to on-site movement. To minimize fuel and cask temperatures the STC cavity, the HI-TRAC annular space between the outer STC wall and the inner HI-TRAC wall, and the HI-TRAC water jacket will be filled with water. The thermal analyses consider only passive rejection of decay heat from the spent nuclear fuel (SNF) to the environment.

During fuel transfer operations the water inside the STC and HI-TRAC expands under heatup to normal operating temperatures. To protect the vessels from excessive hydraulic pressures air spaces are provided under the STC and HI-TRAC lids. The STC and the HI-TRAC are designed to withstand a maximum internal pressure considering maximum accident temperatures and therefore do not require a pressure relief valve.

The potential of a fire accident near the VCT during its movement is considered to be extremely remote because there are no significant combustible materials in the area except the diesel fuel in the VCT fuel tank. An evaluation of the haul path will be performed prior to each inter-unit transfer to ensure the VCT fuel tank fire accident considered remains bounding.

The HI-TRAC 100D transfer cask fire accident is conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible fuel which engulfs the HI-TRAC. The HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire.

4.5.5 Key Assumptions and Inputs

Thermal analysis of the STC is performed under bounding heat load scenarios wherein all fuel assemblies are assumed to be generating heat at the maximum permissible rate. While the assumption of limiting heat generation in each storage cell imputes a certain symmetry to the cask thermal problem, it grossly overstates the total heat duty because it is unlikely that any fuel basket would be loaded with all fuel assemblies emitting heat at their limiting values.

The key assumptions and attributes of the thermal model are:

i. Heat generation in the STC is axially non-uniform with peaking in the mid-section of the active fuel length.

ii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the STC is spatially distributed with the maximum values reached in the central core region.

iii. Heat is dissipated in the fuel basket by internal convection of water. As the rate of heat transfer is a direct function of flow resistance, the thermal analysis is conservatively based on the assumption that all fuel storage locations are populated with the most resistive Westinghouse fuel, W-17x17 fuel assemblies.

iv. Heat is dissipated from the external surfaces of the cask by radiation and natural convection to air.

v. The maximum allowable fuel assembly decay heat is 1105 Watts for the four interior cells and 650 Watts for the eight peripheral cells of the STC. These values correspond to the decay heats calculated for the bounding burnup and cooling time combinations of the fuel permitted by new IP3 LCO 3.7.18 *Spent Fuel Assembly Transfer* to be loaded in the corresponding cells of the STC.

vi. The maximum ambient temperature is 100°F.

vii. The temperature of the fire is assumed to be 1475°F to accord with the provisions of 10CFR71.73 since no guidance is supplied in Part 50.

4.5.6 Results and Conclusions

4.5.6.1 Normal Onsite Transfer of IP3 Fuel

The 3-D model referred to above was used to determine vessel internal pressures and temperature distributions during transfer of IP3 fuel loaded in the STC. The fuel transfer scenario assumes maximum permissible fuel heat load, a maximum ambient temperature, insolation heating and steady state maximum temperatures. The results of the analysis show that component temperatures and vessel pressures are in compliance with the thermal acceptance criteria. For example, the maximum fuel cladding temperature is 221°F, and the maximum STC and HI-TRAC cavity pressures are 34.0 and 17.5 psig, respectively.

4.5.6.2 Long-Term Breakdown of VCT

This accident condition postulates that, for whatever reason, the STC is kept in the transfer cask for an extended period. Theoretically, this condition will result in a gradual heat up of the cask. The thermal hydraulic analyses are carried out assuming that the duration of the STC-in-HI-TRAC condition is infinite so that a steady state condition has been reached. Thus the "VCT breakdown" scenario is subsumed in the normal condition thermal analysis.

Normal onsite transfer is analyzed as a steady state condition and is therefore also representative of an extended time of STC residence in the HI-TRAC that could result from a long term VCT breakdown.

4.5.6.3 Postulated Accident Event Resulting in the Rupture of the HI-TRAC Water Jacket

The integrity of fuel cladding and STC pressure boundary integrity were evaluated under a postulated rupture and loss of water from the HI-TRAC water jacket. The results of jacket water loss evaluation confirm that the cladding, STC and HI-TRAC component temperatures are below design limits and the coincident STC pressure is bounded by the vessel design pressure. For example, the maximum fuel cladding temperature is 241°F, and the maximum STC and HI-TRAC cavity pressures are 53.5 and 25.5 psig, respectively.

4.5.6.4 Postulated 50-gallon Transporter Fuel Tank Rupture and Fire Accident

The HI-TRAC 100D transfer cask fire accident is conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible fuel which engulfs the HI-TRAC. The HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Fuel cladding temperature remains less than the acceptance criterion. For example, the maximum fuel cladding temperature is 239°F, and the maximum STC and HI-TRAC cavity pressures are 53.5 and 25.5 psig, respectively.

4.5.6.5 Lightning Strike

The effect of a lightning strike on the transfer cask is considered in the Entergy HI-STORM 100 Cask System 72.212 Evaluation Report, IPEC Site Specific Appendix F where it is determined that lightning will not impair the safety function of the cask. Lightning may however cause an ignition of the transporter fuel. This scenario was considered above. Therefore, a lightning strike is not considered further.

4.5.6.6 Misloading of a High Decay Heat Fuel Assembly

The misloading of a high decay heat fuel assembly will cause an immediate temperature rise in the STC basket cell. Thermocouples will be placed and monitored such that if a high decay heat fuel assembly is placed into the STC it will be detected and removed prior to the fuel transfer. Misloading of a high decay heat fuel assembly is not an analyzed event as the above actions in addition to the procedural controls that will be in place to verify each fuel move will effectively prevent the transfer of such fuel.

4.5.6.7 Loss of Spent Fuel Pool Cooling

The loss of spent fuel pit cooling remains the same because the thermal design basis for the spent fuel pit cooling loop provides for all fuel pit rack locations to be filled at the end of a full core discharge and therefore the design basis heat load effectively includes any heat load associated with the assemblies within the STC.

4.6 Structural Evaluation

4.6.1 Introduction

The purpose of the structural analyses is to provide the necessary assurance that there will be no unacceptable risk of loss to the fuel configuration assumed by criticality analysis, no unacceptable release of radioactive material, no unacceptable radiation levels, or impairment of ready retrievability of fuel from the STC and the STC from the HI-TRAC transfer cask.

The following load cases have been evaluated:

i. Design pressure for the STC and the HI-TRAC
ii. Normal operating pressure plus temperature
iii. Normal handling
iv. Fuel assembly drop accident
v. HI-TRAC vertical drop accident
vi. Seismic stability of the loaded VCT

vii. Seismic stability of the loaded HI-TRAC
viii. Seismic stability of the STC in the fuel pits

Though not a specific load case evaluated in Enclosure 1 environmental loadings have also been considered.

Drawings of the STC assembly, STC fuel basket, HI-TRAC inter-unit transfer lid, and the HI-TRAC 100D transfer cask assembly are provided in Enclosure 1.

A more detailed discussion of the structural evaluation is provided in Chapter 6 of Enclosure 1.

4.6.2 Methodology

The structural analyses of the STC and the HI-TRAC 100D transfer cask are either based on classical strength of materials solutions or based on finite element numerical analysis. Specifically, the STC containment boundary (baseplate, the shell and the closure lid) and HI-TRAC 100D inner boundary (shell, top lid and bottom lid) under normal operating conditions are evaluated using classical strength-of-materials calculations. The STC trunnions, the lid lifting points and the HI-TRAC lid lifting points also follow the classic strength of materials approach. The STC closure lid is analyzed for normal handling using the finite element code ANSYS 11.0. The fuel assembly drop accident into the STC is analyzed using the principle of conservation of energy. The stability of the loaded VCT, the loaded HI-TRAC, and the loaded STC are analyzed using static force equilibrium. The HI-TRAC vertical drop accident is analyzed using the dynamic analysis code LS-DYNA 970.

4.6.3 Acceptance Criteria

The design of the STC/HI-TRAC is required to meet the following structural objectives, as applicable, under all design basis load combinations for the eight loads cases identified above:

i. Design pressure for the STC and the HI-TRAC:
Demonstrate that the HI-TRAC and STC meet ASME III, Subsection "ND" stress limits

ii. Normal operating pressure plus temperature:
Demonstrate that the HI-TRAC and STC meets ASME III, Subsection "ND" stress limits

iii. Normal handling:
Demonstrate that all lifting operations applicable to the transfer of fuel using the STC and the HI-TRAC are in compliance with applicable codes and standards

iv. Fuel assembly drop accident:
Demonstrate that the fuel storage array remains subcritical

v. HI-TRAC vertical drop accident:
Demonstrate that the peak deceleration is less than 45g

vi. Seismic stability of the loaded VCT:

Demonstrate that the loaded VCT will remain stable under design basis earthquake (DBE) conditions

vii. Seismic stability of the loaded HI-TRAC:
Demonstrate that the loaded HI-TRAC will remain stable under DBE conditions

viii. Seismic Stability of the STC in the fuel pits:
Demonstrate that the unloaded and loaded STC will remain stable under DBE conditions

4.6.4 Design Attributes

Codes and Standards

The HI-TRAC 100D is designed to meet the stress limits of the ASME Code, Section III, Subsection NF (1995 Edition), Class 3 as described in Docket No. 72-1014. It is also noted that, unlike its use in dry storage operations, the HI-TRAC 100D with the new solid bolted top lid, will be subject to an internal pressure when it is used to transport the STC. The HI-STORM FSAR does not consider any internal pressure loads on the HI-TRAC 100D since the lid used for dry storage contains a large circular hole in the middle. Therefore, the newly designed top lid, as well as the HI-TRAC inner shell and bottom lid, have been evaluated for the effects of internal pressure and compared against the stress limits of ASME III, Subsection ND. However, the HI-TRAC is not an ASME Code-stamped pressure vessel. The HI-TRAC has two lifting upper trunnions compatible with the VCT used to lift the cask and move it to IP2 in the vertical orientation.

The STC is a pressure vessel designed to meet the Level A stress limits of the ASME Code, Section III, Subsection ND (2004 Edition). The STC is also not an ASME Code-stamped vessel.

The components of the single-failure-proof lifting system used to handle the STC and HI-TRAC will meet the guidance in Section 5.1.6 of NUREG-0612 and in ANSI N14.6, as applicable. The IP3 crane will be upgraded to be single-failure-proof meeting the intent of NUREG-0554 through the use of ASME NOG-1-2004 as the governing design code. Since both the IP3 and IP2 cask lifting systems will be single-failure-proof, a drop accident involving the STC inside the IP2 or IP3 FSB is not considered credible. The most severe load on the fuel basket inside the STC is a fuel handling accident resulting in a free drop of a fuel assembly onto the top of the fuel basket.

Entergy commits to upgrading the IP3 crane as described prior to the first inter-unit transfer (see Commitment #2 in Attachment 6).

HI-TRAC Drop and Tip Over

The maximum height which the loaded HI-TRAC 100D can be lifted on the VCT is limited to 6". The lift height of the loaded HI-TRAC will be controlled as it is raised on the VCT to ensure this limit is not exceeded. Once the locking pins are engaged on the VCT, redundant drop protection is provided and the lift height/carry is no longer limited. During transfer between the units, the STC/HI-TRAC transfer cask assemblage will be carried by the VCT. The VCT's load-bearing members are designed in accordance with ANSI N14.6.

The HI-TRAC containing the fuel-loaded STC is moved into the IP2 FSB using a LPT on Hilman Rollers, as is used for dry storage cask movements in the same area. This supports the cask from below and precludes a vertical drop. The LPT operation with a loaded HI-STORM has been evaluated to ensure that the HI-STORM would not tip over while on the LPT. This evaluation bounds the HI-TRAC containing the STC since the loaded HI-TRAC with the STC is lighter and has a lower center of gravity than the loaded HI-STORM.

Roadway Collapse

The collapse of the roadway during transfer resulting in a cask rollover is considered to be non-credible based on the following discussion.

First, the collapse of the roadway is considered to be a non-credible accident condition for the following reasons:

(1) The VCT will travel an approved route starting at the IP3 cask loading area and traveling to the IP2 cask loading area that will be evaluated (ground penetrating radar and/or soil compaction studies) prior to the transfer of the spent fuel. Since the site is situated on bedrock there is already an excellent structural base.

(2) The roadways at the site are qualified for American Association of State Highway and Transport Officials H20 loads (typical tractor trailer loads) and the VCT loads on the roadway are similar, however the roadway will be upgraded to support the VCT with the loaded STC in the HI-TRAC. This upgrade includes placing concrete runways along the haul path for the VCT to ride on.

(3) A portion of the haul path has already been analyzed and used multiple times as part of the IP2 and IP1 dry storage campaigns. That analysis is considered bounding since the VCT carrying a loaded MPC inside the HI-STORM weighs more than the VCT carrying a loaded STC inside the HI-TRAC.

(4) Prior to each transfer, the roadway will also be visually inspected and repaired as necessary.

Second, in the unlikely event that the roadway were to collapse, the VCT can withstand an eight foot depression of the roadway, in the most limiting configuration (one track of the VCT being eight feet above the other), without tipping over. This orientation of the VCT is considered bounding since the tracks of the VCT are longer than the width of the VCT. Even in this most extreme scenario the center of gravity of the VCT carrying the HI-TRAC remains low enough so that a tip-over can not occur.

The transport roadway and any safety-related components beneath it that could be affected by the inter-unit transfer of fuel will be shown to have acceptable structural capacity including the consideration of underground utilities. Entergy commits to performing this evaluation and implementing any modifications determined to be necessary prior to the first inter-unit transfer of fuel (see Commitment #3 in Attachment 6).

Floodwater

Potential sources for the flood water could be unusually high water from a river or stream, a dam break, or a hurricane. The IP2 and IP3 FSB truck bay and proposed haul path would require a rise in the Hudson River of over 55 feet to cause flooding; therefore the effect of the flood on the VCT is not considered credible and is not specifically analyzed. Administrative controls will be in place to prevent fuel transfer if severe weather is expected. If unexpected severe weather occurs during fuel transfer, the STC will be moved to the most appropriate safe location given its location at the time of the event.

4.6.5 Key Assumptions and Inputs

Design Pressure, Normal Operating, and Normal Handling Load Cases

General assumptions regarding the structural evaluation are provided in Chapter 6 of Enclosure 1.

Fuel Assembly Drop Accident, HI-TRAC Vertical Drop Accident and Seismic Stability Evaluations

The assumptions regarding these structural evaluations are proprietary and are provided in Chapter 6 of Enclosure 1.

4.6.6 Results and Conclusions

4.6.6.1 Design Pressure for the STC and the HI-TRAC, Normal Operating Pressure plus Temperature and Normal Handling

The design of the STC and HI-TRAC are in compliance with the applicable codes and standards and loads have been shown to be within design limits, including during all lifting operations applicable to the transfer of fuel.

4.6.6.2 Fuel Assembly Drop Accident

A drop of a fuel assembly onto the STC has been analyzed and it has been shown that the criticality control features of the basket remain unaffected.

4.6.6.3 HI-TRAC Vertical Drop Accident

A drop of the HI-TRAC containing a fuel-loaded STC while suspended from the VCT was analyzed for the short period of time outdoors when the VCT does not have its redundant drop protection design features installed. The physical design of the STC and HI-TRAC 100D effectively precludes significant effluent releases as determined by the analysis of the HI-TRAC 100D vertical drop accident that demonstrates that the drop event would not result in any unacceptable consequence for the transfer cask and its contents.

4.6.6.4 Seismic Stability of the Loaded VCT and Loaded HI-TRAC

A seismic analysis has been performed and shows that the loaded VCT and the loaded HI-TRAC will remain stable under design basis earthquake conditions. The HI-TRAC containing the fuel-loaded STC is moved out of the IP3 Fuel Storage Building (FSB) truck bay using air pads. The air pad operation has been evaluated to ensure the HI-TRAC will not tip over. If an earthquake occurs while the air pad is energized, the HI-TRAC will remain stationary on the air pad as the air pad moves along the truck bay floor. The low coefficient of friction between the air pad and the floor precludes the HI-TRAC from tipping over. It has been demonstrated that if the air pad is de-energized or otherwise restrained, the HI-TRAC will not tip over.

The seismic stability of the HI-TRAC with loaded MPC-32 has previously been demonstrated as described in the FSAR for the HI-STORM 100 Cask System. That analyses bounds the loaded HI-TRAC due to the lower weight and height of the loaded STC.

4.6.6.5 Seismic Stability of the Unloaded and Loaded STC in the Spent Fuel Pits

The stability of the loaded STC in either spent fuel pit under the site's Design Basis Earthquake (DBE) has been demonstrated by analysis. A static analysis has been performed to show that the DBE does not cause incipient tipping of the STC. The analysis conservatively neglects the fluid resistance from the spent fuel pit water. The STC does not tip over and does not have unacceptable structural interactions with the spent fuel pit liners or spent fuel racks in either pit. This conclusion also applies to the unloaded STC (without lid) plus fuel basket since the center of gravity height of the empty STC is less than the input value used above.

4.6.6.6 Environmental Loadings

The loadings from an extreme environmental phenomena, such as high winds, tornado, and tornado-borne missiles, as specified in Reg. Guide 1.76, ANSI 57.9, and ASCE 7-88, are considered in the certification of HI-TRAC 100D in Docket No. 72-1014. These loadings bound the environmental loadings at IP2 and IP3. Therefore, a site-specific analysis for the inter-unit transfer operation is not required.

In summary, the STC design and the STC/HI-TRAC assemblage meet all of the acceptance criteria and design objectives specified for these components for all the load cases analyzed.

4.7 Shielding and Dose Evaluation

4.7.1 Introduction

The purpose of the shielding and dose evaluations is to provide the necessary assurance that there will be no unacceptable risk to the health and safety of the public, that occupational doses are within regulatory limits and that these doses are ALARA.

The following have been evaluated:

i. STC dose rates
ii. HI-TRAC 100D dose rates
iii. Dose contribution to site boundary
iv. Occupational exposures for ALARA consideration
v. Effluent doses during normal conditions and postulated accidents

More detailed discussions of the shielding and dose evaluations are provided in Chapter 7 of Enclosure 1.

4.7.2 Methodology

The principal sources of radiation in the STC are the gamma and neutron radiation originating from various sources (e.g., decay of radioactive fission products, spontaneous fission). The neutron and gamma source terms were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 code system using the 44-group library and have been previously utilized in the HI-STORM 100 FSAR. In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. All source term calculations were also performed assuming an infinite array of assemblies during irradiation. The design basis fuel assembly characteristics used in the computations as well as the modeling approach of the gamma and neutron sources are from the HI-STORM FSAR.

The shielding analysis of the STC was performed with MCNP5. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori.

The STC will be placed inside a HI-TRAC for the transfer between IP3 and IP2. Therefore, dose rates are provided for the STC by itself, as well as dose rates for the STC inside a HI-TRAC. In addition, ALARA considerations for occupational exposures received while transferring the STC into the HI-TRAC, from IP3 to IP2, and unloading the STC in IP2 are evaluated. Further, site boundary dose rates from transferring the HI-TRAC containing the STC between IP3 and IP2 for normal and accident conditions are provided.

4.7.3 Acceptance Criteria

The acceptance criteria for the site boundary dose evaluation are 10 CFR 72.104 for normal conditions and 10 CFR 72.106 for accident conditions. The acceptance criteria from 10 CFR 72 were used rather than 10 CFR 100 because the 10 CFR 72 regulations are more restrictive.

The IPEC site boundary is well over 100 meters from any point on the haul path. In the site boundary dose calculations, 50 meters is conservatively used as the site boundary distance except for the effluents which are calculated at 100 meters.

4.7.4 Design Attributes

The STC is a thick-walled vessel containing a fuel basket. The fuel basket design is similar to the Holtec Multi-Purpose Canisters (MPCs) deployed for storing fuel in the HI-STORM system but the STC wall is substantially thicker than the MPC wall to provide enhanced radiation protection. Up to twelve fuel assemblies can be accommodated in the STC fuel basket. For additional shielding the STC is loaded in a HI-TRAC steel-lead-steel transfer cask having a thick bolted lid prior to on-site movement. To minimize fuel and cask temperatures the STC cavity, HI-TRAC annular space between STC and HI-TRAC, and the HI-TRAC water jacket are filled with water.

In addition, procedures and engineering controls based upon sound radiation protection principles will be employed to ensure occupational doses and doses to members of the public are as low as is reasonably achievable (ALARA) in accordance with 10 CFR 20.1101(b).

4.7.5 Key Assumptions and Inputs

4.7.5.1 Source Selection

The source terms for the shielding analysis were selected based on a survey of the current IP3 spent nuclear fuel inventory. The selected source terms represent the most abundant/representative source term characteristics as well as a bounding source term scenario. The initial enrichment is 3.6 wt%, which is a conservative value of minimum enrichment for high burnup fuel.

The source terms were applied in a regionalized loading scheme to the 12 fuel assembly locations available in the STC. A regionalized loading pattern was utilized to more easily be able to transfer hotter fuel in the IP3 spent fuel pit by taking advantage of self-shielding effects. The source terms with the higher cooling times are assigned to the eight outer fuel assembly locations in the STC, while the source terms with the lower cooling time are assigned to the four inner fuel assembly locations.

The following is a summary of the combinations that were utilized for the regionalized loading shielding calculations:

i. Representative
   Outer region (8 assemblies): Burnup of 45 GWd/MTU with a cooling time of 20 years
   Inner region (4 assemblies): Burnup of 55 GWd/MTU with a cooling time of 10 years

ii. Bounding
   Outer region (8 assemblies): Burnup of 45 GWd/MTU with a cooling time of 20 years
   Inner region (4 assemblies): Burnup of 55 GWd/MTU with a cooling time of 5 years

iii. Site Boundary
   Outer region (8 assemblies): Burnup of 55 GWd/MTU with a cooling time of 5 years
   Inner region (4 assemblies): Burnup of 55 GWd/MTU with a cooling time of 5 years

As can be seen for the site boundary calculation, a conservative, uniform source loading pattern, was utilized and the dose rate contribution from the radial and top HI-TRAC surfaces were considered.

4.7.5.2 Shielding Model

The shielding analysis of the STC was performed with MCNP5. This means that no gross approximations were required to represent the STC in the shielding analysis. The MCNP model of the HI-TRAC for normal conditions have the water jacket filled, but does not credit the water for the hypothetical accident condition. In the shielding analysis, the STC and the HI-TRAC inner cavities are only partially filled with water leaving an air gap under the lids. The air gaps are needed to provide an expansion zone for the water and also allow the STC lid operations to occur unhindered by water.

Only intact fuel assemblies are permitted for loading into the STC. Thus the shielding model assumes the source material remains in the fuel lattice configuration with no relocation.

4.7.5.3 Effluent Dose Evaluation

The consequence of an effluent dose release during normal and accident conditions has been analyzed using the calculational methodology in Chapter 7 of the HI-STORM FSAR Revision 1, which is based on the guidance in NUREG-1536 and SFST/ISG-5 and NUREG/CR-6487. For normal conditions, a fraction of fuel rods are assumed to be breached in the STC causing a dose release over a period of time while for accident conditions all 12 fuel assemblies in the STC are assumed to be breached and causing an instantaneous release. Only gases are considered in the analysis, since the shielded transfer canister remains flooded during all operations, such that fines, volatiles and crud would remain entrapped within the water environment. This is also consistent with the analysis of the fuel handling accident dose analysis in Chapter 14 of the IP3 UFSAR for the spent fuel pit.

The radiological inventory is based on the design basis assembly from the HI-STORM FSAR, Revision 1, which assumes a minimum cooling time of 5 years and an assembly average burnup of 70,000 MWD/MTU, which are bounding for the IP3 fuel assemblies to be transferred. The radiological source term and other input parameters that are used in the analysis are provided in Table 7.4.4 of Enclosure 1. These parameters are based on the following consideration:

i. Dose Conversion Factors from EPA Federal Guidance Report No. 11, Table 2.1
ii. The atmospheric dispersion factor is determined at 100 meters, based on Reg. Guide 1.145. This distance is also consistent with 10 CFR 72. In addition, the resulting dose values are so low that this utilized distance is sufficient for the current case.
iii. The cavity free volume is based on a 6" gap between the bottom of the shielded transfer canister lid and the enclosed water. This gap includes thermal expansion of the water after loading.
iv. The transportation time between IP3 and IP2 under normal conditions is 8 hours, while 30 days represents accident conditions.
v. Doses from submersion in the plume are neglected because they are shown to be small compared to inhalation doses.
vi. The release fractions for gases are based on SFST/ISG-5, which specifies 0.30.
vii. The leak rate testing performed on the STC seals verifies the leak rate to be less than or equal to 1×10^{-2} std-cm^3/s. This water-tight acceptance criterion is derived from ASTM E1003-05, paragraph 11.3. However, it is conservatively assumed that the maximum possible leakage rate from the confinement vessel is 150% of the maximum leakage rate acceptance criteria.

4.7.6 Results and Conclusions

4.7.6.1 STC Dose Rates

The calculated dose rate results for the STC are presented in Table 4.5 for a regionalized loading pattern. Total surface dose rates for regionalized loading with fuel burnups of 45 GWd/MTU and 20 years cooling time in the outer region and 55 GWd/MTU and 5 years cooling time in the inner region were found to be on the order of 2 to 3 rem/hr in the radial and top axial directions. Exchanging the inner region for fuel with a burnup of 55 GWd/MTU and 10 years cooling time reduces the surface dose rates by approximately 13% in the radial direction and about 35% in the axial directions.

4.7.6.2 HI-TRAC 100D Dose Rates

The calculated dose rate results for the HI-TRAC 100D containing the STC are presented in Table 4.6 with regionalized loading pattern of fuel burnup of 45 GWd/MTU and 20 years cooling time in the outer region and fuel burnup of 55 GWd/MTU and 10 years cooling time in the inner region. The calculated HI-TRAC surface dose rates were found to be low, with the highest surface dose rate being 0.17 rem/hr in the top axial direction.

4.7.6.3 Dose Contribution to Site Boundary

The dose contribution from the STC inside the HI-TRAC when transported between IP3 and IP2 was found to be 0.62 mrem under normal conditions. The distance from the HI-TRAC to the site boundary is conservatively estimated to be 50 m. This dose value is based on an estimated transportation time of 8 hours. A uniform source term with 55 GWd/MTU burnup and 5 years cooling time was utilized for this calculation and the dose rate contribution from the radial and top HI-TRAC surfaces were considered.

The dose contribution from the STC inside the HI-TRAC when transported between IP3 and IP2 was found to be 64.8 mrem under accident conditions (extended time of residence in the HI-TRAC). This dose value is based on an estimated recovery time of 30 days consistent with the HI-STORM FSAR, at 50 m away from the HI-STAR surface. The accident condition considers loss of water in the HI-TRAC jacket. The source term with 55 GWd/MTU burnup and 5 years cooling time was utilized for this calculation and the dose rate contribution from the radial and top HI-TRAC surfaces were considered. The doses cited are the conservative equivalent to TEDE (without accounting for any self-shielding from the human body).

4.7.6.4 Occupational Exposures for ALARA Consideration

The overall personnel (person-rem) exposure received from transferring the STC in and out of the HI-TRAC have been determined based on dose rates for regionalized loading with fuel burnups of 45 GWd/MTU and 20 years cooling time in the outer region and 55 GWd/MTU and 10 years cooling time in the inner region. Note that these dose rates are based on the usage of long-reach tools for the operators to prevent them from having to come in direct contact with the STC. It is estimated that personnel involved with the inter-unit transfer would receive less than 1.2 person-rem for each inter-unit transfer of fuel.

4.7.6.5 Effluent Dose Evaluation

The activity released from the shielded transfer canister is assumed to be released at the maximum rate for one transfer between IP3 and IP2 during normal (8 hour duration) and accident conditions (30 day duration). Since no filtration or isolation of the release path is included in the model, this analysis supports a potential effluent release during all anticipated operations. The resulting doses are presented in Table 4.7. The doses are much smaller than from direct radiation. In addition, the resulting doses from the accident condition are significantly lower than the limits established in 10 CFR 72.106 (e.g., 5 rem TEDE, 50 rem TODE=CDE+DDE) and well below the doses rates calculated (for the site boundary and control room occupants) for the fuel handling accident presented in Chapter 14 of the IP3 UFSAR. Because the 10 CFR 72.106 limits are lower than those of 10 CFR 100.11, the part 100 limits are met by comparison.

In conclusion, it has been demonstrated that the STC design promotes reasonable dose rates during the short period of time when the STC is moved from the SFP into the HI-TRAC and is acceptable as part of the annual dose incurred at the plant and is in accordance with ALARA. In addition, the HI-TRAC dose rates are very low when containing the STC and are also in accordance with ALARA. It can be further concluded that the dose contribution to the site boundary from the STC is very small. Further, the dose contributions from both normal and accident conditions from the STC are well below regulatory limits.

Table 4.5

Total Dose Rates at Various Distances Around the STC
Regionalized Loading Pattern with Fuel Burnup of 45 GWD/MTU and 20 years Cooling Time in Outer Region and 55 GWD/MTU Fuel Burnup with 5 years Cooling Time in Inner Region

Dose Rate Location	Total (mrem/hr)		
	Radial Surface of STC	Top Lid of STC	Bottom Plate of STC
Surface	3141.7	2033.8	6494.7
10 meters way from surface	37.0	10.7	53.6

Table 4.6

Total Dose Rates at Various Distances Around the HI-TRAC 100D
Regionalized Loading Pattern with Fuel Burnup of 45 GWD/MTU and 20 years Cooling Time in Outer Region and 55 GWD/MTU Fuel Burnup with 10 years Cooling Time in Inner Region

Dose Rate Location	Total (mrem/hr)		
	Radial Surface of HI-TRAC 100D	Top Lid of HI-TRAC 100D	Bottom Lid of HI-TRAC 100D
Surface	1.03	170.1	9.58
10 meters way from surface	0.03	0.70	0.24
50 meters way from surface	0.003	0.02	0.009

Table 4.7

Doses from Effluent Release at 100 meters

Effected Component	Dose (mrem) 8 hour duration*	Dose (mrem) 30 day duration**	Dose (mrem) Instantaneous accident release***
Whole Body - TEDE	0.009	0.796	278.5
Max Organ (Thyroid) - TODE	0.065	5.83	2039.6

* This dose value is based on an estimated transportation time of 8 hours

** This dose value is based on an estimated recovery time of 30 days consistent with the HI-STORM FSAR, at 50 m away from the HI-STAR surface.

*** This dose is based on the accident condition where all 12 fuel assemblies in the STC are assumed to be breached and causing an instantaneous release

4.8 Materials Evaluation

4.8.1 Introduction

The design of the STC includes materials that have a proven history of successful use in nuclear power plant applications, including submersion in borated spent fuel pit water. Guidance from NRC SFST Interim Staff Guidance (ISG)-15 (Reference 2) was used in selecting materials for the STC design. The HI-TRAC is an existing component that has been used several times at IP2. Thus, the STC enclosure vessel materials are the same as those used in HI-TRAC. Similarly, the STC fuel basket and neutron absorber materials are the same as those used in Holtec's MPC.

A more detailed discussion of the materials evaluation is provided in Chapter 8 of Enclosure 1.

4.8.2 Materials of Fabrication

The materials of fabrication for the STC enclosure vessel, the STC fuel basket, and the HI-TRAC transfer cask are discussed in detail in Section 8.2 of Enclosure 1 and are shown on the drawings in Enclosure 1. Table 8.2.1 of Enclosure 1 presents the materials, the components using these materials, and the service environment. ASME SA515 Grade 70, SA516 Grade 70, or SA36 carbon steel is used in the STC enclosure vessel and HI-TRAC transfer cask. All carbon steel materials will be coated with a material suitable for the service conditions of the component (see Table 8.2.1 of Enclosure 1 for specific coating products). The STC fuel basket is constructed of stainless steel with Metamic neutron absorber affixed to the basket panel walls inside stainless steel sheathing.

The "Alloy X" approach to analyzing the stainless steel parts was used in the structural analysis. The Alloy X approach has been successfully used by Holtec in its dry storage canister product line. The Alloy X approach permits fabrication of stainless steel parts from any of the following types of stainless steel: 304, 304LN, 316, or 316LN. The structural analysis bounds the use of all of these stainless steel types by using the most limiting material properties among the four steel types.

The STC top lid and HI-TRAC top and bottom lids are bolted lids that employ elastomeric seals to preserve the pressure boundary integrity of the two enclosure vessels.

4.8.3 Degradation Mechanisms

The potential degradation mechanisms for the materials are discussed in Section 8.3 of Enclosure 1. Chemical and galvanic corrosion, embrittlement, fatigue, stress corrosion cracking, loss of neutron capture capability, flammable gas generation, and neutron absorber panel swelling are addressed, as applicable to the material.

4.8.4 Conclusions

All materials of construction for the STC and HI-TRAC are readily available, proven by previous nuclear service history, and suitable for the service conditions expected in inter-unit transfer operations.

4.9 Acceptance Tests and Maintenance

The applicable tests and inspections applied to the STC are governed by the applicable design codes, and include:

i. Visual inspections and measurements
ii. Non-destructive weld examination
iii. Structural and pressure tests
iv. Leakage tests
v. Component and material tests

An ongoing maintenance program will be developed and incorporated into the Operations & Maintenance (O&M) Manual for the STC, which will be prepared and issued prior to the delivery and first use. This document will delineate the detailed inspections, tests, and parts replacement necessary to ensure continued radiological safety, proper handling, and continued performance of the STC in accordance with the design requirements. There are no active components or systems required to assure the continued performance of the safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from weathering effects and pre- and post-usage requirements. Typical of such maintenance would be the removal of scratches, dents, etc. from accessible external surfaces to eliminate locations for potential contaminant hideout; seal replacement; and re-coating surfaces. Such maintenance requires methods and procedures no more demanding than those routinely used at power plants.

A more detailed discussion of the maintenance program is provided in Chapter 8 of Enclosure 1.

4.10 Operating Procedures

Procedures will be developed for STC preparation and setup, loading, unloading, and transfer of spent fuel in the STC. The design of the STC, including developed procedures, the ancillary equipment and the plant Technical Specifications, serve to minimize risks and mitigate consequences of potential events. The primary objective of the procedures will be to reduce the risk of occurrence and then to mitigate the consequences of the event should it occur.

A more detailed discussion of the operating procedures is provided in Chapter 10 of Enclosure 1.

4.11 Security

The physical security of the STC and the movement of the spent fuel elements contained within the STC will be reviewed by the IPEC Security organization as part of the inter-unit fuel transfer project. This includes a detailed review of the heavy haul path that the VCT will travel from IP3 to IP2 carrying the HI-TRAC/STC assemblage loaded with spent fuel. The heavy haul path will be entirely within the IPEC owner protected area (PA). Entergy has determined that the PA boundary will need to be modified to accommodate the physical clearances and VCT turning radii needed to make the trip from IP3 to IP2. These modifications will be performed under a separate project and scheduled for completion at a time commensurate with the need to transfer fuel from IP3 to IP2. Entergy commits to moving the PA boundary as described prior to the first inter-unit transfer (see Commitment #1 in Attachment 6).

The IPEC security plan and procedures will be reviewed and revised as required to provide the necessary protection for movement of special nuclear material in the form of spent fuel from the IP3 spent fuel pit to the IP2 spent fuel pit in accordance with 10 CFR 73 requirements. A review of the plan and procedure changes will be performed in accordance with the requirements of 10 CFR 50.54(p).

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The no significant hazards consideration for IP2 pertaining to this proposed amendment is provided in Attachment 4. The IP3 no significant hazards consideration pertaining to this proposed amendment is provided in Attachment 5.

5.2 Applicable Regulatory Requirements/Criteria

By virtue of holding 10 CFR 50 operating licenses for IP2 and IP3, ENO also holds licenses under 10 CFR 30, 40, and 70 which together permit ENO to receive, possess, and use special nuclear material, byproduct material, source material, and neutron startup sources required for, and resulting from the operation of the two reactors. 10 CFR 70.42, "Transfer of Special Nuclear Material," permits the transfer of special nuclear material under certain circumstances and between certain entities. 10 CFR 70.42(b)(5) permits a licensee to transfer special nuclear material "to any person authorized to receive such special nuclear material under terms of a specific license or general license or their equivalents issued by the Commission or an Agreement State."

In accordance with 10 CFR 70.42(b)(5), this amendment request seeks to create the appropriate terms in the IP2 10 CFR 50 operating license to permit IP2 to receive and possess, but not use, special nuclear material from IP3. The type of special nuclear material is spent nuclear reactor fuel in its original fuel assembly configuration.

The following 10 CFR 50, Appendix A, General Design Criteria (GDC) have been identified as applicable to this request and the manner in which Entergy continues to comply with each GDC is described in each case. Appropriate modifications to the IP2 and IP3 updated Final Safety Analysis Reports will be made after issuance of the amendment.

- GDC 1, *Quality standards and records*

  Holtec's Nuclear Quality Assurance (QA) program is the governing program for the design and construction of the STC (USNRC Docket No. 71-0784). The program complies with 10 CFR 50 Appendix B. Entergy's QA program will apply to onsite maintenance and testing activities associated with the STC in addition to the balance of the equipment used in the inter-unit transfer.

- GDC 2, *Design Bases for Protection Against Natural Phenomena* requires in part that structures, systems, and components 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.

The HI-TRAC 100D transfer cask is designed to provide protection to the STC and the SNF against extreme environmental phenomena loads, such as tornado missiles, during short term operations. The loadings from an extreme environmental phenomena, such as high winds, tornado, and tornado-borne missiles, as specified in Reg. Guide 1.76, ANSI 57.9, and ASCE 7-88, are considered in the certification of HI-TRAC 100D in Docket No. 72-1014. These loadings bound the environmental loadings at IP2 and IP3. In addition, the haul path between IP3 and IP2 is not subject to flooding.

- GDC 3, *Fire protection* requires in part that structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

  A postulated 50-gallon transporter fuel tank rupture and fire accident has been analyzed. The fuel cladding temperature, material temperatures and STC pressure were determined to be within design limits. Entergy will perform a site specific assessment of the potential for fire along the haul path prior to each inter-unit transfer to ensure that the accident is bounded by the fire analysis which considered a fire, fueled by 50 gallons of combustible fuel, engulfing the loaded HI-TRAC 100D transfer cask.

- GDC 4, *Environmental and Dynamic Effects Design Bases* requires in part that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

  The STC is designed of materials that have a proven history of successful use in nuclear power plant applications, including submersion in borated spent fuel pit water. Guidance from NRC SFST Interim Staff Guidance (ISG)-15 was used in selecting materials for the STC design. The HI-TRAC is an existing component that has been used several times at IP2. Thus, the STC enclosure vessel materials are the same or equivalent to those used in HI-TRAC. Similarly, the STC fuel basket and neutron absorber materials are the same as those used in Holtec's MPC.

- GDC 5, *Sharing of Structures, Systems, and Components* requires that structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

  It is proposed that the IP2 spent fuel pit will be used to store IP3 fuel and therefore the IP2 pit will be shared for the purpose of storing spent fuel which is a safety function from the standpoint of protection against unacceptable radiological releases. The proposed LCO 3.7.18 *Spent Fuel Assembly Transfer*, provides the limits for fuel assembly loading into the STC in the IP3 spent fuel pit. This LCO ensures that the IP3 fuel assemblies selected for loading into the STC and the arrangement of those assemblies in the STC fuel basket is always bounded by the criticality, thermal, shielding, and accident analyses supporting the design. These limits also ensure that the fuel assemblies being transferred in the STC can be stored in the IP2 spent fuel storage rack locations per IP2 TS 3.7.13 *Spent Fuel Pit Storage* as designated on the fuel move sheets. Therefore, the sharing of the IP2 SFP will not adversely affect the ability of the IP2 spent fuel pit to perform its function since adequate storage and cooling are provided for both IP2 and IP3 spent fuel.

- GDC 60, *Control of Releases of Radioactive Materials to the Environment* requires that the nuclear power unit design shall include means to control suitably 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. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

  The provision of residual heat removal capability that meets the requirements of GDC 61 ensures that STC integrity is not challenged due to heatup of the contained water and resulting pressurization.

- GDC 61, *Fuel Storage and Handling and Radioactivity Control* requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These 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 HI-TRAC 100D transfer cask, the STC and other equipment relied upon during the inter-unit transfer of fuel has been designed to accommodate the necessary periodic inspection and testing. The HI-TRAC 100D, LPT, and VCT are existing equipment already utilized at the plant for dry cask storage and have established inspection and testing programs. The STC, upgraded IP3 cask handling crane, and new ancillary equipment will be inspected in accordance with the manufacturer's recommendations.

  The STC will be placed in the IP3 spent fuel pit for loading and moved into the HI-TRAC transfer cask for preparation in the IP3 FSB. The STC will be moved to IP2 inside the HI-TRAC and once again removed from the HI-TRAC in the IP2 FSB for placement into the IP2 spent fuel pit and fuel unloading. Therefore, dose rates have been determined for the STC by itself, as well as for the STC inside a HI-TRAC. In addition, occupational exposures are estimated for transferring the STC into and out of the HI-TRAC, moving the HI-TRAC from IP3 to IP2, and unloading the STC in IP2. Further, site boundary dose rates from transferring the HI-TRAC containing the STC between IP3 and IP2 for normal and accident conditions have been determined. The dose rates external to the STC during the short time it is not inside the HI-TRAC are manageable with appropriate radiation protection controls and the dose rates from the loaded HI-TRAC have been shown to be extremely low. The occupational radiation exposure is estimated to be less than 1.2 person-rem per transfer of 12 spent fuel assemblies. This small increase in radiation dose would not affect the ability to maintain individual occupational doses within the limits of 10 CFR 20 and are as low as reasonably achievable. The estimated annual radiological dose commitments to a maximally exposed member of the public at the site boundary due to the fuel transfer would be approximately 5 mrem. This estimated total annual dose commitment is within the limitations of the IP2 and IP3 Technical Specifications which are based on the offsite dose requirements of 10 CFR Parts 20, 50, and 40 CFR 190.

In the unlikely event that one fuel assembly be dropped on the STC during loading or unloading the radioactive releases would be substantially less than the doses predicted for the existing fuel handling accidents described in Chapter 14 of each units UFSAR. The existing FHA for each unit assumes high burnup and that the dropped fuel assembly has a cooling time of only 84 hours whereas the STC loading requirements dictate that the fuel must have a cooling time of greater than or equal to 5 years. Therefore, the existing FHA accident source term for the dropped fuel assembly is substantially greater than would be encountered for any fuel assembly during STC loading and unloading.

The robust design of the fuel-loaded STC as demonstrated by structural analysis ensures that the pressure boundary integrity of the STC and the HI-TRAC 100D transfer cask are maintained under all normal, off-normal, and extreme environmental conditions, as well as all credible accident events. In addition, the cask transport systems (air pads, LPT with Hilman Rollers, rail platform, VCT) have been shown to be structurally sound as determined by structural analysis.

The transport roadway and safety-related components that could be affected by the inter-unit transfer of fuel must be shown to have acceptable structural capacity including the consideration of underground utilities. Entergy commits to performing this evaluation and implementing any modifications determined to be necessary prior to the first inter-unit transfer operation.

The residual heat removal capability of the STC within the spent fuel pit and also within the HI-TRAC 100D has been determined under adverse environmental conditions and found to be acceptable provided certain limitations on fuel characteristics are met. These limitations are that the cooling time must be greater than or equal to 5 years, that peripheral STC cells contain assemblies with decay heat loads of less than or equal to 650 Watts, and that STC interior cells contain assemblies with decay heat loads less than or equal to 1105 Watts. These fuel characteristic limitations are imposed via proposed TS 3.7.18 *Spent Fuel Assembly*

Transfer and will ensure residual heat removal capability during all phases of the inter-unit transfer operation.

Detailed instructions will be available for use by qualified cask handling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. In addition to these robust fuel handling procedures and safety features, the inadvertent loading of a high decay heat fuel assembly in the STC would be detected by a procedural requirement to take temperature readings for each fuel assembly using a remote indicating temperature measuring device prior to STC lid closure. A fuel assembly identified by an abnormally high exit temperature would be returned to its authorized location within the fuel racks. From a thermal perspective the breakdown of the cask transporter outside for an extended period of time represents an anticipated but not expected condition. A steady state thermal analysis of this condition has determined that under adverse environmental conditions that residual heat removal capability is not compromised and thermal acceptance criteria remain satisfied.

A significant reduction in STC water inventory is prevented under accident conditions by component design and procedural controls that limit the maximum drop height to 6" while being lifted by the cask transporter. The vertical drop of the HI-TRAC 100D while loaded with the STC has been analyzed with no resulting loss of integrity of the STC and HI-TRAC.

The collapse of the roadway during transport resulting in cask rollover and potential loss of water inventory and potential large radioactive release is not a credible accident because of the structural stability of the VCT and the measures taken to ensure that the haul path is structurally sound. These measures include an evaluation of the haul path that includes ground penetrating radar and/or soil compaction studies.

Notwithstanding the fact that a cask roll over is not a credible accident a large radioactive release from the cask has been evaluated. The results of the analysis that hypothetically assumes all 12 of the assemblies release their gases into the STC cavity are bounded by the fuel handling accident presented in Chapter 14 of the UFSAR.

- GDC 62, *Prevention of Criticality in Fuel Storage and Handling* requires that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

  The existing IP2 and IP3 Technical Specifications govern the spent fuel pit boron concentration, the maximum U-235 fuel enrichment that can be stored in the SFP, and the loading restrictions based on cooling time, initial fuel enrichment, IFBA loading, and fuel burnup. In addition, the existing TSs govern criticality requirements, which include maintaining the effective multiplication factor (K_{eff}) less than or equal to 0.95 associated with fuel stored in the SFP. Spent fuel loading is also governed by 10 CFR 50.68, *Criticality Accident Requirements*. Criticality evaluations are performed for spent fuel that will be stored in the SFP based on the requirements set forth in 10 CFR 50.68.

Relative to criticality concerns, the IP2 and IP3 SFPs has been reviewed for placement of an STC in the pit, and the IP2 SFP has been reviewed for storage of IP3 fuel and IP2 Technical Specification 3.7.13 compliance with the zone storage requirements. Controls will be established on loading fuel assemblies such that only the analyzed fuel assemblies will be loaded in the STC in the IP3 SFP. These controls will be based on the requirements of proposed TS 3.7.18 *Spent Fuel Assembly Transfer* that places the appropriate limitations on fuel characteristics (e.g., cooling period, enrichment, burnup, etc.) such that subcriticality is maintained at all times per regulatory guidance provided in 10 CFR 50.68.

Detailed instructions will be available for use by qualified cask handling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. In addition to these robust fuel handling procedures and safety features, the inadvertent loading of a fresh fuel assembly in the STC has been analyzed from a criticality perspective and it has been shown that the requirements of 10 CFR 50.68 continue to be met. Similarly, one fresh fuel assembly dropped on the STC during loading has been analyzed with no impact on criticality.

- GDC 63, *Monitoring fuel and waste storage* requires 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.

  A temperature indicating device will be used to monitor water temperature above each fuel assembly in the loaded STC prior to removal from the SFP. Relatively high temperatures would be indicative of a misloaded high decay heat fuel assembly.

  A pressure gauge or gauges will be used to monitor pressures during leak rate testing of the STC and HI-TRAC.

5.3 <u>Environmental Considerations</u>

Entergy has evaluated the proposed changes and determined that the changes do not involve (1) a significant hazards consideration (2) a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c) (9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

Movement of spent fuel from one unit to another on the same plant site has been previously approved on at least two occasions:

- St. Lucie Unit 2: Amendment 30 to Operating License NPF-16, Docket 50-389, May 10, 1988 (ML013600315).
- San Onofre plant: Amendments 63 and 52 to Operating Licenses NPF-10 and NPF-15, respectively; Dockets 50-361 and 50-362, June 22, 1988 (ML021980192).

Precedence for performing Part 50-based criticality analyses on spent fuel storage canisters while being loaded in a plant's spent fuel pit may be found in the following licensing actions:

- Farley plant: Amendments 169 and 161 to Operating Licenses NPF-2 and NPF-8, respectively; Dockets 50-348 and 50-364, dated June 28, 2005 (ML051860200).
- Fort Calhoun Station: Amendment 239 to Operating License DPR-40, Docket 50-285, dated April 10, 2006 (ML061000606).

7.0 REFERENCES

1. Holtec Report HI-2094289, Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at Indian Point Energy Center, Revision 1.

2. NRC SFST Interim Staff Guidance (ISG)-15, "Materials Evaluation," January 2001.

3. Holtec International Final Safety Analysis Report for the HI-STORM 100 System, Docket 72-1014, latest revision.

4. NRC SFST Interim Staff Guidance (ISG)-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3.

ATTACHMENT 2 TO NL-09-076

MARKED-UP IP2 OPERATING LICENSE CONDITIONS, TECHNICAL SPECIFICATIONS AND TECHNICAL SPECIFICATION BASES PERTAINING TO THE INTER-UNIT TRANSFER OF SPENT FUEL FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

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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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 260, 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. Indian Point Unit 3 spent fuel may be transferred to the Indian Point Unit 2 spent fuel pit, as needed.

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:

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PROPOSED NEW SPECIFICATION

3.7 PLANT SYSTEMS

3.7.15 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 3.7.15 IP3 spent fuel assemblies transferred to IP2 via the STC must be either in an approved IP2 spent fuel pit storage rack location per 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 spent fuel pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more fuel assemblies not in the required location.	---------- **- NOTE -** LCO 3.0.3 is not applicable. ---------- A.1.1 Initiate action to restore compliance with LCO 3.7.15.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	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.

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4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} < 1.0$ if fully flooded with unborated water, and

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

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

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

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

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

4.3.2 <u>Drainage</u>

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

4.3.3 <u>Capacity</u>

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

4.4 Shielded Transfer Canister

4.4.1 <u>Criticality</u>

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, and

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

4.0 DESIGN FEATURES

4.4 Shielded Transfer Canister (continued)

4.4.2 Drainage

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

4.4.3 Capacity

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

B 3.7 PLANT SYSTEMS

B 3.7.15 Shielded Transfer Canister (STC) Unloading

BASES

BACKGROUND

As required by plant operations IP3 spent fuel is transferred to the IP2 spent fuel pit in order to maintain adequate fuel storage capacity in the IP3 spent fuel pit. IP3 spent fuel moved to the IP2 spent fuel pit is subsequently transferred to dry cask storage at the IPEC on-site Independent Spent Fuel Storage Installation (ISFSI) as part of spent fuel inventory management in the IP2 spent fuel pit. This inter-unit transfer is necessary because the cask handling crane at IP3 does not have high enough load-bearing capacity to lift and handle the dry storage transfer cask and canister.

Inter-unit fuel transfer operations are conducted using the Shielded Transfer Canister (STC) and the HI-TRAC 100D transfer cask. The STC is a bolted-lid pressure vessel with an internal fuel basket that accommodates up to twelve IP3 spent fuel assemblies. The STC is loaded in the IP3 spent fuel pit, placed into the HI-TRAC transfer cask in the Fuel Storage Building (FSB) truck bay, and moved outside the truck bay on air pads or other approved conveyance. The STC/HI-TRAC assemblage is transported from outside the IP3 FSB truck bay to just outside the IP2 FSB truck bay with a vertical cask transporter (VCT) and moved into the IP2 FSB truck bay on a low profile transporter.

The STC is removed from the HI-TRAC using the cask handling crane and placed into the IP2 spent fuel pit. The STC lid is removed and the IP3 fuel assemblies are moved to their designated IP2 wet storage rack cell locations with the spent fuel bridge crane.

Fuel assemblies to be transferred are chosen at IP3 based on the requirements for loading in the STC. The STC fuel loading requirements are such that the fuel chosen for transfer to IP2 is suitable for storage in the IP2 spent fuel pits storage racks and there are open fuel cells available. Fuel move sheets will govern the transfer of the spent fuel from IP3 to IP2.

BASES

APPLICABLE SAFETY ANALYSES

The STC has been analyzed for criticality prevention, heat rejection capability, shielding, and structural integrity to ensure safe transfer operations from the time that the STC is loaded at IP3 to the time it is unloaded at IP2 (refer to Reference 1 for details).

The criticality analysis and the limits on fuel selection prescribed in the IP3 Technical Specifications (LCO 3.7.18) ensure that the effective neutron multiplication factor (k_{eff}) of a loaded STC in its most reactive configuration remains less than 0.95.

The thermal analysis shows that the fuel cladding temperature remains below the acceptance criteria of 752°F and 1058°F, for normal and accident conditions respectively, at all times during inter-unit transfer and that the design pressure and temperature of the STC are not exceeded.

The structural analysis shows that the STC and HI-TRAC maintain their structural integrity under all normal, off-normal, and credible accident conditions. There are no radioactive material releases from the STC or HI-TRAC during any condition of operation.

The shielding analysis shows that the dose rates from the STC during the short time it is not inside the HI-TRAC are manageable with appropriate radiation protection controls. Dose rates from the loaded HI-TRAC are shown to be extremely low.

LCO

LCO 3.7.15 governs the presence of the STC in the IP2 spent fuel pit. The STC arrives at IP2 with its bolted lid in place, which preserves the fuel types and fuel locations established at IP3 when the STC was loaded. Once the STC lid is removed at IP2, this LCO requires that a transferred fuel assembly be in one of three places:

1. In an approved IP2 spent fuel pit storage rack location per LCO 3.7.13, or
2. In an authorized STC fuel basket cell, or
3. In transit between these two locations

This LCO preserves the assumptions of the safety analyses for the STC and the IP2 spent fuel pit.

This LCO is modified by two notes. Note 1 specifies that only IP3 spent fuel assemblies are permitted to be in the STC. The STC design and analysis is

BASES

LCO (continued)

based on IP3 fuel assemblies. Loading of IP2 fuel assemblies in the STC is not authorized. Note 2 specifies that once each IP3 fuel assembly removed from the STC is placed in an IP2 spent fuel pit storage rack location and released from the spent fuel bridge crane, it may not be returned to the STC. This note prevents loading IP3 fuel out of the IP2 spent fuel pit for transfer back to the IP3 spent fuel pit. This is not an authorized evolution.

APPLICABILITY

The LCO is applicable whenever the STC is in the IP2 spent fuel pit.

ACTIONS

A.1

When any IP3 spent fuel assembly transferred to the IP2 spent fuel pit is not in one of the three authorized locations, LCO 3.7.15 is not met. Required Action A.1 specifies that action begin immediately to restore compliance with the LCO. The affected fuel assemblies must be placed in an authorized location without delay.

The completion time of "Immediately" is appropriate because fuel located in the STC or the spent fuel pit racks may be in an unanalyzed condition and action is required to be initiated and completed without delay to restore fuel location to an analyzed configuration.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. This is appropriate because inter-unit transfer and storage of spent fuel has no impact on safe reactor or power generation operations. Shutting down the plant due to non-compliance with LCO 3.7.15 would have no safety benefit.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

SR 3.7.15.1 requires that any IP3 fuel assembly being returned to the STC be verified by administrative means to have been returned to the same STC fuel cell location from which it was removed. This SR ensures that the loading pattern authorized when the STC was loaded at IP3 is preserved.

BASES

SURVEILLANCE REQUIREMENTS (continued

This SR does not require the same verification for placing the fuel assembly in a spent fuel pit cell location because that process is governed by a separate LCO (3.7.13) and a fuel move sheet is required to place the fuel assembly in any location in the storage racks.

REFERENCES

1. Holtec Report HI-2094289, Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at Indian Point Energy Center, Revision 1.

ATTACHMENT 3 TO NL-09-076

MARKED-UP IP3 TECHNICAL SPECIFICATIONS AND TECHNICAL SPECIFICATION BASES PERTAINING TO THE INTER-UNIT TRANSFER OF SPENT FUEL FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

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1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro curies per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. If a specific isotope is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988.
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of XE-133 (micro curies per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138, actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil".
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).**
L_a	The maximum allowable primary containment leakage rate, L_a shall be 0.1% of primary containment air weight per day at the calculated peak containment pressure, (P_a).
LEAKAGE	LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pumps seals or valve packing (except for leakage into closed systems and

(continued)

1.1 Definitions (continued)

LEAKAGE (continued)

reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

(Leakage into closed systems is leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except for leakage into closed systems and RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant loop temperature, and reactor

(continued)

PROPOSED NEW SPECIFICATION

3.7 PLANT SYSTEMS

3.7.18 Spent Fuel Assembly Transfer

LCO 3.7.18 INTACT FUEL ASSEMBLIES placed into the Shielded Transfer Canister (STC) shall be classified in accordance with Figure 3.7.18-1 based on initial enrichment and burnup and shall be restricted based on the following:

a. INTACT FUEL ASSEMBLIES classified as Type 2 and meeting the following may be placed in the STC basket (see Figure 3.7.18-2) as follows:

1. Post-irradiation cooling time ≥ 5 years;
2. Average burnup per fuel assembly ≤ 55,000 MWD/MTU;
3. Decay heat ≤ 650 Watts (any cell);
4. Decay heat ≤ 1105 Watts (cell 1, 2, 3 or 4).

- NOTE -

If one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, AND 4 must be empty.

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b. INTACT FUEL ASSEMBLIES classified as Type 1 or Type 2 and meeting the following may be placed in cells 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket (see Figure 3.7.18-2) as follows:

1. Post-irradiation cooling time ≥ 5 years;
2. Average burnup per assembly ≤ 55,000 MWD/MTU;
3. Decay heat ≤ 650 Watts.

APPLICABILITY: Whenever any fuel assembly is placed in the STC.

PROPOSED NEW SPECIFICATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more fuel assemblies in the STC do not meet the LCO limits.	---------NOTE--------- LCO 3.0.3 is not applicable. ---------------------- A.1.1 Initiate action to restore compliance with LCO 3.7.18. <u>OR</u> A.1.2 Initiate action to move fuel to the spent fuel pit in accordance with LCO 3.7.16.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify by administrative means that the fuel assembly meets the requirements specified in the LCO for placement in the STC.	Prior to placing the fuel assembly in the STC.

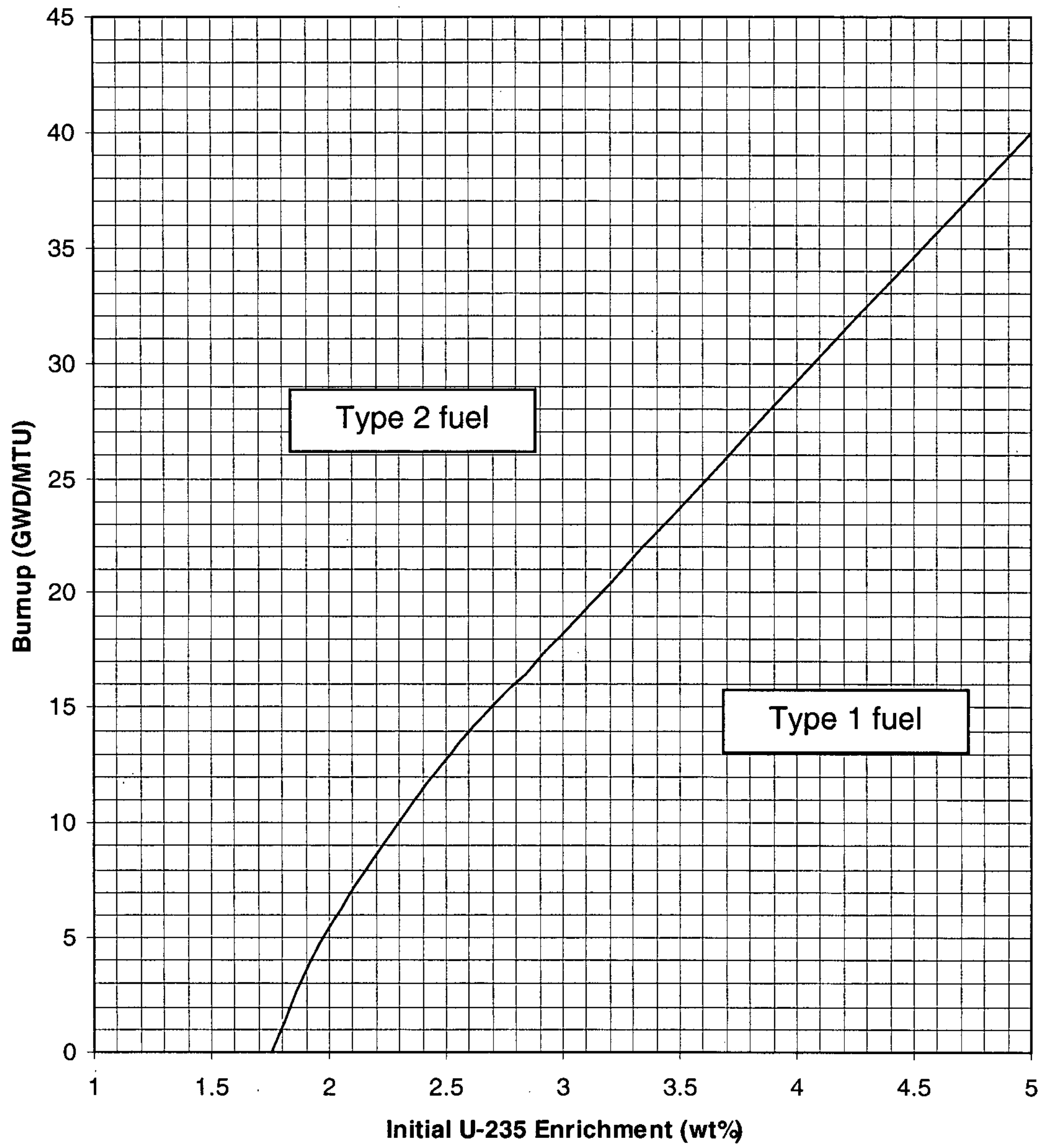

Figure 3.7.18-1
Fuel Assembly Classification for Transfer in the Shielded Transfer Canister

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- NOTE -

For fuel assemblies exposed to Hafnium inserts during irradiation the burnup to be compared to the above curve is the burnup prior to the exposure to the Hafnium insert.

PROPOSED NEW SPECIFICATION

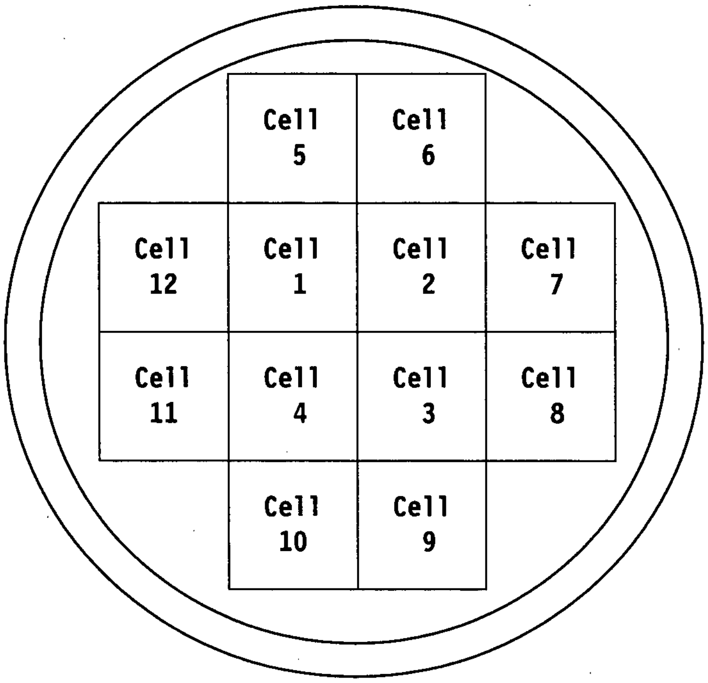

Figure 3.7.18-2
Shielded Transfer Canister Layout
(Top View)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

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

4.4 Shielded Transfer Canister

4.4.1 Criticality

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.

4.4.2 Drainage

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

4.4.3 Capacity

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

B 3.7 PLANT SYSTEMS

B 3.7.18 Spent Fuel Assembly Transfer

BASES

BACKGROUND

In the Shielded Transfer Canister (STC) design, the fuel basket is divided rectilinearly into twelve cells as shown in Figure 3.7.18-2, "Shielded Transfer Canister Layout (Top View)". All cells are sized to contain IP3 spent fuel assemblies. All cells are designed and analyzed for fuel of a certain enrichment, burnup, cooling time, and decay heat. The inner cells are designed and analyzed to contain fuel with slightly higher burnup and shorter cooling time. The fuel in the outer cells provides shielding for the fuel in the inner cells.

Prior to placing the fuel in the STC, the fuel assemblies are classified as to the level of reactivity based on the initial enrichment and burnup. This classification is made using Figure 3.7.18-1, "Fuel Assembly Classification for Transfer in the Shielded Transfer Canister". This classification is used to determine if the fuel assembly may be placed in the STC and where it can be placed.

Figure 3.7.18-1 is used to classify each assembly into one of the following categories based on initial U-235 enrichment and average assembly burnup:

Type 2 assemblies are relatively less reactive assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled "Type 2 fuel" in Figure 3.7.18-1. Type 2 assemblies may be stored in any cell in the STC.

Additional constraints on Type 2 fuel are:

1) Post-irradiation cooling time ≥ 5 years
2) Average Burnup Per Assembly ≤ 55,000 MWD/MTU
3) Decay Heat ≤ 650 Watts may be placed in any STC cell
4) Decay Heat ≤ 1105 Watts may only be placed in inner cells (1, 2, 3, and 4) Type 1 assemblies are relatively more reactive assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the

(continued)

BACKGROUND (continued)

domain labeled "Type 1 fuel". Type 1 fuel must be placed in domain labeled "Type 1 fuel". Type 1 fuel must be placed in the outer cells (5, 6, 7, 8, 9, 10, 11, or 12) of the STC to preserve the assumptions of the design basis criticality analysis.

Additional constraints on Type 1 fuel are:

1) Post-irradiation cooling time ≥ 5 years
2) Average Burnup Per Assembly ≤ 55,000 MWD/MTU
3) Decay Heat ≤ 650 Watts

LCO 3.7.18.b is modified by a note stating that if one or more Type 1 assemblies are in the STC, cells 1, 2, 3, and 4 must be empty. The restriction preserves the assumptions of the bounding criticality analysis for Type 1 fuel assemblies placed only in the peripheral fuel cells.

Together, the limits on Type 1 and Type 2 ensure the criticality, shielding, and thermal analyses remain bounding.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.18-1 and cannot be placed in the STC in accordance with paragraph 4.4.1 in Section 4.4, Shielded Transfer Canister.

APPLICABLE SAFETY ANALYSES

The water in the STC will contain soluble boron at the same levels of the spent fuel pit, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the STC is based on the use of unborated water, which maintains the STC in a subcritical condition during normal operation with the STC fully loaded and in conformance with the fuel storage locations, enrichment and burnup assumed in the analysis and as specified by this LCO.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions because only a single accident need be considered at one time. For example, the accident scenarios include dropping a fuel assembly on top of the STC basket, or accidental misloading of a fuel assembly in STC basket.

These events could increase the potential for criticality in the STC. To mitigate these postulated criticality related accidents, boron concentration is verified to be within the limits specified in LCO 3.7.15, "Spent Fuel Pit Boron Concentration" every 31 days. Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1 (i.e., verification that the spent fuel pit boron concentration is within the limit). The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (References 2, 3, and 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pit (controlled by LCO 3.7.15, "Spent Fuel Pit Boron Concentration") prevents criticality in the STC. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the STC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Fuel assemblies stored in the spent fuel pit are classified in accordance with Figure 3.7.18-1 based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Based on this classification, fuel assembly placement in the STC cells is restricted in accordance with the classification of the fuel and the additional constraints established by this LCO.

(continued)

BASES

LCO
(continued)

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.18-1 because fuel assemblies with this enrichment cannot be placed in the STC in accordance with limits established in Technical Specification Section 4.4 Shielded Transfer Canister.

APPLICABILITY

This LCO applies whenever any fuel assembly is placed in the STC.

ACTIONS

A.1

When the configuration of fuel assemblies in the STC is not in accordance with this LCO, one action is to make the necessary fuel assembly movement(s) to bring the configuration of the fuel in the STC into compliance with this LCO. This action restores the STC to an analyzed configuration.

OR

A.2

When the configuration of fuel assemblies in the STC is not in accordance with this LCO, an optional action to restore compliance with the LCO is to move the fuel assembly or assemblies from the STC back into the IP3 spent fuel pool in accordance with LCO 3.7.16.

Either action places the fuel in equally safe locations.

The completion time of "Immediately" is appropriate because fuel located in the STC may be in an unanalyzed condition and action is required to be initiated and completed without delay to restore the fuel location to an analyzed configuration.

Required Actions A.1.1 and A.1.2 are modified by a Note indicating that LCO 3.0.3 does not apply. This is appropriate because inter-unit transfer and storage of spent fuel has no impact on safe plant reactor or power generation operations. Shutting down the plant due to non-compliance with LCO 3.7.18 would have no safety benefit.

(continued)

BASES

SURVEILLANCE

SR 3.7.18.1

This SR verifies by administrative means that the fuel assembly meets the requirements of the STC location in which it is to be placed in accordance with the accompanying LCO. This SR ensures the LCO limits for fuel selection and location in the STC are met and the supporting technical analyses remain bounding for all inter-unit transfer operations.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2)and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997.

3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

4. Holtec Report HI-2094289, Licensing report on the Inter-Unit Transfer of Spent Nuclear Fuel at Indian Point Energy Center, Revision 1.

NO SIGNIFICANT HAZARDS CONSIDERATION FOR INDIAN POINT UNIT 2 PERTAINING TO THE INTER-UNIT TRANSFER OF SPENT FUEL FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

Entergy Nuclear Operations, Inc.
Indian Point Unit 2
Docket No. 50-247

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment for IP2 by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed amendment proposes to modify Operating License (OL) Condition 2.B.(5), to add a new OL Condition 2.P to allow the possession of IP3 fuel in the IP2 spent fuel pit, and adds a new Technical Specification 3.7.15 *Shielded Transfer Canister (STC) Unloading*. The purpose of the LCO and associated Action and Surveillance Requirement is to place controls on the unloading of the STC within the IP2 spent fuel pit to ensure compliance with accident analysis assumptions and to take immediate action to restore compliance should the LCO not be met. The proposed amendment also adds a new Technical Specification Design Feature 4.4 *Shielded Transfer Canister* that ensures the design of the STC is in compliance with the criticality analysis, prevents inadvertent drainage of the STC, and restricts the maximum capacity of the STC to 12 fuel assemblies.

The previously evaluated accidents that must be considered are a criticality accident, a boron dilution accident in the spent fuel pit, a fuel handling accident (FHA), and a loss of spent fuel pit cooling.

The probability of a criticality accident is not increased because the proposed LCO requires that the IP3 spent fuel assemblies transferred to IP2 via the STC must be either in an approved IP2 spent fuel pit storage rack location per existing LCO 3.7.13 *Spent Fuel Pit Storage*, in their authorized STC fuel basket cell, or be in transit between these two locations. Administrative controls will be established to assure fuel movement under the requirements of this LCO are in compliance with the criticality analyses described in this amendment request, and, therefore, these fuel movements do not increase the probability of a criticality accident. The probability for a criticality event due to neutronic coupling, while the STC is in the spent fuel pit, is negligible because of the STC steel and lead vessel walls and the minimum distance between the STC and the nearest spent fuel rack will be maintained by procedure.

The probability of a boron dilution event remains the same because the proposed change does not alter the manner in which the IP2 spent fuel cooling system or any other plant system is operated, or otherwise increase the likelihood of adding significant quantities of unborated water into the spent fuel pit. In addition, because the STC is loaded in the IP3 spent fuel pit, the boron concentration in the STC is controlled by the IP3 Technical Specifications and UFSAR requirements that impose minimum boron concentration limits of 1000 ppm and 2000 ppm, respectively. The IP3 spent fuel pit boron concentration is controlled to the minimum requirements of the UFSAR and, therefore, the boron concentration in the STC will meet the IP2 minimum TS requirement of 2000 ppm. In addition, should the boron concentration in the STC be only 1000 ppm then the boron dilution accident at IP2 continues to be met as the boron dilution accident required minimum concentration is less than1000 ppm.
The probability of an FHA which includes a fuel assembly drop accident and a fuel cask drop accident remains very remote because of the many administrative controls and physical limitations imposed on fuel-handling operations. The probability of an FHA will not increase significantly due to the proposed changes because the individual fuel assemblies will be moved

from the STC to the spent fuel pit racks in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used. The fuel basket design in the STC emulates that of a spent fuel rack, making the individual fuel assembly movement operation the same as in current practice.

The probability of an STC drop in the truck bay, during transit to the spent fuel pit, or into the spent fuel pit is highly improbable because of the many administrative controls and physical limitations imposed on cask-handling operations and will not increase significantly due to the proposed changes because the cask will be moved into, and removed from, the spent fuel pit using the existing single failure proof lifting system and procedures similar to those already in use for dry cask storage activities. The lifting system includes the IP2 cask handling crane, special lifting devices, and interfacing lift points on the STC and HI-TRAC. The vertical drop of a HI-TRAC 100D containing a STC is considered sufficiently unlikely as not to increase the overall probability of a FHA due to the short period of time when suspended from the VCT after the redundant drop protection design features are removed.

The probability of a loss of spent fuel pit cooling remains the same because the proposed change does not alter the manner in which the IP2 spent fuel cooling loop is operated, designed or maintained.

The consequences of a criticality accident within the spent fuel pit are not changed because the fuel assemblies are spaced in a pattern within the STC and pit that continue to prevent any possibility of a criticality accident.

The consequences of the boron dilution event remains as described in the IP2 UFSAR because the minimum concentration of soluble boron in the spent fuel pit required by the IP2 Technical Specifications remains the same (2,000 ppm). The reactivity of the STC filled with the most reactive combination of approved fuel assemblies in unborated water results in a k_{eff} less than 0.95. Thus, even in the unlikely event of a complete dilution of the spent fuel pit water, the STC will remain safely subcritical.

The consequences of the existing fuel handling accident remain bounding because the IP3 fuel assembly design is essentially the same as the IP2 design and the IP3 fuel assemblies to be transferred to IP2 will be cooled a minimum of 5 years. This compares with a cooling time of 84 hours used in the existing FHA radiological analysis. The 5-year cooling time results in a significant reduction in the radioactive source term available for release from a damaged fuel assembly compared to the source term considered in the design basis FHA radiological analysis. The consequences of the previously analyzed fuel assembly drop accident, therefore, continue to provide a bounding estimated offsite dose for this accident. In addition, the consequences of a dropped fuel assembly on the STC has been evaluated from a criticality perspective and it has been determined that the fuel storage array remains subcritical.

The consequences of the existing fuel cask drop accident inside the FSB as described in the UFSAR remain bounding because the STC and HI-TRAC will only be lifted using a single failure proof lifting system inside the FSB, as described above. A drop of the HI-TRAC containing a fuel-loaded STC while suspended from the VCT was analyzed for the short period of time outdoors when the VCT does not have its redundant drop protection design features installed. The physical design of the STC and HI-TRAC 100D effectively precludes significant effluent releases as demonstrated by analysis of the HI-TRAC 100D vertical drop accident that demonstrates that the drop event would not result in any unacceptable consequence for the transfer cask and its contents.

The consequences of a loss of spent fuel pit cooling remains the same because the thermal design basis for the spent fuel pit cooling loop provides for all fuel pit rack locations to be filled at the end of a full core discharge and therefore the design basis heat load effectively includes any heat load associated with the assemblies within the STC.

In addition to the above, the consequences of a seismic event with an STC in the spent fuel pit, with the STC loaded into the HI-TRAC 100D, and with the HI-TRAC loaded into the vertical cask transporter have been evaluated and it has been determined that these components will not tip over during a design basis earthquake. Therefore, there are no significant dose consequences associated with this event.

The consequences of extreme environmental events such as high winds, tornado, and tornado –borne missiles have also been considered and in fact are part of the certification of HI-TRAC 100D in Docket No. 72-1014, where it is shown that the fuel is protected. In addition, should the water jacket of the HI-TRAC 100D be penetrated by a missile and all jacket water lost, analyses have demonstrated that no design limits would be exceeded. Therefore, there are no significant dose consequences associated with these events. A lightning strike could ignite the fuel in the vertical cask transporter resulting in engulfment of the HI-TRAC 100D, however, analyses have demonstrated that no design limits are exceeded. Therefore, there are no significant dose consequences associated with this event. The design basis flood event results in a water level below the elevation of the heavy haul path, therefore, there are no significant dose consequences associated with this event.

Finally, the consequences of an inadvertent loading and subsequent transport of a high decay heat assembly in the STC is precluded by robust fuel handling procedures and safety features. In addition there will be a procedural requirement to take water temperature readings at the top of each fuel assembly using a remote indicating temperature measuring device prior to STC lid closure. The breakdown of a loaded vertical cask transporter leaving HI-TRAC 100D exposed to potential high ambient temperatures for a prolonged period of time has been analyzed with all design criteria satisfied. Therefore, there are no significant dose consequences associated with these events.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The IP2 spent fuel pit is designed to accommodate fuel having certain characteristics as well as an overall heat load that is commensurate with the capability of the spent fuel pit cooling system's ability to remove the decay heat. The IP3 fuel assemblies selected for storage in the IP2 spent fuel pit are required to meet the existing IP2 Technical Specification LCO 3.7.13 *Spent Fuel Pit Storage* limits for storage. The fuel basket design in the STC emulates that of a spent fuel rack, making the individual fuel assembly movement operation the same as in current practice using appropriate procedural controls to ensure each fuel assembly is moved safely and into the correct fuel cell location in the racks as specified on the fuel move sheet . There are no changes proposed to the spent fuel pit storage racks or the manner in which spent fuel is moved from the reactor to the spent fuel pit. All limits for spent fuel storage in the IP2 spent fuel pit will remain the same and will continue to be complied with, including consideration of the additional IP3 fuel assemblies. The vertical drop of a HI-TRAC 100D containing a STC when suspended from the VCT after the redundant drop protection design features are removed has been evaluated and is bounded by the existing FHA. Thus, the existing fuel handling accident bounds the STC unloading operation and a HI-TRAC vertical drop and, therefore, no new or different fuel handling accident is created.

The proposed amendment does not alter the operation of any plant cooling systems nor does it create a new source of unborated water that could be injected into the spent fuel pit or affect the ability of systems to mitigate a boron dilution event. The STC has been designed and its contents so limited that even if fully de-borated, the content would not reach criticality, with k_{eff} remaining less than 0.95. Administrative controls will be used to ensure the boron concentration of the water in the STC matches that of the IP2 spent fuel pit.

The collapse of the roadway during transport resulting in cask rollover and potential loss of water inventory and potential large radioactive release is not a credible accident because of the structural stability of the VCT and the measures taken to ensure that the haul path is structurally sound. These measures include an evaluation of the haul path that includes ground penetrating radar and/or soil compaction studies.

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

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

Response: No.

The proposed amendment ensures that the IP3 spent fuel to be transferred to the IP2 spent fuel pit racks will meet all existing requirements for fuel storage at IP2. The storage racks and the spent fuel pit cooling system meet the design requirements for the IP3 fuel from a criticality, thermal, shielding, and material perspective because the fuel designs and operating parameters are very similar. The proposed STC loading TS 3.7.18 *Spent Fuel Assembly Transfer* and the associated fuel selection procedures used to control the loading of the STC at IP3 will ensure that all fuel to be transferred meets IP2 requirements for storage in the IP2 spent fuel pit racks.

The presence of the STC in the IP2 spent fuel pit has been analyzed and found to be acceptable. The spent fuel pit was originally designed to accommodate a spent fuel shipping cask and the STC is similar in physical dimensions to a shipping cask. Administrative controls will be in place during STC handling operations to ensure that the spent fuel pit water level is maintained within limits. Any criticality event caused by neutronic coupling between the STC and the fuel in the spent fuel racks is precluded due to the thick steel and lead walls of the STC and the distance between the STC and the nearest spent fuel in the racks.

The proposed amendment does not involve changes to any plant operating systems used to cool the spent fuel pit water or respond to unanticipated events or accidents.

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

Based on the above, Entergy concludes that the proposed amendment to the IP2 Operating License and Technical Specifications presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

NO SIGNIFICANT HAZARDS CONSIDERATION FOR INDIAN POINT UNIT 3 PERTAINING TO THE INTER-UNIT TRANSFER OF SPENT FUEL FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

Entergy Nuclear Operations, Inc.
Indian Point Unit 3
Docket No. 50-286

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment for IP3 by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed amendment adds a new Technical Specification 3.7.18 *Spent Fuel Assembly Transfer*. The purpose of the LCO and associated Action and Surveillance Requirement is to place controls on the loading of the STC within the IP3 spent fuel pit to ensure compliance with accident analysis assumptions and to take immediate action to restore compliance should the LCO not be met. The proposed amendment also adds a new Technical Specification Design Feature 4.4 *Shielded Transfer Canister* that ensures the design of the STC is in compliance with the criticality analysis, prevents inadvertent drainage of the STC, and restricts the maximum capacity of the STC to 12 fuel assemblies.

The previously evaluated accidents that must be considered are a criticality accident, a boron dilution accident in the spent fuel pit, a fuel handling accident (FHA), and a loss of spent fuel pit cooling.

The probability of a criticality accident is not increased because the proposed LCO requires that the IP3 spent fuel assemblies transferred to the STC must be in an approved STC cell location. Administrative controls will be established to ensure fuel movement under the requirements of this LCO are in compliance with the criticality analyses described in this amendment request and, therefore, these fuel movements do not increase the probability of a criticality accident. The probability for a criticality event due to neutronic coupling while the STC is in the spent fuel pit is negligible because of the STC steel and lead vessel walls and the minimum distance between the STC and the nearest spent fuel rack will be maintained by procedure.

The probability of a boron dilution event remains the same because the proposed change does not alter the manner in which the IP3 spent fuel cooling system or any other plant system is operated or otherwise increase the likelihood of adding significant quantities of unborated water into the spent fuel pit.

The probability of an FHA which includes a fuel assembly drop accident and a fuel cask drop accident remain remote because of the many administrative controls and physical limitations imposed on fuel-handling operations and will not increase significantly due to the proposed changes because the individual fuel assemblies will be moved from the spent fuel pit racks to the STC in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used. The fuel basket design in the STC emulates that of a spent fuel rack, making the individual fuel assembly movement operation the same as in current practice.

The probability of an STC drop in the truck bay, into the spent fuel pit, or in transit between these two locations is highly improbable because of the many administrative controls and physical limitations imposed on cask-handling operations and will not increase significantly due to the proposed changes because the STC will be moved into, and removed from, the spent fuel pit using a single failure proof lifting system and procedures similar to those already in use for dry cask storage activities. A single failure proof lifting system includes the 40-ton IP3 upgraded cask handling crane, special lifting devices, and interfacing lift points on the STC. The vertical drop of a HI-TRAC 100D containing a STC is considered sufficiently unlikely as not to increase the overall probability of a FHA due to the short period of time when suspended from the VCT before the redundant drop protection design features are installed.

The probability of a loss of spent fuel pit cooling remains the same because the proposed change does not alter the manner in which the IP3 spent fuel cooling loop is operated, designed or maintained.

The consequences of a criticality accident within the spent fuel pit are not changed because the fuel assemblies are spaced in a pattern within the STC and pit that continue to prevent any possibility of a criticality accident.

The consequences of the boron dilution event remain unchanged because the minimum concentration of soluble boron in the spent fuel pit required by the IP3 Technical Specifications remains unchanged (1,000 ppm). The reactivity of the STC filled with the most reactive combination of approved fuel assemblies in unborated water results in a k_{eff} less than 0.95. Thus, even in the unlikely event of a complete dilution of the spent fuel pit water, the STC will remain safely subcritical.

The consequences of the existing fuel handling accident remain bounding because only IP3 fuel is being moved in the IP3 spent fuel pit. Moreover, the IP3 fuel assemblies to be transferred to the STC must have a post-irradiation cooling time of greater or equal to 5 years. This compares with a cooling time of 84 hours used in the existing FHA radiological analysis. The 5-year cooling time results in a significant reduction in the radioactive source term available for release from a damaged fuel assembly compared to the source term considered in the design basis FHA radiological analysis. The consequences of the previously analyzed fuel assembly drop accident, therefore, continue to provide a bounding estimated offsite dose for this accident. In addition, the consequences of a dropped fuel assembly on the STC has been evaluated from a criticality perspective and it has been determined that the fuel storage array remains subcritical.

The consequences of the existing fuel cask drop accident inside the FSB as described in the UFSAR remain bounding because the STC will only be lifted using a single failure proof lifting system inside the FSB, as described above. A drop of the HI-TRAC containing a fuel-loaded STC while suspended from the VCT was analyzed for the short period of time outdoors when the VCT does not have its redundant drop protection design features installed. The physical design of the STC and HI-TRAC 100D effectively precludes significant effluent releases as determined by the analysis of the HI-TRAC 100D vertical drop accident that demonstrates that the drop event would not result in any unacceptable consequence for the transfer cask and its contents.

The consequences of a loss of spent fuel pit cooling remains the same because the thermal design basis for the spent fuel pit cooling loop provides for all fuel pit rack locations to be filled at the end of a full core discharge and therefore the design basis heat load effectively includes any heat load associated with the assemblies within the STC.

The consequences of a seismic event with an STC in the spent fuel pit, with the STC loaded into the HI-TRAC 100D, and with the HI-TRAC loaded into the vertical cask transporter have been evaluated and it has been determined that these components will not tip over during a design basis earthquake. Therefore, there are no significant dose consequences associated with this event.

The consequences of extreme environmental events such as high winds, tornado, and tornado –borne missiles have also been considered and in fact are part of the certification of HI-TRAC 100D in Docket No. 72-1014, where it is shown that the fuel is protected. In addition, should the water jacket of the HI-TRAC 100D be penetrated by a missile and all jacket water lost, analyses have demonstrated that no design limits would be exceeded. Therefore, there are no significant dose consequences associated with these events. A lightning strike could ignite the fuel in the vertical cask transporter resulting in engulfment of the HI-TRAC 100D, however, analyses have demonstrated that no design limits are exceeded. Therefore, there are no significant dose consequences associated with this event. The design basis flood event results in a water level below the elevation of the heavy haul path, therefore, there are no significant dose consequences associated with this event.

Finally, the consequences of an inadvertent loading and subsequent transport of a high decay heat assembly in the STC is precluded by robust fuel handling procedures and safety features. In addition there will be a procedural requirement to take water temperature readings at to the top of each fuel assembly using a remote indicating temperature measuring device prior to STC lid closure. The breakdown of a loaded vertical cask transporter leaving HI-TRAC 100D exposed to potential high ambient temperatures for a prolonged period of time has been analyzed with all design criteria satisfied. Therefore, there are no significant dose consequences associated with these events.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The IP3 fuel assemblies selected for loading into the STC in the IP3 spent fuel pit are required to meet the proposed new Technical Specification 3.7.18 *Spent Fuel Assembly Transfer.* The fuel basket design in the STC emulates that of a spent fuel rack, making the individual fuel assembly movement operation the same as in current practice using appropriate procedural controls to ensure each fuel assembly is moved safely and into the correct STC cell location as specified on the fuel move sheet . There are no changes proposed to the spent fuel pit storage racks or the manner in which spent fuel is moved from the reactor to the spent fuel pit. All limits for spent fuel storage in the IP3 spent fuel pit will remain the same and will continue to be complied with. The vertical drop of a HI-TRAC 100D containing a STC when suspended from the VCT prior to the redundant drop protection design features being installed has been evaluated and is bounded by the existing FHA. Thus, the existing fuel handling accident bounds the STC unloading operation and a HI-TRAC vertical drop and, therefore, no new or different fuel handling accident is created.

The transfer of certain IP3 spent fuel assemblies to IP2 does not adversely alter the design, analysis, or operation of the IP3 spent fuel pit. This amendment involves the removal of IP3 fuel from the IP3 spent fuel pit as part of needed fuel management and to maintain fuel core offload capability in the IP3 pit. No changes to the spent fuel pit capacity or cooling system design are involved.

The proposed amendment does not alter the operation of any plant cooling systems nor does it create a new source of unborated water that could be injected into the spent fuel pit or affect the ability of systems to mitigate a boron dilution event. The STC has been designed and its contents so limited that even if fully de-borated, the content would not reach criticality, with k_{eff} remaining less than 0.95. Administrative controls will be used to ensure the boron concentration of the water in the STC matches that of the IP2 spent fuel pit.

The collapse of the roadway during transport resulting in cask rollover and potential loss of water inventory and potential large radioactive release is not a credible accident because of the structural stability of the VCT and the measures taken to ensure that the haul path is structurally sound. These measures include an evaluation of the haul path that includes ground penetrating radar and/or soil compaction studies.

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

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

Response: No.

The proposed STC loading TS 3.7.18 *Spent Fuel Assembly Transfer* and the associated fuel selection procedures used to control the loading of the STC at IP3 will ensure that all fuel to be loaded into the STC meets the requirements of the accident analyses.

The presence of the STC in the IP3 spent fuel pit has been analyzed and found to be acceptable. The spent fuel pit was originally designed to accommodate a spent fuel shipping cask and the STC is similar in physical dimensions to a shipping cask. Administrative controls will be in place during STC handling operations to ensure that the spent fuel pit water level is maintained within limits. Any criticality event caused by neutronic coupling between the STC and the fuel in the spent fuel racks is precluded due to the thick steel and lead walls of the STC and the distance between the STC and the nearest spent fuel in the racks.

The proposed amendment does not involve changes to any plant operating systems used to cool the spent fuel pit water or respond to unanticipated events or accidents.

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

Based on the above, Entergy concludes that the proposed amendment to the IP3 Technical Specifications presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ATTACHMENT 6 TO NL-09-076

NEW COMMITMENTS
PERTAINING TO
THE INTER-UNIT TRANSFER OF SPENT FUEL
FROM INDIAN POINT UNIT 3 TO INDIAN POINT UNIT 2

Entergy Nuclear Operations, Inc.
Indian Point Units 2 and 3
Docket Nos. 50-247 and 50-286

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE
1	The heavy haul path will be entirely within the IPEC owner protected area (PA). In order to achieve this objective the PA boundary will need to be modified. These modifications will be performed under a separate project and scheduled for completion at a time commensurate with the need to transfer fuel from IP3 to IP2. Entergy commits to moving the PA boundary as described prior to the first inter-unit transfer	Prior to the first inter-unit transfer of fuel
2	The IP3 crane will be upgraded to a single failure proof crane meeting the intent of NUREG-0554 through the use of ASME NOG-1-2004 as the governing design code.	Prior to the first inter-unit transfer of fuel
3	The transport roadway and any buried safety-related components beneath it that could be affected by the inter-unit transfer of fuel will be shown to have acceptable structural capacity including the consideration of underground utilities. Entergy commits to performing this evaluation and implementing any modifications determined to be necessary prior to the first inter-unit transfer of fuel.	Prior to the first inter-unit transfer of fuel