REQUEST FOR ADDITIONAL INFORMATION

REGARDING SPENT FUEL TRANSFER

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

By letter dated July 8, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091940177 and ML091940178), and supplemented by letters dated September 28, 2009, (ADAMS Accession Nos. ML092950437 and ML093020080), and October 5, 2010 (ADAMS Accession Nos. ML102910511, ML103080112, and ML103080113) Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a license amendment request for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3). The proposed changes are requested to provide the necessary controls and permission required for Entergy to move spent fuel from the IP3 spent fuel pool (SFP) to the IP2 SFP using a newly designed shielded transfer canister (STC), which is placed inside a HI-TRAC 100D cask for outdoor transport. The chapters listed below refer to the safety analysis report (SAR) for the STC, HI-2094289, Revision 3, ADAMS Accession No. ML103080113. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions.

The following provides the NRC questions together with Entergy's responses.

CHAPTER 1 – GENERAL INFORMATION

NRC RAI 1-1

Attachments 5 and 7 to your letter dated October 5, 2010, included a proposed Appendix C to the unit operating licenses, with technical specifications (TSs) for the spent fuel transfer operations. Proposed TS 3.1.3, "Shielded Transfer Canister (STC) Pressure Rise," included a limiting condition for operation (LCO) of less than a 4.2 psi pressure increase over 24 hours. The associated proposed TS Bases were provided in Attachments 6 and 8, and the bases indicated that the LCO ensures that fuel assemblies selected for loading in the STC satisfy design basis limits because an analysis of the pressure rise for the design heat load of 9.6 kW was less than 4.2 psi. However, the licensing report (Enclosure 1 to letter dated October 5, 2010) includes additional analyses verifying that the STC design pressure would not be exceeded for various postulated accident conditions. (SBPB)

Paragraph (c) (2) of Title 10 of the Code of Federal Regulations (10CFR) 50.36, "Technical Specifications," requires in part that a LCO be established for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In addition, Paragraph (c)(3) of 10 CFR 50.36 requires the inclusion of Surveillance Requirements (SRs) related to test or inspection that assure that the LCOs will be met. Finally, Paragraph (c)(4) of 10 CFR 50.36 requires the inclusion in the TSs of design features of the facility that, if altered or modified, would have a significant effect on safety.

The NRC staff considers the STC pressure boundary, which is identified as a confinement boundary, to be a fission product boundary. The licensing design-basis analyses examine the

effects of several postulated events with respect to challenges to the integrity of the fuel cladding and the STC pressure boundary. Accordingly, propose additional LCOs, SRs, and Design Feature TSs necessary to satisfy the requirements of 10 CFR 50.36.

For example, address methods to verify that appropriate initial conditions have been established for STC overpressure protection (e.g., a steam bubble of appropriate volume exists within the STC). To ensure this steam space is formed, additional controls are needed. Proposed TS 4.1.4.6 should be converted to a TS SR and LCO. This SR would verify that the required volume of water is removed through the STC drain line following the application of steam pressure. Also, in the absence of an analysis showing the acceptability of an air-filled space rather than a steam-filled space, a TS SR needs to demonstrate that the space is filled with steam. One method would be to use the pressure change in the STC during the 24 hours following establishment of the steam space. LCO 3.1.3 could be modified to show that the pressure stays within an analyzed range over time (e.g., a response graph). The lower limit of the graph could be the zero heat load curve, and the upper limit could be the response with the design heat load. See RAI TS-8 for additional comments on LCO 3.1.3.

Design features essential for the assumed heat transfer capabilities necessary to prevent overpressure conditions (e.g., materials of construction [for thermal conductivity] and emissivity of outer surface) need to be described in TS. For example, TS section 4.1 could be modified to add a section (after Criticality) titled "4.1.3 Thermal Features." The type of thermal features described here would be those critical to the heat transfer abilities of the STC and the HI-TRAC.

This information is needed to confirm compliance with 10 CFR 50.36.

Response to RAI 1-1

The proposed LCOs and SRs have been revised to satisfy the requirements of 10 CFR 50.36. The STC pressure boundary is a confinement boundary and therefore, by definition, a fission product boundary. The accident analyses assume that the STC initial water level has been established, that the STC void space is filled with steam, and that the STC contains a design basis heat load. Therefore, LCOs are proposed that establish these design basis conditions prior to transfer operations.

<u>STC Initial Water Level</u>

The originally proposed TS 4.1.4.6 has been converted to TS LCO 3.1.3 STC Initial Water Level and SR 3.1.3.1. The SR verifies that steam is emitted through the STC drain line and that the required volume of water is removed following the application of steam pressure.

<u>Demonstration that the STC void space is filled with steam and detection of a severe fuel misload</u>

As suggested in the RAI one method to demonstrate that the STC void space is filled with steam would be to use the pressure change in the STC during the 24 hours following establishment of the steam space wherein an LCO and SR could be developed to show that the pressure stays within an analyzed range over time. The lower limit of the graph could be the zero heat load curve, and the upper limit could be the response with the design heat load. RAI TS-8 also suggests that 25 data points be taken over a 24 hour period to allow more accurate checking of the pressure change.

To use upper and lower bound pressure or pressure rise measurements on an hourly basis as an LCO requirement for fuel transfer would be problematic for a number of reasons. The predicted thermodynamic response of the loaded STC is a complex issue that is a function of the accuracy of the thermodynamic model, a number of variables, and boundary conditions. These include, but are not limited to, pool water temperature, ambient air temperature, length of time required to load fuel in the STC, length of time required to transfer the loaded STC from the Spent Fuel Pool to the HI-TRAC. Each one of these variables would need to be measured and recorded in order to determine the proper pressure or pressure rise curve to use to confirm that the heat load of the fuel is below the design basis limit and that the STC void space is filled with steam.

Table 1 below shows how the pressure rise inside the STC will change as a function of a variation in the initial temperature in the STC. There are multiple other variables that also can affect the STC pressure or pressure rise, in both a positive and negative direction.

<u>Table 1: 24 Hour Pressure Rise for Design Basis Heat Load, 100°F Ambient Temperature</u>

[PROPRIETARY TEXT REMOVED]

Given the uncertainties in the thermal model predictions and pressure readings, it is not practical to develop an LCO and associated SR that would require the comparison of predicted and measured pressure or pressure rise on an hourly basis. Such a comparison could lead to both unwarranted fuel handling and transfer delays.

Based on the above discussion it is also not practical to attempt to distinguish between a significant misload and significant amounts of air in the STC steam space as both conditions result in an STC pressure rise greater than that predicted by the thermal analysis for design basis conditions. Therefore, an LCO and associated SR are proposed that would detect either of these conditions and require specific actions to be taken in the event that the rate of STC pressure rise, over a rolling four hour period, exceeds the specified rate of change. A rolling 4 hour period, with measurements taken hourly for a total of 25 data points over a 24 hour period, was selected to ensure that uncertainties in the thermal analysis and pressure readings do not result in unwarranted actions and also to ensure that there is sufficient time available to mitigate either condition (significant fuel misload or significant amount of air).

As documented in Chapter 5 of the Licensing Report, the 48 hour STC pressure rise has been determined based on a design basis heat load and with steam in the STC void space (Figure 1 below). The STC pressure rise for a severe misload is compared to the design basis pressure rise in Figure 2 and likewise the rate of change of STC pressure are compared in Figure 3. This Figure shows that a rate of change of [PROPRIETARY TEXT REMOVED] can be used to differentiate between a design basis heat load and a severe misload shortly after commencement of the STC pressure rise surveillance. This criterion also serves to detect a significant amount of air inside the STC, since that condition would also substantially increase the rate of change of STC pressure.

A rate of STC pressure rise above [PROPRIETARY TEXT REMOVED] during any rolling 4 hour period is an indication that the design basis of the STC is not being met. This indication could represent one or more of the following conditions: not establishing the correct STC water level, misloading a fuel assembly, or a significant ingress of air into the STC. In this case the STC would be depressurized and actions taken to verify: the STC water level, that a fuel

misload had not occurred and that air had been effectively precluded from entering the STC. Once the cause of the non conforming condition is identified and corrective actions taken, the SR would need to be re-performed satisfactorily prior to STC transfer operations.

In support of the proposed LCO and SR it should be noted that the STC is designed for pressures well above those required to demonstrate compliance with this LCO. Calculations, as documented in Chapter 5 of the Licensing Report, have been performed to demonstrate that the STC pressure boundary can accommodate pressures of up to 90 psig, which is [PROPRIETARY TEXT REMOVED] more than the steady state pressure for design basis heat loads, and two times more than the steady state pressure for [PROPRIETARY TEXT REMOVED]. Therefore, although the LCO and SR will preclude it, the STC could be loaded with heat loads even greater [PROPRIETARY TEXT REMOVED]before the steady state pressure would exceed the pressure limits for the STC. In addition, once compliance with this LCO is established, it is judged that there are no other credible design basis accidents during the transfer which challenge the pressure boundary of STC. The design basis accidents are described in Chapter 5 of the licensing report and include a loss of jacket water, a fire, HI-TRAC annulus plus a jacket water loss, fuel misload, and a tip-over accident. Apart from the tip-over accident there are sufficient margins to include any minor misload not detectable via the proposed LCO and surveillance. While the tip-over accident has been analyzed, there is no credible mechanism for this to take place and it therefore considered to be a non-mechanistic event and demonstration of a misload coincident with a tipover is not considered credible.

[PROPRIETARY TEXT REMOVED].The STC pressure rise determined based on full STC design basis heat, pool water temperature of 120°F and a significant amount of air entrapment in the STC [PROPRIETARY TEXT REMOVED]. Therefore, the [PROPRIETARY TEXT REMOVED]criterion over a rolling 4 hour period ensures that a severe fuel misload or significant amounts of air in the STC would be detected by the pressure monitoring system within 8 hours.

[PROPRIETARY TEXT REMOVED]

Figure 1: 48-hour STC Pressure Rise under Design Basis Heat Load

[PROPRIETARY TEXT REMOVED]

Figure 2: Comparison of STC Pressure Rise between Design Basis Heat Load and Severe Fuel Misload Accident

[PROPRIETARY TEXT REMOVED]

Figure 3: Comparison of Rate of Change of STC Pressure with Time between Design Basis Heat Load and Severe Fuel Misload Accident

The RAI also requests that "thermal features" be added to the TS. The thermal model is based on the nominal dimensions provided in the licensing report and explicitly considers all the features (such as gaps between basket and the STC shell) that affect the thermal-hydraulic behavior of the STC system. Importance of individual design features of the STC in determining the thermal-hydraulic behavior of the STC cannot be quantified and measured. In order to conservatively predict the temperature/pressure in the STC and HI-TRAC, an array of conservatism assumptions have been used in the analyses (see section 5.3.1 of the LAR).

Conservative material properties from valid references have been used in the analyses. No special treatment of the materials is required to ensure their thermal capabilities or meet the values assumed. Also note that the following statement has been added to the TS Bases B 3.1.4, "...24 hour STC pressure test will also be a check on the thermal properties of the STC and HI-TRAC system, such that if there were any significant changes in the thermal properties it *would be expected to show as anomalous readings during the pressure test." Therefore, further* controls in the TS on the thermal features are not needed.

NRC RAI 1-2

Modify licensing drawing No. 6013, Sheet 2 of 4, to state that the lead thickness shown is a minimum thickness. (CSDAB)

The lead thickness for the STC in the referenced drawing is the minimum thickness allowed per the proposed TS, Appendix C, Part I, Section 1.0 description of the STC; thus, the drawing's specification of the lead thickness should be consistent with that description and indicate that the stated dimension in the drawing is a minimum value.

This information is needed to confirm compliance with 10 CFR Part 50, Appendix B, and the intent of 10 CFR 72.104, 72.106 and 72.44(a).

Response to RAI 1-2

Drawing 6013 (Sheet 2 of 4) has been revised to state that the lead thickness shown is a minimum thickness (Section 1.5 in the Licensing Report).

CHAPTER 4 - CRITICALITY EVALUATION (CSDAB and SRXB)

General Response to Criticality RAIs

Before the detailed responses to the individual RAIs are presented, this introductory section gives an overview of the criticality methodologies, regulatory bases, applicable acceptance criteria, embedded margins and conservatisms, and discusses general adjustments that were made as a result of the RAI responses.

Governing Regulation, B-10 Areal Density, Initial Methodology

The STC is a device that is to be licensed under 10CFR50, and its criticality safety performance and acceptance criteria are therefore governed by 10CFR50.68. The design of the STC basket is based on the specifications of the basket of the MPC-32 dry storage and transportation canister. The STC basket consists essentially of the 12 inner cells of an MPC-32 basket. From a criticality perspective, the STC has a larger neutron absorption capability than almost all other wet storage structures (racks) licensed under 10CFR50.68. Specifically, the B-10 areal density and steel wall thicknesses are substantially larger than those in the IP Unit 3 and Unit 2 Region 2 spent fuel racks (See Licensing Report Section 2.2.1 and Table 2.2.1). Based on this fact

alone, all fuel qualified for the Region 2 racks in the IP Unit 3 and Unit 2 pool would automatically qualify for loading into the STC. Specifically,

- The B-10 areal density of the neutron absorbers in the STC basket is 0.0310 g/cm^2 (minimum), which is higher than that of typical wet storage racks in the US. The IP2 and IP3 racks have neutron absorbers with nominal B-10 areal densities of 0.026 and 0.020 g/cm^2, respectively.
- The thickness of the wall of the steel basket is about 0.28 inch, while typical wet storage racks have thicknesses of about 0.075 inch. The wall thickness in the IP2 and IP3 racks are 0.075 and 0.170 inches, respectively.

Based on this situation, the initial licensing report for the STC utilized a typical wet storage methodology to demonstrate criticality safety, with soluble boron credit for accident conditions but not for normal conditions, and applied essentially the same burnup requirements (loading curves) that existed for the IP2 and IP3 Region 2 racks to the STC.

<u>Change to HI-STAR 100 Criticality Methodology, Retaining 10CFR50.68 Acceptance Criteria</u>

When responding to the first round of RAIs in 2010 for criticality, an additional aspect was taken into consideration. Over the last two years, the general criticality analysis methodology for wet storage systems has been undergoing an extensive review and revision. As of this writing, this process is not finalized, and new durable NRC guidance is only expected in early 2013. In the interim, a draft ISG (DSS-ISG-2010-01) was issued by the NRC in September 2010; however that also was not available when the previous RAIs were received in April 2010. Therefore, in the absence of a durable guidance, a methodology different from and more conservative than the typical Part 50 wet storage criticality methodology was used. This methodology is based on the burnup credit methodology developed for the HI-STAR 100 Transport cask, which was reviewed and approved by the NRC under 10CFR71 (ML062860201, October 12, 2006). The HI-STAR criticality methodology introduced additional margin and conservatisms, mainly in the following areas:

- In addition to the traditional critical experiment benchmarks used in wet storage, the HI-STAR 100 methodology uses benchmarks based on chemical assays and commercial reactor criticals. Application of those additional benchmarks limits the number of isotopes credited, applies highly conservative correction factors to minor actinides and fission products, and adds additional bias and bias uncertainties. While the Part 50 wet storage criticality methodology combines uncertainties statistically, including the depletion uncertainty, worst case combinations of tolerances are used in the HI-STAR 100 methodology.
- In the HI-STAR 100 methodology, assemblies with potential control rod insertion are conservatively analyzed as if control rods had been inserted to the fullest extent in those assemblies, although this condition is not permitted during full power operation. This approach maximizes the spectrum hardening in those assemblies and therefore increases reactivity.

It is important to note that using the Part 71 methodology vs. the Part 50 methodology is only a change in methodology, not the applicable regulations and acceptance criteria originally stated. The STC will still be licensed under 10CFR50, and the acceptance criteria from 10CFR50.68

still apply. This means that soluble boron credit is an option in limiting k-eff under normal conditions. 10CFR50.68 also permits credit for soluble boron to protect against the consequences of accidents, and this is applied in the analysis for the STC with respect to misloading conditions. A burnup measurement to protect against misloading, as suggested in ISG-8 Revision 2 as additional guidance to NUREG-1617 for applying burnup credit to transport packages licensed under10 CFR Part 71, is therefore not necessary for the STC. A more detailed comparison of the methodologies is presented below.

Application of the HI-STAR 100 Methodology to the STC

Most aspects of the burnup credit methodology for the STC are identical to those previously developed for the HI-STAR 100. However, there are some differences, either to account for site-specific conditions or in consideration of the different acceptance criteria (Part 71 vs. Part 50). This section provides a brief description of those differences, and provides or references the relevant discussions and justifications in each case.

The differences are presented in the following tables, which briefly describe the aspects, any changes made in the context of the current RAI responses, together with any discussion, justification, and references. The first table lists those aspects that were initially different (in response to the first round of RAIs), and how the STC approach is now aligned with the HI-STAR 100 approach in the context of the current RAI responses. The second table presents those aspects where a difference remains, either in the initial calculations, or as a result of the RAI responses.

Table 4.0.1- Resolved Methodology Differences HI-STAR 100 vs. STC

HI-STAR 100	**STC, First Submittal using HI-STAR 100 Methodology (2010)**	**STC, Revised Methodology for current RAI responses**	**Discussion / Justification / Reference**
Burnable Poison Rods assumed over entire active length	Burnable Poison Rods over part of active length based on actual burnable poison design used at IP	Now same as HI-STAR 100 – Burnable Poison Rods assumed over entire active length	Revised approach is more conservative. See Response to RAI 4-6
Burnup Credit *limited* to 50 GWd/mtU	Burnup credit *exceeds* 50 GWd/mtU	Now same as HI-STAR 100 - Burnup Credit *limited* to 50 GWd/mtU	Revised approach is more conservative. See Response to RAI 4-8. Note that this is a limit of the credited burnup, not a limit of the actual burnup of an assembly.

Table 4.0.2 - Retained and New Methodology Differences HI-STAR 100 vs. STC

HI-STAR 100	STC, First Submittal using HI-STAR 100 Methodology (2010)	STC, Revised Methodology for current RAI responses	Discussion / Justification / Reference
Upper bound specific power used in depletion analyses	Same as HI-STAR 100	Conservatively low specific power used in depletion calculations	Revised approach is more conservative. See Response to RAI 4-5
Radial Burnup Profiles evaluated, but not considered in design basis calculations	Same as HI-STAR 100	Radial Burnup Profiles now considered in design basis calculations	Revised approach is more conservative. See Response to RAI 4-7
No optional loading pattern for fresh fuel or fuel that does not meet the loading curves.	Optional partial loading pattern for fuel that does not meet the loading curves.	Unchanged	Needed for fuel that does not meet the minimum required burnup. Note that fresh fuel is in fact not permitted in the pool during STC loading.
Normal conditions analyzed with pure (unborated) water flooding.	Normal conditions analyzed with pure (unborated) water flooding,	Unchanged	Note that 10CFR50 regulations allows some soluble boron credit under normal conditions.
Burnup Measurement to protect against misloading	Soluble Boron Credit to protect against consequences of misloading accidents.	Unchanged	Standard approach for 10CFR50 based on double contingency principle. Soluble boron provides large subcriticality margin even under misloading conditions.
Criticality Benchmarks based on Fresh Fuel and MOX critical experiments	Criticality Benchmarks based on Fresh Fuel, MOX fuel and simulated spent fuel (HTC) critical experiment	Unchanged	Increased confidence in criticality calculations
No additional considerations for neutron absorber surveillance program	Additional 5% B-10 penalty for surveillance program	Unchanged	In consideration of the multiple loading/unloading cycles of the STC
For assemblies with potential control rod	As a bounding approach, only insertion	Unchanged	Bounding approach to simplify analyses.

HI-STAR 100	STC, First Submittal using HI-STAR 100 Methodology (2010)	STC, Revised Methodology for current RAI responses	Discussion / Justification / Reference
insertion, several durations of this insertions are considered	during the entire irradiation period is considered		

In summary, for the differences presented in Table 4.0.2, the approach taken for the STC provides the same or larger conservatism than that of the HI-STAR 100.

Recent Part 50 Wet Storage Criticality Approval

It is also considered beneficial here to highlight recent developments in the wet storage criticality safety analysis area: While there is no durable guidance yet, there are clear indications that successful and acceptable paths forward for criticality safety evaluations for wet storage systems are now available. Just recently, NRC approved the analysis for re-racking of the Beaver Valley Unit 2 Spent fuel pool, after a two year licensing process that involved extensive interaction with the NRC technical staff, including a week-long technical audit, and review of analyses by Oak Ridge National Laboratory experts (For the Beaver Valley SER see ML110890844). Beaver Valley is a Westinghouse plant with a similar fuel type and similar reactor conditions as Indian Point. It is noted that the burnup acceptance criteria developed and approved for Beaver Valley is much less restrictive than developed for the STC basket using the HI-STAR 100 methodology. The STC will therefore provide a higher criticality margin than the recently approved Beaver Valley racks.

Comparison of minimum required burnups

To indicate the additional level of conservatism in the HI-STAR 100 methodology, the table below lists the minimum required burnup for the fuel to be placed in the STC , in comparison with the minimum required burnup for fuel to be placed in the Indian Unit 2 and Unit 3 pools, and for the Beaver Valley Unit 2 pool. For this comparison, all fuel is 4.0% enriched. As discussed above, the Beaver Valley information is from a wet storage criticality license amendment request approved very recently and is for fuel that is similar to the Indian Point fuel.

Table 4.0.3 – Comparison of Minimum Required Burnup

Calculation	Minimum Required Burnup for fuel of 4 wt%- U-235 Enrichment [GWd/mtU]
Indian Point 3 Fuel in STC	
Wet Storage Methodology (Initial Submittal)	28.26
HI-STAR 100 Methodology, Fuel NOT exposed to control rods (First Submittal	33.1 / 35.1

with HI-STAR Methodology / Revised for RAI Responses)	
HI-STAR 100 Methodology, Fuel POTENTIALLY exposed to control rods (First Submittal with HI-STAR Methodology / Revised for RAI Responses)	40.5 / 42.5
Spent Fuel Pools	
Indian Point Unit 2	28.80
Indian Point Unit 3	29.75
Beaver Valley Unit 2 (ML110890844, Table 3.7.14-1E, Region 3)	28.84

The minimum required burnup for the STC using the HI-STAR methodology is higher than even the requirements for the recently approved Beaver Valley pool. Again, this additional conservatism to use the Part 71 HI-STAR 100 methodology was added intentionally to decouple the STC criticality analyses from the currently developing wet storage methodology.

Margin Analysis

The HI-STAR 100 methodology is based on ISG 8, Rev 2. This ISG suggests performing a margin analysis to allow for an evaluation of any remaining uncertainties (or modeling deficiencies) that are not explicitly considered in the design basis cases and compare against this margin. This margin analysis was performed for the HI-STAR 100, and also for the STC. For the STC, it indicates a potential margin on the order of 0.07 delta-k. With the revised loading requirements, there are essentially no uncertainties not explicitly considered in the analysis. The entire margin is therefore available to cover any unspecified uncertainties.

Indian Point Unit 3 Pool Inventory

The minimum required burnups are also viewed in the context of the actual inventory in the Indian Point Unit 3 pool that needs to be moved into the Unit 2 pool. The current requirements, with the substantial increase in minimum required burnups due to adopting the HI-STAR 100 methodology (see Table 4.0.3), are fairly limiting in terms of the 12 assembly loading configuration. Any further increase in the minimum required burnups would move a larger number of assemblies into an 8 assembly loading configuration, thereby increasing the number of transfers impacting both operations and ALARA considerations.

Soluble Boron Credit

There is one difference between the actual condition of the STC and the condition that is assumed for the HI-STAR 100 for the criticality analyses: The STC is filled with borated water with a minimum soluble boron concentration of 2000 ppm, as specified in the proposed Technical Specifications, whereas the HI-STAR 100 was analyzed with fresh water, as required by the regulations applicable to the HI-STAR 100 (Part 71).

This soluble boron concentration is equivalent to an additional reactivity margin of about 0.20 delta-k. The assumption of unborated water in the STC results in an additional and substantial margin to any limit, i.e. a k-eff of 0.9 in unborated water would be a k-eff of about 0.7 with this concentration of soluble boron.

The soluble boron level of the STC is controlled operationally through a proposed LCO in the TS, and since the system is designed and tested to be leak tight, there is no credible mechanism to reduce this soluble boron level during transfer operations.

Note that soluble boron, about 1000 ppm, is credited for various potential misloading conditions, as permitted by the regulation that governs the operation of the STC, 10CFR50.68. This regulation also permits credit for soluble boron under normal conditions, as long as the maximum k-eff remains below 1.0 for flooding with unborated water. This option is currently not applied to the STC, and unborated water was used for the calculation to show that the maximum k-eff is below 0.95.

Revised Loading Curves

RAI responses 4-5 through 4-7 required changes to the design basis calculations that resulted in revised loading requirements. Specifically considered were:

- A conservatively low specific power is used in the depletion calculations instead of an upper bound power (see response to RAI 4-5)
- Burnable Poison is now assumed over the full active length of the active fuel region, instead of only over the reduced length of the burnable poison design (see response to RAI 4-6)
- A conservative radial burnup gradient is now considered in all assemblies (see response to RAI 4-7)
- All design basis calculations are performed with increased number of cycles (see responses to RAI 4-2).
- The maximum credited burnup is now limited to 50 GWd/mtU. This necessitates an enrichment limit for Configuration 1-B (see response to RAI 4-8)

The minimum required burnup, and the calculated maximum k-eff values are listed in the following tables.

Parameter	Configuration 1 A			
Enrichment, wt% U-235	2.0	3.0	4.0	5.0
Burnup credited in Calculation, GWd/mtU	5.1	21.0	35.1	46.4
Minimum required Burnup including 5% Burnup Uncertainty, GWd/mtU	5.4	22.1	36.9	48.7

Maximum k-eff	0.9354	0.9445	0.9464	0.9478

Parameter	Configuration 1 B			
Enrichment, wt% U-235	2.0	3.0	4.0	4.5
Burnup credited in Calculation, GWd/mtU	5.7	27.2	42.5	50
Minimum required Burnup including 5% Burnup Uncertainty, GWd/mtU	6.0	28.6	44.6	52.5
Maximum k-eff	0.9409	0.9445	0.9465	0.9468

Note that only the design basis calculations demonstrating compliance with the regulations were updated in the licensing report to additionally consider the effect of those aspects listed above. Other studies documented in the licensing report and in these RAI responses do not include these revised assumptions.

Summary

The STC is a 10CFR50 component, with criticality safety requirements meeting the acceptance criteria in 10CFR50.68. However, the design of the STC basket is based on a dry storage and transportation design (MPC-32), which has a much higher neutron absorption than typical wet storage systems. Also, the burnup credit methodology from a transport package design was used, which is much more conservative (i.e. has higher minimum required burnup values) than typical wet storage criticality methodologies. Additionally, the STC basket is flooded with borated water (2000 ppm) although it is not credited for normal conditions, and during transfer operations there is no credible event which would cause a loss of boron. Together, this combination results in a substantial subcriticality margin for the fuel-loaded STC in its most reactive configuration.

NRC RAI 4-1

Provide a list of the isotopes and a summary table to show the biases and uncertainties that are applied to the major actinides and the best estimate correction factors that are applied to minor actinides and fission products for which burnup credits are taken. The information provided in

the revised SAR pages supplied with the previous RAI response does not adequately address this.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-1

The list of major actinides considered in the analyses is presented in Section 4.A.1.3 of the licensing report. The minor actinides and fission products considered are listed in the supporting criticality analysis report, HI-2084176, Table A-3. As a summary, a complete list of those isotopes is presented below.

PROPRIETARY TABLE REMOVED

The bias and bias uncertainty for the major actinides is shown in the licensing report, Table 4.7.14, and also shown graphically in Figures 4.A.1 and 4.A.2.

PROPRIETARY TEXT AND TABLE REMOVED

It is noted that isotopic correction factors may also be available from publicly available sources, as an example see NUREG/CR-7012. However, correction factors can only be compared if they were calculated for the same depletion code, with the same version of the code, and the same cross section library. For the example of the NUREG/CR-7012 this is not the case, since the NUREG presents correction factors for a different code, namely Triton from the Scale code system. Fuel compositions, if they were determined for the same enrichment, burnup, cooling time and core operating conditions, could be determined with different depletion codes.

NRC RAI 4-2

Provide justification for the conclusion that 10000 histories, 80 skipped cycles, and 200 accumulated cycles are acceptable for both configuration A and B, the 8 fuel assembly basket in particular.

In general, the MCNP code is sensitive to the number of cycles skipped and the ksrc specifications for loosely coupled systems like the 8 fuel assembly basket configurations. For cases like this, the users often have to check the convergence of both the effective neutron multipilation factor (Keff) and the source term. The Shannon Entropy often is used to test the convergence of the eigenvalue calculations. It is not clear, however, how these control parameters are determined. The applicant is requested to provide justification for the selection of the values of these control parameters.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-2

For MCNP4a, which does not automatically calculate the Shannon Entropy, the convergence

has been previously checked by performing selected calculations with larger number of cycles, as an example see [K.C], Section 6.E.4.7. However, a newer version of the code, MCNP5, does calculate that value. MCNP5 was therefore used to perform the evaluation of the Shannon Entropy for the three principal configurations (Configuration 1 A and 1 B for 5% enrichment listed in Table 4.7.1, and Configuration 2 for the 8 assembly basket with fresh fuel listed in Table 4.7.2) and original and increased number of calculational cycles. The acceptance criteria from the MCNP5 manual were used in the evaluation, which provides a verification of the minimum number of skipped cycles that need to be used. Several combinations of increased number of cycles and/or increased numbers of particles per cycle were evaluated. The evaluations show that the total number of cycles needs to be increased by a factor of 4 while the number of particles per cycle is kept constant to meet the Shannon-Entropy convergence criteria. This requires increasing the numbers of cycles from 200 to 800 for the configurations with spent fuel (12 assemblies), and from 160 to 640 for the fresh fuel configuration (8 assemblies). Note that the number of particles (histories) per cycle is kept at 10000, since studies involving changes in this amount, with or without changes to the number of cycles did not indicate any further improvements.

Results are summarized in the following table, which shows the calculated k-eff for the original and the increased number of cycles, and for MCNP4a and MCNP5. Note that the same cross section libraries were used in both MCNP4a and MCNP5 for this comparison.

Configuration	200 (Spent) / 160 (Fresh) Cycles		800 (Spent) / 640 (Fresh) Cycles	
	MCNP4a	MCNP5	MCNP4a	MCNP5
Spent Fuel, Configuration 1 A	0.9189	0.9189	0.9220	0.9215
Spent Fuel, Configuration 1 B	0.9199	0.9190	0.9214	0.9211
Fresh Fuel, Configuration 2	0.9244	0.9228	0.9250	0.9240

The following observations are made:

- Results for MCNP4a and MCNP5 are in good agreement for each configuration, indicating that conclusions from the MCNP5 calculations are applicable to the MCNP4a calculations
- The higher number of cycles results in slightly higher values in k-eff.

In order to ensure that the maximum k-eff values are determined, the revised design basis MCNP calculations presented in the introduction to these RAI responses are performed with the number of cycles increased to 800 for spent fuel and 640 for fresh fuel. The design basis calculation for fresh fuel shown in Table 4.7.2 of the licensing report is also re-performed, with the new result shown below.

Parameter	8 Assemblies
Enrichment, wt% U-235	5.0
Burnup, GWd/mtU	0.0
Maximum k-eff	0.9357

The licensing report Section 4.7.7 and the corresponding TS section 3.1.2 have been revised accordingly.

It is noted that there is a recent publication ("Statistical Coverage Concerns in a Revised 'k-Effective of the World' Problem", Brian C. Kiedrowski and Forrest B. Brown, ANS Conference Summer 2011) where in addition to running a large number of cycles, the number of neutrons per cycle (NPC) needed to be increased from 10,000 to 50,000 to obtain sufficiently accurate results. However, this was for a hypothetical configuration specifically designed to make convergence difficult, with features that are not present in the STC. Those features in comparison to the STC are as follows:

- The geometry in the publication is a 9x9x9 array of spheres in water, with the center sphere having a larger diameter (i.e. being more reactive). For 10,000 neutrons, only 10000/9x9x9 = 14 particles would have been started initially in the one larger sphere, making it difficult to detect the specific condition of this sphere. Even for 50,000 NPC, this value increases only to about 70. For spent fuel (in the STC and any other basket), the more reactive sections are typically the top and bottom end, consisting of at least 5% of the active length at both ends. Therefore, at least 1000 particles would start in those areas even with 10,000 NPC, far more than in the center sphere with 50,000 NPC.
- The spheres in the geometry in the publication are loosely coupled, with a center-to-center distance of 20 cm and a diameter of 8 cm, i.e. a surface-to-surface distance of about 12 cm. In the STC, the dominating geometry from a criticality perspective (even for the 8 assembly basket configuration), is the close face-to-face positioning of the assemblies that creates intensive coupling between the assemblies.
- The geometry in the publication contains asymmetric coupling through a cadmium layer on the inner (larger) sphere. No such strong asymmetry exists in the STC, although there would be some small asymmetry due to the fact that the neutron absorber plates are only located on one side of every basket wall.

In summary, the geometry in the STC is different from the relevant features in the publication that result in the convergence problem. The number of 10,000 NPC derived from the evaluation of the Shannon Entropy is therefore considered sufficient for the calculations for the STC.

NRC RAI 4-3

Provide justification for the applicability of the selected critical experiments to the code benchmark for the system of the eight fuel assembly configuration.

From the selected benchmark critical experiments, it appears that none of them has a configuration similar to the configuration of the eight fuel assembly basket. The applicant is requested to provide justification for the applicability of the selected critical experiments for the eight fuel assembly basket.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-3

The benchmark experiments do not have a configuration similar to the 8 assembly basket configuration, where a large water space exists in the center of the basket. However, this large water gap is not the dominating feature from a criticality perspective since there is practically no neutronic interaction across such a large water gap of more than 18 inches. The dominating configuration from a criticality perspective is therefore the close location of several assemblies, which is well represented by the standard benchmark experiments. The critical experiments that were used are therefore sufficient, and no further critical experiments are required for this configuration. Further note that the results for the 8 assembly basket configuration show substantial margin to offset any small differences in bias and bias uncertainty.

NRC RAI 4-4

Review and provide a corrected term of relative burnup in reference to Table 4.5.3.

Table 4.5.3 contains two pairs of columns, one pair is relative burnup at zero burnup and the second pair is relative burnup at 45 gigawatt days per metric ton uranium (GWD/MTU). It appears that the term "relative burnup" should be "normalized power." The term "relative burnup" is not determined if there is no burnup; 0/0 is not defined. The applicant is requested to review and provide corrected information as necessary to accurately represent the data.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-4

The intent of Table 4.5.3 is to specify two data points for each axial node to allow the calculation of the relative burnup for any given assembly average burnup, but not to imply that there is an axial burnup distribution for an assembly with zero burnup. The term "relative burnup" is therefore correct. Table 4.5.3 has been changed to the version below, where the zero burnup column is replaced by a 5 GWd/mtU column, with appropriate relative burnups to represent the same linear relationship between assembly average burnup and relative burnup for each node as before. Additionally, changes and additions were made to the data in the table as follows:

- Initially, generic, though conservative, axial burnup profiles were used. However, IP3 profiles are now available for three recent cycles (Cycles 14 through 16) of unloaded fuel, for a total of about 290 fuel assemblies. These profiles are for fuel with enriched axial blankets. A single axial profile has been determined that bounds the relative burnup profile in all those assemblies, regardless of the burnup. This profile is also listed below, together with the

corresponding axial segmentation, which is non-uniform in the areas near the ends of the assembly.
- The evaluation for the Westinghouse 15x15 non-blanketed assemblies was based on a dataset of only 4 assemblies, over a narrow burnup range. Determination of linear functions of the relative burnup in each node may therefore not be appropriate. Instead of this linearization, all four individual burnup profiles are shown and evaluated.

For a discussion on how those profiles are used please see response to RAI 4-18.

Table 4.5.3
Axial Burnup Distribution

PROPRIETARY TABLE REMOVED

NRC RAI 4-5

Provide a justification of the burnup credit calculations submitted, or resubmit with lower bounding specific power.

In Section 4.7.1.2.1 of the SAR, the applicant states that the maximum value for all plants with Westinghouse 15x15 fuel assemblies is used as bounding assembly specific power. However, studies published in "NUREG/CR-6665, Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel" and "M. D. DeHart, Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996" indicate that using lower specific power produces conservative results for casks taking burnup credit for both actinides and fission products. To ensure criticality safety, lower specific power should be used for applications that take credit for both actinides and fission products. The applicant is requested to review and redo its burnup credit analyses using appropriate specific power.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-5

While the effect of the specific power on the maximum k-eff is comparatively small (see studies presented in Table 4.7.16 of the licensing report), it is recognized that a lower specific power, not a higher specific power, is the more conservative assumption, as discussed in NUREG/CR-6665. To take this into consideration, the revised design basis analyses have been performed with a conservatively low specific power. A specific power of 60% of the core average specific power was used as the conservatively low value. This assumption is based on the studies

performed for the HI-STAR 100 Methodology, documented in Appendix A of [L.K], which indicate that only a small fraction of assemblies (less than 1%) would have a discharge burnup (and hence average specific power) of less than 60% of the corresponding core average value, and is well below the lowest specific power for any assembly at IP3.

Section 4.7.1.2.1 of the licensing report has been updated to incorporate the above discussion. Note that this change presents a slight deviation from the original HI-STAR 100 methodology; it is more conservative.

NRC RAI 4-6

Demonstrate that the fuel assembly depletion code, CASMO-4, is capable of modeling both the poisoned portion and the non-poisoned portion of the wet annular burnable absorber (WABA) rods in a fuel assembly.

Page 4-25 of the revised SAR states that the length of the poisoned region of WABA rods varies from 120 to 128 inches. The top and bottom parts of a WABA rod are just cladding. This leaves the top and bottom parts of a fuel assembly unexposed to poison. The SAR further states: "Most of the calculations with WABAs take that into consideration, i.e., the depletion calculations for the top and bottom node are performed with the cladding of the WABA in place, but without any poison while the depletion calculations for the central part of the assembly contains both the WABA cladding and the absorber." It is not clear, however, how the CASMO-4 code models this three dimensional effect for the fuel assembly containing WABAs. The applicant is requested to demonstrate that the fuel assembly depletion code, CASMO-4, is capable of modeling both the poisoned portion and the non-poisoned portion of the WABA rods in a fuel assembly.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-6

To alleviate any concerns regarding the modeled length of the burnable poison rods, and to make the methodology consistent with that used for the HI-STAR 100, the revised design basis calculations have been modified such that the burnable poison rods and IFBA coatings now cover the entire active length of the fuel.

NRC RAI 4-7

Demonstrate that reactivity effect of the burnup gradient is negligible for the 12 fuel assembly loading configuration.

Page 4-29 of the revised SAR discusses the potential reactivity impact of burnup gradients across the fuel assemblies and further states: "However, since this is a highly unlikely configuration, and since even then the reactivity is small compared to the remaining safety margin (see Section 4.7.9), all design basis calculations are performed with a uniform planar burnup distribution." This assessment, however, may not be justified without a quantitative analysis of the system with consideration of the burnup gradient in the fuel assemblies. The staff is particularly concerned with the small system 12 fuel assembly shielded transfer cask. The applicant is requested to demonstrate that reactivity effect of the burnup gradient is negligible for the 12 fuel assembly loading configuration.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-7

The effect of burnup gradients has now been evaluated, using the same approach that was utilized in [K.C]. For this evaluation the bounding burnup gradient is assumed in each assembly, in addition to assuming that the lower burned section of each assembly is faced inwards toward the center of the basket. This is a simple but very conservative assumption, creating a configuration that is highly unlikely in reality. Results are listed in the table below, together with results of the corresponding calculations that do not consider any burnup gradients.

Axial Burnup profile	Constant					
Enrichment, wt%	2		3.5		5	
Burnup, GWd/mtU	10		30		45	
	K-calc	Delta k-calc	K-calc	Delta k-calc	K-calc	Delta k-calc
WABA - Reference	0.8751	-	0.8916	-	0.9098	-
WABA - Tilted Burnup	0.8766	0.0015	0.8952	0.0036	0.9144	0.0046
RCCA - Reference	0.8985	-	0.9294	-	0.9479	-
RCCA - Tilted Burnup	0.8974	-0.0011	0.9324	0.0030	0.9508	0.0029
Axial Burnup Profile	Profile					
Enrichment, wt%	2		3.5		5	
Burnup, GWd/mtU	10		30		45	
	K-calc	Delta k-calc	K-calc	Delta k-calc	K-calc	Delta k-calc
WABA - Reference	0.8754	-	0.9041	-	0.9169	-
WABA - Tilted Burnup	0.8765	0.0011	0.9099	0.0058	0.9212	0.0043
RCCA - Reference	0.8960	-	0.9322	-	0.9471	-
RCCA - Tilted Burnup	0.8964	0.0004	0.9365	0.0043	0.9497	0.0026

The results show reactivity effects of up to 0.0058 delta-k. This is slightly larger than the effect taken from [K.C], which was 0.0038.

To alleviate any concerns regarding the radial burnup gradients, the revised design basis calculations assume radial burnup gradients in all analyses. Note that this change presents a slight deviation from the original HI-STAR 100 methodology; it is more conservative.

NRC RAI 4-8

Demonstrate that the chemical assay data used for code benchmarking are sufficient to cover fuel assemblies with burnup exceeding 50 GWD/MTU.

Page 4-32 of the revised SAR states: "Note that ISG-8 Rev. 2 recommends an upper limit for burnup credit of 50 GWD/MTU, based on an apparent lack of data above this value. However, the evaluation of the bias and bias uncertainties and isotopic correction factors from

benchmarking have been performed using a statistical approach that accounts for the limited amount of data available and assigns higher uncertainties for parameters that are further away from (i.e. at higher burnups) the experimental data. Also, isotopic benchmarking includes data for fuel exceeding 50 GWD/MTU. An additional burnup limit does therefore not appear to be necessary and is therefore not applied here." The staff reviewed the relevant publications and found that the bias and bias uncertainties determined based on the comparisons of the calculated and experimental isotopic concentrations are valid only for the burnup range of the experimental data. Based on the staff's review, it appears that there is a very limited number of chemical assay data that have burnup in excess of 50 GWD/MTU and the statistical analyses using the very limited data points may not satisfy the criterion for obtaining meaningful results. In addition, it appears that the burnup values are the local burnup of samples rather than the fuel assembly average burnup that is used in fuel qualification calculation. The applicant is requested to demonstrate that the chemical assay data used for code benchmarking are sufficient to cover fuel assemblies with burnup exceeding 50 GWD/MTU. If the criticality analyses are limited to 50 GWD/MTU, revise TS LCO 3.1.2.a.2, 3.1.2.b.2, and Note b to TS Table 3.1.2-1 to show the revised limit.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-8

Entergy understands that this aspect of the STC evaluation did not follow the approach developed and presented for the HI-STAR 100 in [K.C]. To bring the STC analysis in line with the HI-STAR 100 analysis, the credit for burnup is now limited to 50 GWd/mtU. This only affects Configuration 1 B, since only for Configuration 1 B burnup requirements above 50 GWd/mtU were initially calculated. For this configuration, a maximum enrichment of 4.5 wt% is now used, with a credit of 50 GWd/mtU (fuel above this enrichment, will need to be transferred using the 8 assembly configuration that does not credit burnup). Note that to account for a burnup record uncertainty of 5%, the corresponding minimum required fuel burnup may be as high as 52.5 GWd/mtU, however, the amount of burnup *credited* is still only 50 GWd/mtU, consistent with ISG-8 Rev. 2.

NRC RAI 4-9

[deleted]

NRC RAI 4-10

Page 4-35 of revision 3 of the licensing report makes reference to Table 4.7.11 and explains the cases evaluated in that table. To clarify the change between the second to last and last rows please see Table 4.7.1. This table shows the combination of bias and bias uncertainty for all design basis calculations. It is important to note there that the bias for various conditions is listed as a negative value, i.e. one that would reduce the maximum k-eff if considered. This reduction in not applied in the design basis calculation, consistent with guides such as NUREG/CR-6698. Biases are therefore truncated to 0 if they are below 0 in Table 4.7.1 However, in the second-to-last row of Table 4.7.11, the negative biases are applied, i.e. they would offset some uncertainties and reduce the maximum k-eff. Note that this the negative bias is only applied to the studies presented in Table 4.7.11 that evaluate margin, not for the design basis calculation, and there is no claim that such an approach would be acceptable for design basis calculations that determine loading curves. Therefore, going from the second-to-last row

to the last row in Table 4.7.11, all biases that would reduce maximum k-eff are set to zero, so the last row represents the design basis calculation as discussed in the text on Page 4-35.

Response to RAI 4-10

Page 4-35 of revision 3 of the licensing report makes reference to Table 4.7.11 and explains the cases evaluated in that table. To clarify the change between the second to last and last rows please see Table 4.7.1. This table shows the combination of bias and bias uncertainty for all design basis calculations. It is important to note there that the bias for various conditions is listed as a negative value, i.e. one that would reduce the maximum k-eff if considered. This reduction is not applied in the design basis calculation, consistent with guides such as NUREG/CR-6698. Biases are therefore truncated to 0 if they are below 0 in Table 4.7.1. However, in the second-to-last row of Table 4.7.11, the negative biases are applied, i.e. they would offset some uncertainties and reduce the maximum k-eff. Note that this negative bias is only applied to the studies presented in Table 4.7.11 that evaluate margin, not for the design basis calculation, and there is no claim that such an approach would be acceptable for design basis calculations that determine loading curves. Therefore, going from the second-to-last row to the last row in Table 4.7.11, all biases that would reduce maximum k-eff are set to zero, so the last row represents the design basis calculation as discussed in the text on Page 4-35.

NRC RAI 4-11

[deleted]

NRC RAI 4-12

Provide detailed information on and justification for the interpolation scheme used in determining the isotopic concentrations as a function of burnup.

Page 4-30 of the revised SAR states: "Since it is necessary to model the axial burnup distribution, a large number of isotopic compositions at irregular burnups are required. Given the significant number of criticality calculations and studies performed for the burnup credit evaluations, it would be impractical to perform CASMO-4 depletion calculations for each of these burnups. Instead, CASMO-4 runs are performed for fixed burnups at 2.5 GWD/MTU intervals (or less), and intermediate isotopic values are determined by linear-linear interpolation. Calculation presented in [16] show that this is an acceptable approach by comparing results with calculations that were based on a smaller interval of 1.0 GWD/MTU in the CASMO calculation." From these statements, it is not clear what interpolation scheme was used in determining the isotopic concentrations at various burnup values. It is not clear whether the isotopic concentrations of the various isotopes of interest can be determined with this interpolation scheme. The applicant is requested to provide detailed information on and justification for the interpolation scheme used in determining the isotopic concentrations as a function of burnup.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-12

The interpolation scheme is linear-linear (see Section 4.7.3.1 of the licensing report), i.e. the

concentration of each isotope is assumed to be a linear function of burnup within each burnup interval. In the original development of the methodology, additional calculations with a 1 GWd/mtU interval had been performed to justify the interpolation scheme, as recognized in the RAI. To show that this conclusion is also applicable to the STC, those studies were re-performed for the STC, and results are listed below together with current results for 2.5 GWd/mtU intervals.

Axial Burnup profile	Constant					
Enrichment, wt%	2		3.5		5	
Burnup, GWd/MTU	10		30		45	
	K-calc	Delta k-calc	K-calc	Delta k-calc	K-calc	Delta k-calc
WABA - Reference	0.8751	-	0.8916	-	0.9098	-
WABA - 1 GWD	0.8752	0.0001	0.8901	-0.0015	0.9097	-0.0001
RCCA - Reference	0.8985	-	0.9294	-	0.9479	-
RCCA - 1 GWD	0.8988	0.0003	0.9287	-0.0007	0.9473	-0.0006
Axial Burnup Profile	Profile					
Enrichment, wt%	2		3.5		5	
Burnup, GWd/MTU	10		30		45	
	K-calc	Delta k-calc	K-calc	Delta k-calc	K-calc	Delta k-calc
WABA - Reference	0.8754	-	0.9041	-	0.9169	-
WABA - 1 GWD	0.8753	-0.0001	0.9044	0.0003	0.9167	-0.0002
RCCA - Reference	0.8960	-	0.9322	-	0.9471	-
RCCA - 1 GWD	0.8963	0.0003	0.9332	0.0010	0.9466	-0.0005

The statistical uncertainty is about 0.0015 delta-k (95/95) for all delta-k values. As can be seen, all delta-k values are at or below that value, indicating that the differences are statistically insignificant. The current interpolation scheme using 2.5 GWd/mtU is therefore acceptable.

Note that the design basis calculations are always performed with burnable absorbers or control rods inserted for the entire irradiation period of the assembly. There are therefore no situations with possible discontinuities in isotopic concentration from the insertion or removal of rods that may require an adjustment of the interpolation scheme.

NRC RAI 4-13

Provide a list of isotopes included in the baseline cases presented in page 4-35 and Table 4.7.11 and justification for including all isotopes in the baseline case.

Page 4-35 of the revised SAR presents a baseline scenario and comparisons with the results of various ways of treating the uncertainties involved in the isotopic concentrations of spent fuels calculated using the CASMO code. From the description in the text and the results presented in Table 4.7.11 of the SAR, it seems that the applicant indicates that all isotopes that are tracked by the CASMO code were included in the STC criticality calculations. From page C-6 of Appendix C to BURNUP CREDIT FOR THE MPC-32, Holtec Report No. HI-2012630, it appears that Kr-83 was included in the burnup credit analysis. Page C-64 of Appendix C to BURNUP CREDIT FOR THE MPC-32, Holtec Report No: HI-2012630, further states: "Two series were

performed, one for an enrichment of 4 percent at a burnup of 32.5 GWD/MTU, and one for an enrichment of 5 percent and at a burnup of 50 GWD/MTU. The reference calculations use all isotopes from CASMO without any corrections factors, including the lumped fission products." From these statements or input files in the various documents, it was not clear what exactly those isotopes are and what the justification is for including each of those isotopes. It is the staff's understanding that some of the isotopes, such as krypton and xenon, cannot be credited because these isotopes are gaseous at the expected temperature. Because the exact locations of these isotopes in the fuel rods cannot be determined, it is not considered justified to included in criticality calculations, even in the Commercial Reactor Critical experiments. The applicant is requested to provide a list of all the isotopes that were included in the baseline calculation and justification for inclusion of each of the isotopes in the criticality calculations.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-13

The text in the Holtec Report HI-2012630 referred to in the RAI belongs to earlier calculations that used a larger set of isotopes. The final design basis calculations in [K.C] were documented in Supplement 2 of report HI-2012630. Those calculations, as well as the design basis calculations for the STC, use only the isotopes listed in the table provided in the response to RAI 4-1. Specifically, those calculations do not credit isotopes such as krypton, xenon, or lumped fission products. The only isotopes credited are those that are validated through the chemical assays. However, note that the full set of isotopes available from CASMO is utilized in the calculations listed in the first row in Table 4.7.11 to estimate the margin of the analysis.

The full set of isotopes is also used in the CRC calculations. This is necessary since only a best estimate approach (using all isotopes) in a benchmark calculation can provide a meaningful bias and bias uncertainty. This is recognized in NRC's SER for the HI-STAR 100 (ML062860201), which states on Page 22 "The CRC benchmark calculations included 179 isotopes, whereas, the design calculations included 25 isotopes. This approach is consistent with the principle that a best estimate which includes all known effects that reduce k_{eff} should be used when establishing the bias of the criticality calculations even if these effects are not used in the design calculations". In this context please see also the discussion on the relevance of the CRC benchmarks for the safety analysis, as discussed in the response to RAI 4-16.

NRC RAI 4-14

Revise the context of the discussions regarding the temperature impact to neutron fluxes at CRCs and MPC on page A-25 of the Appendix A to the Holtec Report No. Hi-2084176 to make the context consistent with the referenced figures.

Page A-25 of the Appendix A to Holtec Report No. Hi-2084176 discusses the temperature impact to neutron fluxes at CRCs and MPC. However, it appears that there are no differences between Figure 53 and Figure 53a. The discussions on page A-25 of the report does not make reference to Figure 53a either. The applicant is requested to revise the context of the discussions regarding the temperature impact to neutron fluxes at CRCs and MPC on page A-25 of the Appendix A to the Holtec Report Hi-2084176 to make the context consistent with the referenced figures. More importantly, the applicant should explain why there is no spectral difference between these two systems given the fact that the Doppler Effect at 600K is significant different from that at 300K.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-14

It is correct that there is no difference in the data presented in Figure A.53 and Figure A.53a. Figure A.53a simply shows the same information as Figure A.53, just with a linear y-axis. The text contains a typographical error as follows: the sentence "Figure A.53 shows the same data on a linear plot." should have stated "Figure A.53a shows the same data on a linear plot." The Holtec Report No. HI-2084176 has been updated to correct this.

The reason that there is essentially no spectral difference between the two systems (CRC and MPC) in those plots is that for those plots, the MPC was also evaluated at 600K, which is the same temperature used in the CRC calculations. As discussed on Page A-25, this is to show that given the same temperature, the spectra between CRCs and MPCs are essentially the same. The discussion on Page A-24, together with Figures A.50 through A.52a, addresses the comparison of spectra at different temperatures. Further discussion on the temperature is also included on Page A-25. In this context please see also the discussion on the relevance of the CRC benchmarks for the safety analysis, as discussed in the response to RAI 4-16.

NRC RAI 4-15

Explain what has been done in the benchmark calculations to account for the differences in the neutron absorption rates between the commercial reactor criticals and the MPC.

Pages A-22 and A-25 of Appendix A of the Holtec Report No. Hi-2084176 provides detailed comparison of the B-10 reaction rates in the Commercial Reactor Criticals and MPC. From figure A.55a or figure A.55b, it appears that the difference is fairly significant and the applicant recognized this difference in the discussion. However, it was not clear what has been done in the benchmark calculation to account for the differences. The applicant is requested to explain what has been done in the benchmark calculations to account for the differences in the neutron absorption rates between the commercial reactor criticals and the MPC.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-15

The studies on B-10 discussed on Pages A-22 through A-25 of Holtec Report HI-2084176 were originally performed during the HI-STAR licensing process since there had been the concern that the reactivity effect of the B-10 would be substantially different between the CRCs and the MPC. However, the results presented in Figure A.35 indicate that the reactivity effect of B-10 in the CRC and MPC are in fact comparable. This alleviated the concerns about the representation of B-10 in those benchmarks. Therefore, while an additional comparison of reaction rates was also performed (presented in Figures A.55a and A.55b), no further actions resulted from this study in the HI-STAR 100 burnup credit methodology. In this context, please also see the response to RAI 4-16 below on the purpose and relevance of the CRC benchmarks for the safety analyses.

NRC RAI 4-16

Demonstrate that the CRC benchmark calculations could be used as the sole benchmarking for the burnup credit methodology.

On pages A-26 of Appendix A to Holtec Report No. Hi-2084176, the applicant concluded, after some discussions, "Based on these results and conclusions, the CRC benchmark calculations could justifiably be used as the sole benchmarking for the burnup credit methodology. The only bias would then be the uncertainty of the CRCs (when conservatively ignoring the negative bias), and this would support the use of all isotopes generated by CASMO, without any correction factors." Given the fact that the CRC models that the applicant built included many isotopes that are not available to burnup credit (see RAI 4-13), if not carefully considered some significant errors might have been introduced in the benchmark calculations using the CRCs. Although the results of code benchmark with inclusion of all isotopes CRC models show good results, using all of the isotopes generated by CASMO may unintentionally affect the accuracy of the bias and bias uncertainty. The applicant is requested to demonstrate by calculations and analyses that all isotopes from the CASMO calculations are available and appropriate for burnup credit. Quantitative calculations are necessary to quantify the bias and bias uncertainties associated with each isotope to be included in the burnup credit analyses.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-16

The CRC benchmarks were *not* used as the sole benchmark as explained in the text following the quoted section from Page A-26. The text continues "However, as a more conservative approach, and more consistent with the recommendations in ISG 8, the primary benchmark experiments to support the burnup credit methodology are selected to be the criticality benchmarks based on fresh UO_2 and MOX fuel, and the isotopic benchmarks of spent fuel. The CRC benchmarks are used only to provide additional support for credit of the fission products and minor actinides, specifically regarding the uncertainty in the cross sections of these isotopes." This is reflected in NRC's SER for the HI-STAR 100 (ML062860201), which states on Page 22 "Staff determined that the CRCs can provide some useful information when coupled with other benchmark data but does not believe that the applicability of the CRCs has been demonstrated to the degree that they can be used as the sole means of benchmarking the criticality calculations for the analysis of the MPC-32 package for transport of spent fuel.", but then continues "Based on the margin of safety and on risk informed considerations of the probability and expected consequences to public health and safety from the use of the MPC-32 for transport, staff accepts the CRCs as part of this overall benchmark analysis.". Using the CRCs only to support credit for selected minor actinides and fission products significantly reduces the overall importance of the CRC benchmarks on the safety analyses, compared to calculations that would be intended as sole benchmarks. Nevertheless, the full uncertainty derived from the CRCs is applied for the design basis safety analyses, and is combined with the uncertainties and adjustments from the other benchmarks.

It is certainly true that a more desirable situation from a validation perspective would be to have a bias and bias uncertainty for each isotope and for both depletion analyses (i.e. concentration of the isotope) and criticality analysis (i.e. reactivity worth of the isotope). Unfortunately, the current state of the available benchmarking experiments does not fully support such an approach: We have chemical assays for actinides and fission products in spent fuel and only use those isotopes validated through those experiments, with appropriate correction factors

(see also response to RAI 4-1). We also have the validation of the criticality calculations of major actinides through the fresh and MOX fuel critical. However, there are no significant publicly available experiments to validate the reactivity worth of individual minor actinides and fission products. For this very reason, the CRCs were analyzed as additional benchmarks and with NRC concurrence (see excerpts from NRC's SER listed earlier in this response), since they at least evaluate the combined effect of those isotopes. While this is not the perfect validation, it was considered sufficient to support the safety evaluation of the HI-STAR system.

In order to determine any bias and bias uncertainty from the CRC benchmarks, it is in fact necessary to perform those as best estimate calculations and include all isotopes available in CASMO. As already discussed in the response to RAI 4-13, this is recognized in NRC's SER for the HI-STAR 100 (ML062860201), which states on Page 22 "The CRC benchmark calculations included 179 isotopes, whereas, the design calculations included 25 isotopes. This approach is consistent with the principle that a best estimate which includes all known effects that reduce k_{eff} should be used when establishing the bias of the criticality calculations even if these effects are not used in the design calculations".

NRC RAI 4-17

If applicable, review and correct the reference on page 11 of Holtec Report No: HI-2094486, "MCNP Benchmark Calculations."

Page 11 of Holtec Report No: HI-2094486, "MCNP Benchmark Calculations," cites reference 5 in the reference list of this report. There appears to be an error in this reference because reference 5 of the report is NUREG/CR-6998 while the text of the report seems to refer to an error analysis technique for normality test of data used in statistical analyses. The applicant is requested to review the document and make corrections if necessary.

Response to RAI 4-17

No corrections are necessary. The Shapiro-Wilk test and the corresponding tables are listed in NUREG/CR-6698 (not 6998), which is reference [5] in HI-2094486.

NRC RAI 4-18

Regarding the burnup profile used in the burnup credit analyses of the Indian Point fuel transfer shielded transfer canister:

- Explain why the burnup profile derived in Appendix D to Holtec report HI-2012630 is not consistent with that developed in NUREG/CR-6801;
- Explain why the burnup profiles, as shown on pages D-19 to D-23, are not normalized;
- Explain why the burnup profiles, as shown on pages D-35 and D-36, have the diverged sections and justification for their applicability to the STC;
- Provide justification for the applicability of the burnup profile for future discharged fuel assemblies.

This information is needed for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-18

Before the four sub-questions of this RAI are addressed, a brief discussion on axial profiles (enrichment and burnup) is presented, together with the comparisons of the effects of profiles on the design basis calculations.

Enrichment Profiles

As at many other plants, Indian Point 3 has initially used fuel with an axially constant enrichment, followed by using fuel with natural uranium axial blankets, and finally using fuel with enriched (2.6% and 3.2%) axial blankets. Fuel with different axial enrichment profiles also show different burnup profiles, with lower relative burnups in the blanket sections. Previous comparisons of fuel with different enrichment profiles have consistently shown that given the same assembly average burnup, blanketed fuel (both natural and enriched blankets) is bounded by (i.e. less reactive than) fuel with constant enrichment over the entire active height. This is due to the reduced U-235 amount in the blanket areas that only have a low burnup. Other criticality analyses, such as the recently approved Beaver Valley analysis, have credited this fact by developing different loading curves for different enrichment profiles, with lower burnup requirements for blanketed fuel. For IP3 fuel, no such approach is used, and the burnup requirements are based on the most reactive, i.e. axially constant, enrichment profile. Nevertheless, various enrichment and burnup profiles are evaluated for comparison purposes and to show that the approach is conservative.

Burnup profiles used in the Analyses

Initially, only generic burnup profiles were used: In the initial calculation that used a typical wet storage burnup credit methodology, profiles from NUREG/CR-6801 were used, whereas later, the profiles developed for Westinghouse 17x17 assemblies with the HI-STAR 100 methodology were used when the HI-STAR 100 methodology was used for the STC, supplemented with additional profiles from the same source used in the NUREG (i.e. the Profile Database from Yankee Atomic Report YAEC-1937). Burnup profiles from discharged fuel at IP3 are now available. However, those are only for three recently discharged cycles (Cycles 14 through 16), and for fuel with enriched blankets. To qualify all currently discharged fuel, and provide a basis for qualifying fuel discharged in the future, calculations have been performed with all applicable axial burnup and enrichment distributions. These distributions are discussed below, with a specific focus on applicability and inherent conservatisms:

1. Profiles for Westinghouse 17x17 assemblies without any axial blankets, developed for the HI-STAR 100 burnup credit methodology. These are the bounding profiles for most conditions, i.e. they result in the highest maximum k-eff. Since assemblies without axial blankets were only used in earlier cycles, they have a much longer cooling time than the 5 years assumed in the design basis criticality analysis, which provides additional but unspecified margin.
2. The profile database that was used to develop the Westinghouse 17x17 profiles for the HI-STAR 100 only contained 4 profiles for Westinghouse 15x15 assemblies, which is the design used at IP3. All four profiles were used, to confirm that results for those 15x15 profiles are equivalent to or bounded by results obtained for the 17x17 profiles.
3. Profiles from NUREG/CR-6801 are used, to provide further confirmation that the profiles from the HI-STAR 100 methodology are appropriate.

4. An axially constant burnup and enrichment distribution. While this distribution does not represent any realistic condition, it is traditionally used for added conservatism since it typically result in higher k-eff values at lower enrichments and burnups than the actual profiles.
5. Profiles for Westinghouse 15x15 and 17x17 assemblies with natural blankets, also from the Profile Database form Yankee Atomic Report YAEC-1937.
6. Profiles from IP3 for recently discharged fuel (Cycles 14 through 16) with enriched blankets.

Results for the calculations with the various profiles are summarized in the following tables.

Parameter	Configuration 1 A							
Enrichment, wt% U-235	2		3		4		5	
Burnup, GWd/mtU	5.1		21		35.1		46.4	
	Max. k-eff	Delta-k	Max. k-eff	Delta-k	Max. k-eff	Delta-k	Max. k-eff	Delta-k
1 - Reference profile - W17x17	0.9349	-	0.9445	-	0.9464	-	0.9478	-
3 - NUREG	0.9339	-0.0010	0.9386	-0.0059	0.9427	-0.0037	0.9468	-0.0010
4 - Flat	0.9354	0.0005	0.9325	-0.0120	0.9296	-0.0168	0.9362	-0.0116
5 - Natural Blankets	-	-	0.9244	-0.0201	0.9170	-0.0294	0.9208	-0.0270
6 - Enriched Blankets	-	-	-	-	0.9301	-0.0163	0.9370	-0.0108
Maximum k-eff	0.9354		0.9445		0.9464		0.9478	
Profile with Max. k-eff	Flat		W17x17		W17x17		W17x17	

Parameter	Configuration 1 B							
Enrichment, wt% U-235	2		3		4		4.5	
Burnup, GWd/mtU	5.7		27.2		42.5		50	
	Max. k-eff	Delta-k	Max. k-eff	Delta-k	Max. k-eff	Delta-k	Max. k-eff	Delta-k
1 - Reference profile - W17x17	0.9409	-	0.9445	-	0.9465	-	0.9468	-
3 - NUREG	0.9400	-0.0009	0.9423	-0.0022	0.9428	-0.0037	0.9450	-0.0018
4 - Flat	0.9409	0.0000	0.9425	-0.0020	0.9446	-0.0019	0.9456	-0.0012
5 - Natural Blanktes	-	-	0.9349	-0.0096	0.9334	-0.0131	0.9335	-0.0133
6 - Enriched Blanktes	-	-	-	-	0.9421	-0.0044	0.9414	-0.0054
Maximum k-eff	0.9409		0.9445		0.9465		0.9468	
Profile with Max. k-eff	W17x17		W17x17		W17x17		W17x17	

Parameter	Configuration 1 A			
Assembly No.	1	2	3	4
Enrichment, wt% U-235	4	4	4	4
Burnup, GWd/mtU	35.1	35.1	35.1	35.1
1 - Reference k-eff - W17x17	0.9464			
2 - W15x15	0.9283	0.9333	0.9320	0.9414
Delta k-eff	-0.0181	-0.0131	-0.0144	-0.0050

Parameter	Configuration 1 B			
Assembly No.	1	2	3	4
Enrichment, wt% U-235	4	4	4	4
Burnup, GWd/mtU	42.5	42.5	42.5	42.5
1 - Reference k-eff - W17x17	0.9465			
2 - W15x15	0.9408	0.9431	0.9415	0.9477
Delta k-eff	-0.0057	-0.0034	-0.0050	0.0012

The results support the following conclusions:

- The non-blanketed profiles (Westinghouse 15x15, 17x17, Flat) are always bounding. As stated above, this provides additional margin since those assemblies have a longer cooling time than the 5 years used in the design basis criticality calculations.
- The Westinghouse 15x15 profiles and the profiles from the NUREG result in similar k-eff values as, or are bounded by the Westinghouse 17x17 profiles, providing additional assurance that the selection of the axial profiles is appropriate and conservative.
- Profiles for fuel with natural and enriched blankets result in maximum k-eff values that are significantly lower than those for the non-blanketed assemblies. The difference is between about 0.01 and 0.03 delta-k for natural blankets, and between about 0.005 and 0.015 delta-k for enriched blankets. This provides additional margin for those assembly types, which may have cooling times closer to 5 years used in the analysis.

Overall, the analysis is considered adequate and conservative for all fuel used at IP3, and all fuel assemblies currently in the IP3 spent fuel pool, up to and including assemblies unloaded in cycle 16. Future assemblies, i.e. assemblies unloaded from cycle 17 and following, will be evaluated before they are loaded into the STC to ensure they are bounded by the design basis analyses.

Responses to the individual questions in the RAI are listed below.

1. HI-2012630 vs. NUREG/CR-6801:

The axial burnup profile approach for the HI-STAR 100 was developed in parallel to NUREG/CR-6801, but both are essentially based on the same profile database, namely the database documented in the Yankee Atomic Report YAEC-1937, which was also used in an earlier report from Texas A&M University, "Bounding Axial Profile Analysis for the Topical Report Database" (Ref. 7 in the NUREG). The main technical reasons for developing a

methodology different from NUREG/CR-6801 and the Texas A&M University report in the HI-STAR 100 burnup credit methodology are as follows:

- The NUREG and the Texas A&M University report base the comparison of the profiles on one-dimensional (1-D) calculations. Holtec developed bounding profiles directly from the profile database. This is believed to be a more rigorous approach since it avoids the reliance on the 1-D calculations.
- The profiles listed in the NUREG and the Texas A&M University report create steps in the loading curves, which make application difficult. The Holtec-developed profiles are continuous functions of burnup in each axial section.

To alleviate any concerns with respect to the selection of burnup profiles, additional calculations were added with the profiles from NUREG/CR-6801. Results and comparisons to those calculations that use the profiles developed for the HI-STAR 100 methodology are presented earlier in this response.

2. Normalization:

A lower bound relative burnup as a function of the assembly average burnup is determined separately for each axial node. This results in non-normalized profiles. Since the value in each axial section is a lower bound value, the average over all sections in any given profile will always be below 1.0. However, the linearizations for the non-blanketed 15x15 assemblies were only performed from a small dataset of 4 assemblies over a small burnup range, resulting in average values that exceeded 1.0 for higher burnups. The linearization has therefore been removed, and the four profiles are individually analyzed (see also response to RAI 4-4).

3. Diverged Sections on Pages D-35 and D-36:

These show the effect of axial power shaping rods present in the active region during full power operation, an approach used in B&W reactors. As a result, the methodology in [K.C] distinguishes between B&W and Westinghouse assemblies in terms of bounding axial burnup profiles, and the profiles for B&W fuel show the effect of those axial power shaping rods while the profiles for Westinghouse fuel do not. Note that IP3 is a Westinghouse reactor and has not used axial power shaping rods, except in the first cycle, i.e. in fuel that has a significant cooling time. However, the profiles developed in NUREG/CR-6801 are now used in the analyses in addition to the profiles developed for the HI-STAR 100, and those do include all profiles from the database.

4. Future Profiles:

Procedures will be implemented to screen axial burnup distributions for future cycles to ensure that future fuel transferred with the STC will be bounded by the design basis criticality calculations for spent fuel. If it cannot be demonstrated that the spent fuel is bounded by the design basis calculations, then the fuel assembly will be transferred in the 8 assembly configuration that does not credit fuel burnup. This applies to all fuel discharged from cycle 17 and following, since Cycle 16 is the last cycle whose profiles are already included in the current evaluation.

Note that this represents a deviation from the HI-STAR approach, where no additional screening of axial distributions is required. However, it is recognized that this screening requirement is

consistent with recently approved 10CFR50 criticality analyses such as that for Beaver Valley Unit 2 (ML110890844, SER Section 3.2.4).

NRC RAI 4-19

Provide a justification for the conclusion made on page 2 of Holtec Report HI-2032973 that the CRC benchmarks tend to overestimate the actual reactivity and demonstrate that the models built for code benchmarking did not introduce additional bias in the criticality safety analyses.

Page 2 of Holtec Report, "Commercial Reactor Critical Benchmarks for Burnup Credit," states: "The average of the calculated reactivities is 1.0023, i.e., the calculations tend to overestimate the actual reactivity...." It is not clear, however, if the models include considerations of the impact of background neutron sources to the critical states, which would bring the fluxes to constant while the reactors were in fact at subcritical states. It was not clear either what would be the result if the models do not include gaseous or volatile isotopes such as krypton or xenon, that may accumulate in different parts of the fuel rods. It is not clear either how these isotopes are treated differently for CRCs at Beginning of Cycles (BOC) and CRCs at End of Cycles (EOC). The applicant is requested to reevaluate the conclusion and demonstrate that the models built for code benchmarking did not introduce additional bias in the criticality safety analyses.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-19

Discussions with a company that develops core design and core analysis software (Studsvik), and that has also performed many core analyses, indicate that aspects listed in the RAI, namely the impact of background neutron sources or the role of gaseous or volatile isotopes accumulating in different parts of rods, have a negligible effect on critical state points. Those aspects are in fact not modeled in the current state-of-the-art core analysis software, and there is no indication that this results in any relevant deficiencies of those analyses. We therefore believe the scope of the CRC benchmarking analyses is appropriate and sufficient, specifically in light of the limited role of the CRCs as Benchmarking experiments (see response to RAI 4-16). Note also that the input data compiled and documented by DOE, which describe and characterize the critical conditions, did not address those aspects discussed in the RAI. Further, we are unaware of any publicly available information that indicates these aspects could have a noticeable effect on critical state points.

Regarding the bias please note that any bias that would reduce reactivity is conservatively neglected. The fact that the CRC bias is slightly above 1.0 has therefore no impact on the safety analysis. However, the bias uncertainty from the CRCs is fully applied. Please see also response to RAI 4-16.

NRC RAI 4-20

Provide justification for modeling fully inserted control rods (CRs) with the fuel assembly having partially inserted CRs.

Page 5 of Holtec Report, "Commercial Reactor Critical Benchmarks for Burnup Credit," states: "However, in a few cases, the model for the fully inserted CRs was used when the CRs were not

fully inserted, or the model for the partly inserted CRs was used when the CRs were actually fully inserted." The staff understands that modeling partially inserted CRs as fully inserted would produce conservative result for criticality analysis. It is not clear, however, why a model with partially inserted CRs can be used for a fuel assembly having fully inserted CRs. The applicant is requested to provide justification for modeling fully inserted CRs with the fuel assembly having partially inserted CRs.

This information is necessary for the staff to determine if the spent fuel shielded transfer cask meets the regulatory requirements of 10 CFR 50.68, 10 CFR 72.24(c)(3), 72.24(d), and 72.124.

Response to RAI 4-20

The quoted section of the text from Holtec Report on the Commercial Reactor Criticals refers to modeling details outside and above the active region of the fuel. Within the active region of the fuel, CRs are modeled accurately as stated at the end of the paragraph that contains the quoted text, which reads "Note that the insertion position of the CRs in the active fuel region is always modeled correctly to about 1 cm". Since the modeling discrepancy occurred outside of the active region it was considered not significant.

In this context, please also see the response to RAI 4-16 on the purpose and relevance of the CRC benchmarks for the safety analyses.

NRC RAI 4-21

[deleted]

REFERENCE

Reference [K.C] in the Responses to the RAIs on Chapter 4 refer to the corresponding reference in the licensing report, and is also repeated here for completeness

[K.C] HI-951251, Latest Revision, "Storage, Transport, and Repository Cask System (HI-STAR Cask System) Safety Analysis Report", USNRC Docket 71-9261

CHAPTER 5 - THERMAL-HYDRAULIC EVALUATION

NRC RAI 5-1

Verify that all thermal properties used in the analyses properly cover the expected temperature range during normal, loading, off-normal and accident conditions. (TCB)

For example, Table 5.2.9 of the SAR includes thermal properties of steam which appear to be for superheated steam. Thermal evaluation results provided in the SAR do not indicate the presence of superheated steam. The use of these properties to calculate the effective thermal properties of the steam gap could overestimate heat transfer and could result in incorrectly calculated temperatures.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-1

It should be noted that the thermal conductivity of superheated steam is lower than saturated water vapor [5-1.1]. Since superheated steam properties were used for the steam gap within the STC, the effective thermal conductivities of the steam gap used in the licensing basis calculations were conservative. Therefore, the predicted temperatures and pressure presented in the licensing report were conservative.

The thermal calculations presented in Chapter 5 of the licensing report have been revised and now use the properties of materials over the full temperature range during normal, loading, off-normal and accident conditions. The steam properties have also been revised to use the thermal properties of saturated steam. The results of the modified analyses show small increases in the predicted peak cladding temperature for all the licensing basis analyses.

In summary, the thermal properties of all materials in the licensing report have been revised to cover the expected temperature range.

Reference
[5-1.1] "Fundamentals of Heat and Mass Transfer", Frank P. Incropera and David P.DeWitt, fourth edition.

NRC RAI 5-2

Justify the emissivity value of the water surface used to calculate the effective thermal conductivity of the steam gap region inside the STC. Provide also the reference where the water emissivity is obtained. (TCB)

The staff needs to verify realistic parameters are used to properly characterize regions represented by effective properties which would assure a realistic or conservative representation of the heat transfer characteristics of the system. Table 5.2.1 provides the thermo-physical property references for all materials used. Note 1 on this Table states the water emissivity is not reported, as radiation heat dissipation from these surfaces is conservatively neglected. However, water emissivity is used to calculate effective thermal conductivity of air and steam spaces.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-2

The emissivity of water was used in the analyses but was not reported in the licensing report. The emissivity of water used in the analysis was 0.96 [5-2.1] and this value has been added to Table 5.2.4 of the licensing report. It should be noted that the emissivity of water used to determine the radiation heat transfer [PROPRIETARY TEXT REMOVED]. Note that the footnote in Table 5.2.1 has been modified to clarify that water is opaque to thermal radiation.

Reference:
[5-2.1] D. Q. Kern, "Process Heat Transfer", McGraw Hill Kogakusha, (1950).

NRC RAI 5-3

Clarify why the SAR reports lower temperatures compared to the thermal evaluation included in the original application. (TCB)

The staff needs to verify how the updated thermal evaluation resulted in lower temperatures (e.g., if model conservatisms were removed, provide adequate justification, etc.)

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-3

The temperatures reported in the licensing report (Revision 3) are lower than those reported in the original application (Revision 1) due to the following reasons:

1. [PROPRIETARY TEXT REMOVED].

2. Air below the STC lid was replaced with steam. The emissivities of surfaces inside the STC remained unchanged. The thermal conductivity of steam is lower than that of air. However, the effective thermal conductivity of space below the STC lid was lower by a small amount (less than 1%) when this space is occupied by steam instead of air. This change had no significant impact on temperatures.

3. The net heat balance of the system in the original application was higher than [PROPRIETARY TEXT REMOVED] which lead to higher temperatures.

In summary, the changes mentioned above caused the reported drops in the predicted temperatures. Please note that the temperatures and pressures have changed again due to the revised thermal analyses reported in the licensing report. The reasons for these changes are provided in the responses to RAIs 5-1, 5-4 and 5-10.

NRC RAI 5-4

Revise all thermal analyses to assure energy balance has sufficiently converged to provide assurance calculated temperatures are representative of all applied thermal loads. (TCB)

When performing analysis audits of some of the calculations, the staff noticed there is a heat imbalance of about 7% which indicates the calculation is not fully converged which could result in lower non-conservative temperatures.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-4

The thermal analyses have been revised to assure that the energy balance has sufficiently converged. To ensure that steady state conditions were achieved in these analyses a sufficiently high number of iterations was utilized and the peak temperatures of the fuel and STC components were tracked until it was concluded that the peak temperatures of components did not change significantly with any further iterations.

To achieve a greater heat balance the following changes were made in the analyses:

1. [PROPRIETARY TEXT REMOVED].
2. [PROPRIETARY TEXT REMOVED].

[PROPRIETARY TEXT REMOVED]These analyses also incorporate the changes made to the material properties in accordance with the response to RAI 5-1. The peak cladding temperatures along with the heat balances are reported below:

Description	No. of cells in the radial direction of the Annulus	No. of cells in the radial direction of Water Jacket	No. of cells in the axial direction of Fuel	Heat Balance on outer surfaces of the HI-TRAC	Peak Cladding Temperature (°C)
Mesh 1[Note1]	[PROPRIETARY TEXT REMOVED]				
Mesh 2					
Mesh 3					
Note 1: This mesh was used in the licensing basis calculations with design basis heat in revision 3 of the licensing report. [PROPRIETARY TEXT REMOVED]					

The results show that the mesh used for the calculations in Revision 3 of the licensing report predicted higher temperatures. The peak temperatures due to mesh refinement and with a better heat balance are lower than the peak temperatures obtained for the mesh used in Revision 3 of the licensing report. All the steady state cases presented in this licensing report, therefore, continue to use Mesh 1. For cases that use [PROPRIETARY TEXT REMOVED]model (e.g. simultaneous loss of water in the HI-TRAC annulus and the water jacket), [PROPRIETARY TEXT REMOVED]. For all the transient analyses, Mesh 3 is used since it results in a mesh independent solution.

Reference:
[5-4.1] Section 25.4.3 of FLUENT 6.3 User's Guide.

NRC RAI 5-5

Perform the transient pressure rise for at least 48 hours to make sure the pressure increase used to monitor for fuel misload is approaching steady state. (TCB)

Figure 5.3.2 of the SAR indicates a pressure increase of about 4.2 psi is expected for the first 24 hours. The staff needs to have additional assurance the pressure increase is converging to the steady state value which would indicate a safe onsite transfer may proceed.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-5

A transient STC pressure rise calculation has been performed for 48 hours assuming a design basis heat load. A mesh independent solution has been achieved by performing mesh sensitivity studies as discussed in the response to RAI 5-4. Since Mesh 3 provides a mesh independent solution, the transient pressure rise has been determined using this mesh. The pressure rise curve is shown in Figure 5.3.2 of the licensing report and in Figure 1 below. The STC loaded with fuel assemblies and filled with water has a high thermal inertia. [PROPRIETARY TEXT REMOVED]. It is observed that the rate of increase of the temperature inside the STC and the STC pressure with time is extremely small over the 24-hour duration analyzed, of the order of [PROPRIETARY TEXT REMOVED]. Therefore, this provides assurance that the solution will not be altered by the use of a smaller time step. The above mentioned features of the thermal model ensure that the uncertainties in the transient analysis are small and will not alter the solution as presented.

[PROPRIETARY TEXT REMOVED]

Figure 1: 48-hour STC Pressure Rise under Design Basis Heat Load

It should be noted that as a result of the reanalyzes performed in response to RAI 5-4 the 24 hour expected STC pressure rise has decreased from [PROPRIETARY TEXT REMOVED]. The pressure increase at the end of 48 hours is [PROPRIETARY TEXT REMOVED]. The rate of pressure rise at the end of 48 hours indicates that a steady state condition has not been reached yet; however, the rate of pressure rise over the 48-hour duration is only approximately [PROPRIETARY TEXT REMOVED]. If it is very conservatively assumed that this rate of pressure rise continues for an additional 28 days (30 day VCT breakdown accident condition) then the total STC pressure rise would be limited to approximately [PROPRIETARY TEXT REMOVED]. As the initial STC pressure would be sub atmospheric the maximum expected STC pressure would be [PROPRIETARY TEXT REMOVED] which is lower than the STC design pressure limit of 50 psig for normal operations and significantly below the applicable accident limit of 90 psig. As noted this is a very conservative evaluation as it neglects the increase in heat loss from the system with increasing STC water temperature. A more realistic analysis shows, under design basis assumptions as reported in Table 5.3.2 of the licensing report, that the steady state STC pressure would be [PROPRIETARY TEXT REMOVED]demonstrating significant additional margin to the STC design limits.

In summary, even though the STC pressure increase has not converged to the steady state value the rate of pressure at 48 hours is small indicating that a safe onsite transfer may proceed.

This information has been added to Section 5.3.4 of the licensing report.

NRC RAI 5-6

Perform a transient pressure rise calculation for the case when a possible misload has occurred based on heat loads representative of the IP3 spent fuel fuel inventory to assure the pressure rise can be managed by the licensee's in-place operating control procedures. (TCB)

The transient pressure rise provided in the SAR applies only for design basis heat load. The staff needs to have assurance the licensee has the capability to implement corrective actions, if needed, in the case of an occurrence of a misload.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-6

[PROPRIETARY TEXT REMOVED]. The heat load for a fuel assembly that has been subcritical for at least 90 days with a bounding burnup of 55,000 MWD/MTU, was determined to be [PROPRIETARY TEXT REMOVED] [5-6.1]. Proposed TS 4.1.5.4 dictates that transfer operations shall only be conducted when the irradiated fuel assemblies in the Unit 3 spent fuel pit have been subcritical for at least 90 days.

The hypothetical accident condition assumes that a single fuel assembly with a heat load of [PROPRIETARY TEXT REMOVED]is mis-loaded in the STC basket. Mesh 3 has been used in this analyses (as discussed in the response to RAI 5-4) since it provides a mesh independent solution. The misloaded fuel assembly was conservatively placed in the inner cell region to maximize STC temperatures and, consequently, pressure rise inside the STC. All other STC basket storage locations were assumed to be at their design basis heat load. [PROPRIETARY TEXT REMOVED]. The transient pressure rise inside the STC for the first 8 hours after loading, both for the design basis heat load and the severe mis-load accident conditions, are presented in Figure 1 below and included as Figure 5.3.3 in the licensing report. Likewise, the rate of change of STC pressures are presented in Figure 2 below and included as Figure 5.3.4 in the licensing report. Figure 2 shows that a rate of change of [PROPRIETARY TEXT REMOVED]can be used to differentiate between a design basis heat load and a severe misload shortly after commencement of the STC pressure rise surveillance.

The response to RAI 1-1 provides justification that the [PROPRIETARY TEXT REMOVED]STC pressure rise acceptance criterion, over a rolling 4 hour period, ensures that a severe fuel misload would be detected by the pressure monitoring system within 8 hours. If a severe fuel misload is detected, corrective actions that could include unloading fuel from the STC to the spent fuel pool would be performed. These actions are described in Subsection 10.2.3 of Chapter 10 of the licensing report and are included in the TS. These actions provide assurance that the capability exists to implement corrective actions, if needed, in the case of an occurrence of a severe fuel misload. This information has been added to Section 5.3.4 of the licensing report.

Reference:

[5-6.1] "Spent Nuclear Fuel Source Terms", Holtec Report HI-2022847, Revision 5 (Holtec Proprietary).

[PROPRIETARY TEXT REMOVED]

Figure 1: Comparison of STC Pressure Rise between Design Basis Heat Load and Severe Fuel Misload Accident

[PROPRIETARY TEXT REMOVED]

Figure 2: Comparison of Rate of Change of STC Pressure with Time between Design Basis Heat Load and Severe Fuel Misload Accident

NRC RAI 5-7

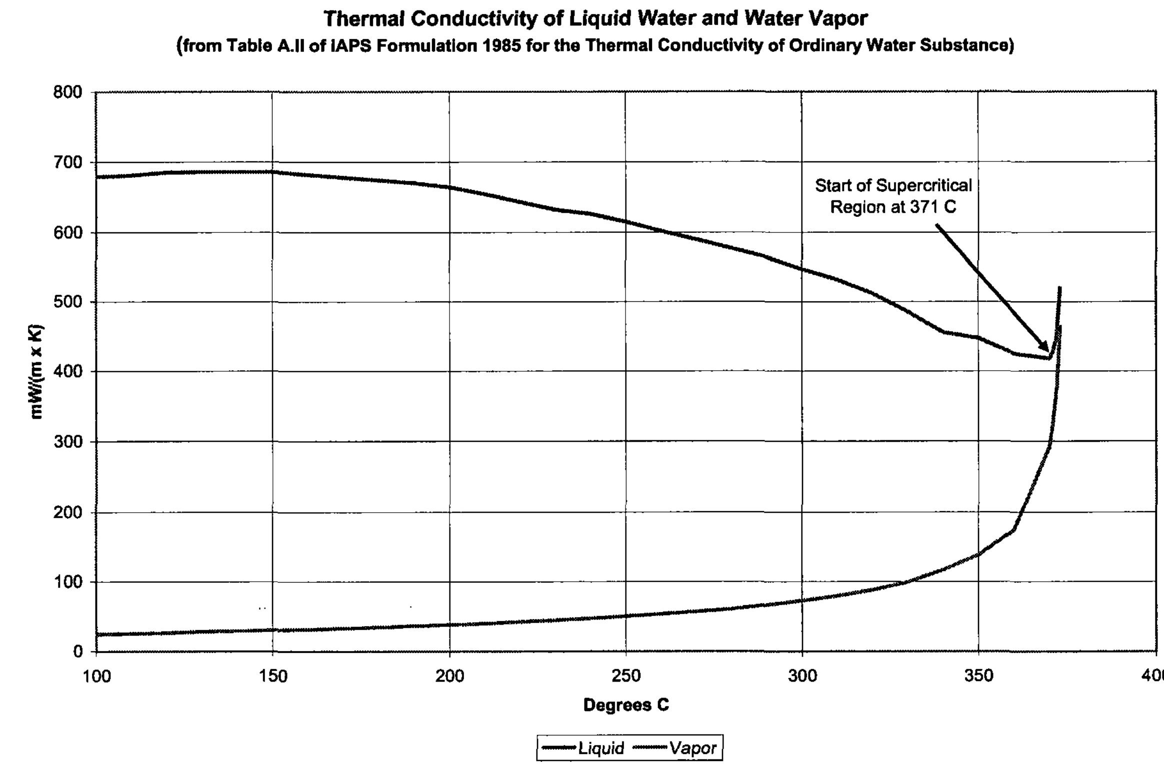
Review the thermal properties used to determine the thermal conductivity of saturated water vapor as a function of temperature. (TCB)

Page B-7 of Holtec report HI-2084146 provides saturated vapor thermal conductivity as a function of temperature. Thermal conductivity values tabulated at 371 C and 372 C appear to correspond to saturated liquid water. This may increase the effective conductivity at these temperatures.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-7

The thermal properties used to determine the thermal conductivity of saturated water vapor have been reviewed and it has been confirmed that the thermal conductivity values on page B-7 of Holtec report HI-2084146 are for saturated vapor. Table A.II of the listed reference provides separate values for saturated liquid and saturated vapor, both as functions of temperature. As the temperature approaches the supercritical region (beginning at 371°C) the saturated vapor conductivity rises to approach that of saturated liquid. This is readily apparent in the following plot of the data provided by the reference. There is typically little difference in thermal conductivity between saturated liquid water and saturated water vapor in the supercritical region.



Reference:
[5-7.1] "IAPS Formulation 1985 for the Thermal Conductivity of Ordinary Water Substance."

NRC RAI 5-8

Clarify how the STC pressure rise would be controlled if, based on the thermal analysis for this configuration, it is predicted that water would be boiling under normal conditions of transfer. (TCB)

Based on auditing of some of the thermal calculations (see also RAI 5-4 above), the staff noticed there is not adequate convergence in the heat balance. For a properly converged solution (maximum temperatures reached, adequate heat balance), the licensee could predict higher temperatures with water reaching the boiling point.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-8

The thermal analyses have been re-performed as documented in the revised licensing report to address the RAIs above and have achieved greater numerical convergence and a better heat balance. The results of these analyses for the normal conditions of transfer show that bulk boiling of the water within the STC will not occur.

The water and vapor (no air) inside the STC form a closed self-equilibrium system. The water

surface temperature is always equal to the local saturation temperature at the water surface. The local water saturation temperature below the water surface increases with water depth due to an increase in the local pressure.

In a small region in the center of the STC, that is less than 3 feet deep, the water temperature is slightly (~2°C) higher than the local saturation temperature. The phenomenon of boiling may occur in this small region. However, it is to be noted that the fuel in this region is inactive part of the fuel assembly. The bubbles due to local boiling will rise upwards and condense due to lower temperatures near the water surface. This may result in a slight increase in the surface temperature of water inside the STC, thereby resulting in a slight increase in the STC pressure. Then the local boiling will be suppressed due to pressure increase inside the STC. Furthermore, the enhancement of heat transfer due to phase change will decrease PCT and volume of water which exceed the local saturation temperature. Since this is a closed system, the water temperature and vapor pressure will reach an equilibrium state.

It should be noted that the phase change is not modeled in the thermal analyses and therefore the actual water surface temperature may be a little higher than the computational result, but it cannot go beyond the PCT inside the STC. Consequently, the maximum possible STC pressure is the saturation pressure corresponding to peak computational temperature inside the STC. The saturation pressure corresponding to the peak temperature within the STC during design basis normal transfer conditions was determined to be [PROPRIETARY TEXT REMOVED], which is still significantly lower than the design STC pressure of 50 psig specified in Chapter 3 of the licensing report.

NRC RAI 5-9

Describe the code assessment conducted for the software models used to calculate the thermal response of the STC and HI-TRAC. (SBPB)

Section 5.3, "Thermal Evaluation of Fuel Transfer Operation," of the SAR included a description of the three-dimensional modeling used for thermal evaluations. The discussion described the general modeling and some conservative assumptions included in the model. In addition, the response to NRC staff requests for additional information provided in Attachment 1 to the letter dated October 5, 2010, indicated that the models were used for similar evaluations of spent fuel stored in spent fuel pools. However, in this application the model was used to evaluate conditions involving natural convection heat transfer in air environments and to evaluate radiation heat transfer. Section 15.0.2, "Review of Transient and Accident Analysis Methods," of the NRC Standard Review Plan (NUREG-0800) describes specific areas of review for models used for transient analyses. The review areas include code assessment, which the staff described as a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

Provide a code assessment for the accident scenarios involving natural convection heat transfer in air environments and radiation heat transfer that demonstrate the adequacy of the model for those scenarios.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-9

FLUENT is the principal code deployed to evaluate scenarios involving natural convection heat transfer in air environments and radiation heat transfer. This code was benchmarked in the late 90s to evaluate similar scenarios in support of Holtec's dry storage applications licensed under 10CFR72 (on-site storage) and 10CFR71 (transportation). FLUENT was also deployed to evaluate scenarios involving natural circulation of water in spent fuel pools to demonstrate thermal-hydraulic requirements for safe storage of spent nuclear fuel in water environment. A partial list of Dockets under which the code has been used for safety analysis of casks is as follows:

Application	Docket No.
HI-STAR 100 Storage	72-1008
HI-STAR 100 Transportation	71-9261
HI-STAR 60 Transportation	71-9336
HI-STAR 180 Transportation	71-9325
HI-STORM 100 Storage	72-1014
HI-STORM FW	72-1032
St. Lucie Unit 1	50-425
Diablo Canyon Units 1 and 2	50-275, 50-455
Beaver Valley Unit One	50-334

The FLUENT code is validated using data from PNNL tests conducted with the TN-24P prototype cask loaded with irradiated SNF. The FLUENT code is benchmarked with independent COBRA-SFS thermal analysis of the HI-STORM cask by PNNL. The code validation and benchmark work is archived in the QA validated Holtec report, "Topical report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Scale Cask Test Data", HI-992252, Rev. 1. A succinct description of the code validation work and results confirming the adequacy of the FLUENT computer code is provided below.

The TN-24P test cask is a 24-cell prototypical metal cask designed to store PWR fuel. The cask thermal tests were conducted under an EPRI sponsored work[1] by Pacific Northwest Lab (PNNL) and Virginia Power Company (VPC) at the Idaho National Engineering Lab (INEL). The test cask was loaded with irradiated PWR fuel from the Surry reactor. The cask was tested with significant decay heat (20.6 kW) under an array of scenarios wherein one or more of the principal modes of heat transfer - conduction, natural convection and radiation - were active. The test scenarios are listed below.

Case no.	Scenario	Principal Heat Transfer Modes
1.	Cask vertical, Vacuum	Radiation
2.	Cask horizontal, Vacuum	Radiation, Conduction
3.	Cask horizontal, Nitrogen filled	Radiation, Conduction
4.	Cask vertical, Helium filled	Radiation, Conduction
5.	Cask vertical, Nitrogen pressurized	Natural Convection, Radiation
6.	Cask vertical, Helium pressurized	Natural Convection, Radiation

The principal results obtained from the validation study are summarized below:

Note 1 "The TN-24P PWR Spent Fuel Storage Cask: Testing and Analyses", EPRI NP-5128, April 1987.

[PROPRIETARY TEXT REMOVED]

The above results support the conclusion that FLUENT code provides conservative assessment of cask thermal condition and confirms its suitability for safety analysis of casks loaded with irradiated SNF.

The COBRA-SFS analysis of the HI-STORM storage cask by PNNL is included as Attachment 1 of the Holtec benchmarking report discussed above. The principal results of the FLUENT benchmarking study are graphed in the Figure below. The Figure provides additional assurance of the adequacy of the FLUENT code to yield conservative assessment of cask thermal condition.

[PROPRIETARY TEXT REMOVED]

Comparison of FLUENT Peak Clad Temperature Solution with PNNL Result

NRC RAI 5-10

Discuss the effect of the centering assembly on heat transfer from the STC to the HI-TRAC and then to the atmosphere during scenarios involving loss of water from the HI-TRAC annulus. (SBPB)

Holtec Report No. HI-2084146, "Thermal Hydraulic Analysis of IP3 Shielded Transfer Canister," provided information about analyses of various scenarios evaluated for passive rejection of decay heat to the environment. Section 7.4 discussed the analysis of the simultaneous loss of water from the water jacket and HI-TRAC annulus, and mentioned the use of the Discrete Ordinates radiation heat transfer model in FLUENT. This analysis credited radiation and natural convection heat transfer within the annular space between the STC and the HI-TRAC.

The annular space between the STC and the HI-TRAC contains the HI-TRAC/STC centering assembly. The staff believes the centering assembly may adversely affect heat transfer during the scenario involving loss of water from the HI-TRAC annulus because it would interfere with internal radiation heat transfer and reduce internal natural circulation air flow. Describe in detail the internal natural circulation in air and radiation heat transfer models for the loss of water from the HI-TRAC annulus scenario. Specifically address how the effect of the centering assembly was incorporated in the models.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 5-10

A thermal model that explicitly includes the HI-TRAC/STC centering assembly (12 total) has been developed to evaluate its effect on the temperature and pressure of the system. The thermal analysis of the scenario involving the loss of water from the HI-TRAC annulus and water jacket has been performed with the centering assembly explicitly included in the model. An emissivity of 0.2 [5.10-1] was used for the Aluminum centering assembly surfaces. [PROPRIETARY TEXT REMOVED]. A steady state analysis was performed and the component temperatures, STC and HI-TRAC internal pressures are reported in Tables 5.4.5

and 5.4.6 of the licensing report. The results show [PROPRIETARY TEXT REMOVED] in peak cladding temperature due to the presence of centering assemblies. The STC internal pressure [PROPRIETARY TEXT REMOVED]. The effect of including the centering assembly on temperatures and pressure is small compared to their respective safety margins to the accident limits, which are provided in Tables 3.1.1 and 3.2.1 respectively.

The details of the analysis and its results are summarized in the table below and have been included in Section 5.4.3 of the licensing report.

Reference:

[5-10.1] "Fundamentals of Heat and Mass Transfer", 4th Edition, F.P. Incropera and D.P. DeWitt, John Wiley & Sons, Inc., New York, 1996.

<u>SIMULTANEOUS HI-TRAC ANNULUS AND JACKET WATER</u>
<u>LOSS ACCIDENT TEMPERATURES</u>

[PROPRIETARY TEXT REMOVED]

CHAPTER 6 - STRUCTURAL EVALUATION OF NORMAL AND ACCIDENT CONDITION LOADINGS

6-1. Provide the technical basis for ignoring the water inside the hollow aluminum tubes when modeling the tube assembly for mitigating side impact effects associated with the non-mechanistic tipover of loaded HI-TRAC cask analysis. (SMMB)

SAR Section 6.2.8, Rev. 3 states, "the tip-over of the transfer cask has been carried out without considering the fluid coupling (cushioning) effect of water inside the annular space around the STC and within the centering assembly tubes." It is not entirely clear why water inside of the hollow aluminum tubes is not included in the model, given that the water incompressibility may add significant stiffness to the tubes during a tipover event.

This information is required to demonstrate that the system can withstand the worst-case loads and successfully preclude an unacceptable release of radioactive materials to the environment, in compliance with GDC 61 and 10 CFR 72.122.

Response to RAI 6-1:

The reviewer is correct in the observation that neglecting the effect of water inside the impact limiter tubes may underestimate the crush resistance of the tubes. It is indeed true that a closed end tube full of water will be considerably stiffened by the captive water because water is essentially incompressible. For this reason, the ends of the impact limiter tubes in the HI-TRAC system are designed to be fully open and un-constricted so as to permit free flow of water out of the tube from both ends if the tube were to be squeezed by a lateral force. Nevertheless, under the postulated tip-over scenario (Load case 9 in Section 6.2.8 of the Licensing Report) a part of the kinetic energy of the STC will be expended in hydrodynamic squeezing of the water inside the impact limiter tubes during their crushing under the inertial momentum of the STC's internals. The hydrodynamic squeezing of water translates to a net increase in velocity head of the water. It is intuitively obvious that the velocity of the water stream escaping from the tube

ends will increase as the opening begins to ovalize under the crush load. Thus the kinetic energy associated with the escaping water will increase as the crushing of the tube ends proceeds. This phenomenon continues until the crushing process stops. If the ends of the tube were to close completely then the apparent stiffness of the tube against crushing will undoubtedly increase markedly. However, as can be seen from Figure A (extracted from the LS-DYNA run files), the extent of crushing is localized near the top end of the impacted tubes (only one tube is subject to any visible crushing) and the extent of crushing, indicated by the reduction in the diameter is less than 4% (conservatively) of the overall tube length. An estimate of the hydrodynamic consequence of this local deformation can be computed using the procedure outlined in Levy & Wilkinson[2] which suggests the use of the principles of classical hydrodynamics to estimate the associated expenditure of mechanical energy.

Following Levy, et al, a first order estimate of this effect can be obtained by the simple expedient of computing the energy extracted from the system during the crush event by the expulsion of the water as the STC advances towards the HI-TRAC surface.

The total volume of water expelled from the two ends of the aluminum tube, V_e, is obtained by subtracting the crushed inner space of the tube from the uncrushed (virgin) volume. The velocity of ejection of water from the tubes, v, is given by

Time of crush (Figure B)	$\tau = 0.043$ sec
Inside area of tube cross-section	$A = \pi (6.5 \text{ in})^2/4 = 33.18 \text{ in}^2$
Height of water in the tubes in vertical config.	L = 179.75 in
Total volume of water inside a single tube	$V = L \times A = 5964 \text{ in}^3$
Volume of water expelled from the tube (approx. 4% of length "L", Figure A)	$V_e = 238.6 \text{ in}^3$
Density of water (approx.)	$\rho = 62.4 \text{ lbf/ ft}^3 = 0.036 \text{ lbf/ in}^3$
Average open area at top end	$A_{av} = A/2 = 16.59 \text{ in}^2$
Velocity of the water	$v = V_e/(A_{av}\, \tau) = 334.5$ in/sec.

where τ is the duration of crush, and A_{av} is the time-averaged open area at the top end of the tube through which water is squeezed out. The calculated value for A_{av} conservatively assumes that the top end of the tube is completely crushed (A = 0) at time τ. Furthermore, the above velocity calculation takes no credit for any exit flow of water through the bottom end of the tube even though it is substantially open.

The total energy E carried away by the squeezing of water is given by,

$$E = \frac{\rho\, V_e^{\,3}}{2\, A_{av}^{\,2}\, \tau^2 g_c} = 1243 \text{ lbf-in}$$

It is seen from the above, the energy extracted by the expulsion of the water inside the tube is miniscule (less than 0.06%) compared to the total kinetic energy of 2.23e6 lbf-in (see Figure C) of the STC at the onset of the impact, which must be dissipated during the impact event.

Thus it is concluded that the stiffening effect caused by the energy required to expel the

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2 See page 344 "Component Element in Dynamics by Levy & Wilkinson, McGraw Hill, 1976".

contained water from the tubes to satisfy (hydraulic) continuity is extremely small (a second or higher order effect) for the specific case of the HI-TRAC tip-over event.

Based on the above, it can be concluded that the effect of neglecting the stiffening effect of water in the tubes is inconsequential and is more than offset by other explicit conservatisms in the model such as the assumption of a rigid STC shell and doubling of the HI-TRAC's radial plate thickness subject to direct impact, among others, as discussed in the Licensing Report, Table 6.2.8.2.

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PROPRIETARY FIGURES REMOVED

CHAPTER 7 – SHIELDING DESIGN AND ALARA CONSIDERATIONS

The proposed amendment seeks to perform a wet transfer of spent fuel from the IP3 SFP to the IP2 SFP using an STC. The STC functions as a transfer cask in these operations and may be considered a lightweight transfer cask since it is designed for a limited-capacity crane that cannot handle an approved transfer cask like those used for approved spent fuel dry storage systems. The STC presents some unique shielding and radiation protection considerations, with significantly higher dose rates for the proposed allowable contents. The STC must be used in conjunction with a HI-TRAC transfer cask, which provides additional shielding so that transfers between FSBs as well as preparation for the transfer and unloading of the STC can be done in a safe manner. This is a new, first-of-a-kind review for a system and operations of this kind.

Normally, transfer casks provide sufficient biological radiation shielding such that workers may safely be in the vicinity of the transfer cask. This does not appear to be the case with the STC design. The staff is not only concerned about occupational doses during normal, off-normal, and accident conditions, but also public doses. Since this is an amendment under 10 CFR Part 50 and the action is limited to the Indian Point Energy Center site, considerations are given with regard to the site features. Still, the staff must ensure that enough controls are in place to provide reasonable assurance that both the public and occupational dose limits in 10 CFR Parts 20, 50 and 100 (via compliance with the intent of Part 72 limits) are not exceeded.

The following RAIs are geared towards obtaining enough information so that the staff may make a determination regarding whether there is reasonable assurance that the STC may be used safely and in accordance with the regulations. The staff is particularly concerned about a potential off-normal event involving either the hang-up of the crane or a malfunction of the remote handling equipment (if used). Crane hang-ups are not uncommon, especially when cranes are loaded with weights approaching their capacity limits. Additionally, the staff is asking for clarification and/or additional information to ensure that the license, TS, and licensing report each contain the appropriate level of information needed to control the design basis for this unique transfer system.

General Response to Shielding Design and ALARA RAIs

Entergy recognizes the high importance of the shielding design and ALARA considerations for the STC, and the concerns about doses and dose rates from the system. In response to those dose concerns for the STC, the previous calculations and results have been reviewed. The

conclusion of the review is that some of the high dose rates that were previously reported were the result of overarching conservatisms, and are not indicative of the doses to be expected from the system. An effective radiation protection approach for the STC can only be based on meaningful calculated doses and dose rates, such that dose rates are conservative but at the same time realistic. Accordingly the analyses have been revised to be more consistent with this approach. A further enhancement is that the sealing system of the STC will be tested to ANSI 14.5 leak tight criteria, which eliminates concerns regarding effluent doses from the STC.

Following is a brief summary of revisions to the dose analyses, while the individual RAI responses below and the revised licensing report provide additional detail. Note that for some aspects of the analyses, a more realistic approach is taken for the doses and dose rates from the bare STC, while more conservative assumptions are made for dose rates from the STC located inside the HI-TRAC. The more conservative assumptions for the HI-TRAC simplify the analyses, and can be used since dose rates from the HI-TRAC are low:

- Design Basis Assemblies: The STC is designed for regionalized loading in both the thermal and shielding designs. However, while the previously proposed TS clearly defined the heat load limits for each region, only a single upper bound burnup and a single lower bound cooling time were defined.. The combination of those values in the shielding analysis resulted in dose rates that were up to a factor 5 higher than those for any loading configuration that could actually meet the thermal requirements. To address this, the revised proposed TS now contain explicit burnup, cooling time and enrichment limits for the two regions of the basket, and those limits are the sole basis for the dose evaluations. Assembly heat load, burnup and cooling time are selected in a consistent fashion for each of the two regions, and based on the inventory of fuel assemblies to be transferred.

- Water in the STC: The previously reported dose analyses assumed unborated water inside the STC. However, for the STC, a minimum soluble boron level of 2000 ppm is specified in the proposed TS. This boron level substantially reduces the neutron dose from the cask, and is now used in all STC dose analyses[3], when the STC is outside of the HI-TRAC. Unborated water is used for the HI-TRAC dose evaluations.

- Azimuthal dose variations: Azimuthal dose variations were evaluated for the STC, but only the peak dose rates were reported in the licensing report. Minimum, maximum and average dose rates are now reported, together with axial and radial variations indicating in which areas minimum and maximum dose rates are to be expected from the bare STC. However, maximum dose rates considering the azimuthal variance are reported for HI-TRAC.

- Material Thicknesses: Since the STC has been manufactured, as-built material thicknesses are used in the dose analyses for the bare STC, while minimum material thicknesses are used for the analyses of the STC inside the HI-TRAC.

3 Note that no changes were made to the criticality calculations in that respect, i.e. soluble boron in criticality calculations is only credited under accident conditions, but not under normal conditions.

- Effluent Dose: The STC seal system will be tested as leak-tight per ANSI 14.5. Consequently, effluent is no longer expected, and effluent dose rates are no longer considered in the analyses.

In summary, more realistic dose rates are evaluated for the bare STC, while more conservative approaches are taken for the HI-TRAC dose evaluations. Additionally, the maximum dose locations for the bare STC are clearly identified. In general, STC dose rates are a factor of 3 to 5 less than the dose rates reported for the bounding loading combination of the previous submittal. Dose rates for the bare STC are therefore comparable to those for standard transfer casks used in the industry. For example, the average dose rate on the radial surface of the STC at mid height is about 3 rem/hr, while the corresponding value for the HI-TRAC 100 in HI-STORM 100 FSAR is about 4 rem/hr, both with design basis fuel. However, actual field measurements are expected to be much lower than the estimated dose rates based on the experience with the HI-TRAC. Additionally, in response to the shielding design RAIs Entergy is proposing controls that will provide reasonable assurance that both the public and occupational dose limits in 10 CFR Parts 20, 50 and 100 (via compliance with the intent of Part 72 limits) are not exceeded. Compliance with the intent of Part 72 is demonstrated by adopting the Part 72 numerical limits where appropriate.

NRC RAI 7-1

Justify the homogenization of the fuel assembly with the moderator in the shielding model. (CSDAB)

Question 7-16 in the first round RAI dealt with accounting for neutron multiplication and its effect on dose rates with an appropriate geometry. Modeling assumptions regarding fissile material and moderator can influence the neutron multiplication of a system, with homogenized models under-predicting multiplication occurring in heterogeneous systems of fissile material and moderator. The applicant's response does not address this question and instead assumes a k-effective value to define multiplication without consideration of how k-effective differs between a heterogeneous and a homogeneous system of the same materials or how the different systems behave from a shielding perspective. Further, the response only considers the loaded HI-TRAC; it should also include the impact on dose rates for the loaded STC outside of the HI-TRAC.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR50.90 and 50.34a(c) and the intent of 10 CFR 72.104, 72.106(b) and 72.126(a).

Response to RAI 7-1

To further justify the homogenization of the fuel assembly with the moderator in the shielding model, additional studies have been performed. The cases in the study, including the design basis case, are described below. For each case, the specific modeling details that are important to the neutron multiplication effect are specified. Note that all other aspects of those calculations are identical between the cases, i.e. are identical to the design basis calculations.

A
- Design Basis Case
- Homogenized Fuel Model
- Fuel assembly assumed fresh, 3.6 wt% Enrichment

B - Heterogenuous Fuel Model
- Fuel assembly assumed fresh, 3.6 wt% Enrichment

C - Heterogenuous Fuel Model
- Fuel assembly modeled as spent fuel, 45 GWd/mtU, 5 years cooling time

The average STC and HI-TRAC surface and 1 m dose rates are as follows:

Location	Homogeneous Fuel Model, Fresh Fuel (mrem/hr) (Case A)	Heterogeneous Fuel Model, Spent Fuel (mrem/hr) (Case C)	Heterogeneous Fuel Model, Fresh Fuel (mrem/hr) (Case B)
STC			
Surface	586.4	350.7	586.8
1 m	142.7	87.5	144.0
HI-TRAC			
Surface	0.97	0.32	Not evaluated
1 m	0.38	0.15	Not evaluated

Note that the total dose rates reported in the above table only comprise of neutron and captured gamma, since dose contributions from other sources (fuel gamma, Co-60 gamma) are essentially unaffected by the fuel modeling approach.

The results show that a heterogeneous fuel model results only in a marginally higher dose rate when the fresh fuel assumption is maintained. Typically, higher multiplication factors, and thus higher neutron dose rates, are expected from heterogeneous models. However, this is apparently offset by the presence of the soluble boron credited in the models. Accounting for fuel burnup instead of assuming fresh fuel results in a significant reduction in the neutron dose rate, although the relative effect on the total dose is much less since the neutron dose is not the dominant dose contributor. Overall, this shows that using a homogenized fuel model with fresh fuel is appropriate and conservative. This study is reported in Reference [L.G].

NRC RAI 7-2

Provide a dose rate evaluation for the STC with all shielding materials at their minimum thickness specified in the proposed TS, Appendix C, Part I, Section 1.0, addressing the impacts on the occupational and public doses. (CSDAB)

Based upon the sample input file provided as part of the RSI response, the current dose rate calculations are based upon nominal steel dimensions and minimum lead thickness. While not as strong as lead, steel is still a significant shield material. For example, the half-value thickness of steel for 1.0 MeV photons is about 1.5 cm and for 1.5 MeV photons is about 1.8 cm; thus, the difference in steel between a minimum thickness present and a nominal thickness

present is nearly 1 half-value thickness at this gamma energy; indicating the potential for significant differences in dose rates. Given the high dose rates for the STC with the representative loading, the impact on dose rates could be significant and lead to higher estimated occupational and public doses and/or the need to further modify operations to reduce doses and keep them as low as reasonably achievable (ALARA).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 20.1301(a) and (b), and the intent of 10 CFR 72.104 and 72.106.

Response to RAI 7-2

With respect to manufacturing tolerances of the STC, specifically the tolerances of the radial steel, it is important to note that larger than usual tolerances were selected to provide the flexibility that is necessary to ensure the crucial weight limit of the system is met. As the STC has now been already manufactured in the shop, as-built dimensions are used for dose evaluations when the STC is outside of the HI-TRAC. The only difference found between the as-built and the nominal dimension is the thickness of the STC inner shell. Nominal dimension of the STC inner shell is 1", whereas the as-built dimension is 3/4" with 3/16" of weld overlay. As machining is performed on this weld overlay, the 3/16" thickness of the weld overlay is neglected. The revised licensing report Tables 7.4.1 to 7.4.8, which report dose rates from the bare STC, assume an STC inner shell thickness of 3/4". However, minimum STC dimensions are applied for all the evaluations when the STC is within the HI-TRAC. HI-TRAC dose rates are reported, with minimum STC dimensions, in Tables 7.4.9 to 7.4.22.

The proposed TS have been revised to include the as built thickness of the STC inner shell. In addition to the manufacturing records for the individual parts of the STC, a simple verification has been performed based on the weight of the as-built STC (empty, without lid) in comparison with the weights that would be consistent with nominal and minimum radial thicknesses. The results of this comparison are as follows:

Condition	**Weight (lbs)**
Nominal Dimensions	44,200
Minimum Dimensions	39,500
As-modeled (Nominal Dimensions, except 3/4" for inner shell)	42,200
As-built (measured)	44,400

The comparison shows that the as-built weight exceeds the as-modeled weight, and is in fact close to the nominal dimensions. The as-built weight is also significantly higher than the weight for the minimum dimensions. Using the as-built dimensions in the model and neglecting the weld overlay as discussed above is therefore appropriate and conservative.

NRC RAI 7-3

Justify the changes to the dose rates reported in the tables in Section 7.4 of the STC licensing report, correcting the reported dose rates and dose evaluations as necessary. (CSDAB)

In response to staff's RAIs, the applicant modified the dose rates reported in the licensing report for the STC and HI-TRAC. However, the changes appear to be inconsistent. For example, the surface dose rates on the STC for the representative loading case are nearly double their previously reported values; however, a number of the dose rates reported at the axial surfaces and at distance from the axial and radial surfaces have either negligibly changed or have decreased compared to the previously reported values. Correct dose rate values should be provided, and changes in dose rates should be adequately explained and justified. The dose evaluations in the report should also be updated, as necessary, to account for the correct dose rates.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 20.1301(a) and (b), and the intent of 10 CFR 72.104, 72.106, and 72.126(a).

Response to RAI 7-3

In the previous submittal, in response to RAI 7-5, azimuthal tallies were added and only the azimuthal peak dose rates were reported for the surface of the STC and at 0.5 m from the STC. This resulted in the observed increase in dose rates on and near the surface of the STC. For all other distances average dose rates were provided which resulted in a negligible or slight change in those dose rates.

In the current submittal, Chapter 7 of the licensing report has been substantially revised and the tables contain new dose rates that reflect the current analysis and are both accurate and conservative. The new submittal now clearly identifies the average dose rates, azimuthal dose rate distributions and peak dose rates, and axial dose rate distributions and peak dose rates.

NRC RAI 7-4

Provide dose evaluations for crane hang-up and other off-normal conditions occurring with a loaded STC outside the SFP and HI-TRAC in the FSBs (both IP2 and IP3), including the information described below, and provide adequately detailed operations descriptions in Chapter 10 for addressing these conditions. (CSDAB)

The STC function is like that of the transfer cask that many spent fuel storage systems use to load fuel from the pool and transfer it to the storage overpack. Modifications made to the licensing report in response to staff RAIs indicate that the STC dose rates (representative loading) are significantly higher than have been analyzed for nearly all spent fuel transfer devices currently approved under 10 CFR Part 72. Given the STC's high dose rates, off-normal events, such as crane hang-ups, may result in conditions that would not be encountered were the crane hang-up to occur with a standard transfer cask. Thus, the applicant should provide dose evaluations for personnel involved in performing operations to recover from a crane hang-up, including manual crane operations and crane repair. These evaluations should include descriptions of personnel actions during these operations, personnel numbers and locations *relative to the STC, duration of each operation segment, and appropriate justification for each* aspect of the evaluation. Additionally, the applicant should provide dose evaluations for other plant personnel (e.g., administrative staff, guards, plant technicians, etc.) and describe the impacts this event would have on operations/activities in site facilities adjacent to, or near the respective units' FSBs. Evaluations should also be performed for members of the public on-site

and at the controlled area boundary, with adequately justified bases and assumptions. Chapter 10 of the licensing report should provide an adequately detailed description of these operations, with which the evaluations in Chapter 7 should be consistent.

This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a) and (b), GDC 61, and the intent of 10 CFR 72.104 and 72.126.

Response to RAI 7-4

A dose evaluation for the off-normal condition of a crane hang-up has been performed. Note that this crane hang up is the only off-normal condition during the STC movement from the spent fuel pool to the HI-TRAC. The following have been addressed in Chapters 7 and 10 of the licensing report. All operations will be performed following Indian Point's radiation protection program which will ensure that ALARA objectives are met.

- Dose to the Operator Evaluation: The dose rates to the primary and secondary operators in case of crane hang up are documented in Table 7.4.6. Two locations are considered in this respect, one directly above the STC lid (8 feet from the STC lid) for manual crane operation and the other one 22 feet from the surface of the STC for supervising the manual crane operation. Note that the 8 feet distance represents the minimum approach distance to the STC and the distance will increase as the crane lowers the STC into the SFP or HI-TRAC. Dose rates (mrem/hr) are reported in Table 7.4.6, so that Indian Point Radiation Protection personnel can estimate the person-rem depending on the particular situation. It is not expected that more than 3 personnel will be needed for maneuvering the crane manually. Two operators will be stationed at the sides of the trolley to manually release the brakes and one will observe the load from a distance. Four hours duration is considered and used for the annual dose calculations in case of crane hang up. Four hours is sufficient time to either return the STC to the SFP or place it into the HI-TRAC by manually operating the crane. Dose to the operator from the crane hang up is discussed in Section 7.4.4, while the operational part is described in Section 10.5.1. Table 7.4.6 demonstrates that the dose to the operators in case of crane hang up will be reasonable considering the short duration of the event.

- Other Plant Personnel: Note that only plant personnel required to perform the wet transfer operation will be allowed near or inside the FSB during STC movement from the SFP to the HI-TRAC.

- Dose to the Public (On-site): Dose to the public is discussed in Section 7.4.5 for normal STC movement between the SFP to the HI-TRAC and also in case of crane hang up. The dose rates to the member of public during normal operation are documented in Table 7.4.7, and Table 7.4.8 demonstrates the dose rates for crane hang up meet regulatory limits. A 60 m distance is required for the member of the public from the surface of the STC. Conservatively, the same 60 m distance around the FSBs at Unit 3 and 2 will be imposed as restricted area and will be controlled in accordance with the Technical Specifications. Note that the fuel storage building is not credited for this calculation.

- Impact in Adjacent Facilities: As a 60 m restricted area from the FSBs is required for normal operation and will be controlled in accordance with the Technical Specifications, the crane hang up event will have minimal impact on the adjacent facilities. In other

words, all the adjacent buildings within 60 m radius from the FSBs will be restricted to other plant personnel and member of public during the STC movement from SFP to HI-TRAC. Note that 60 m is conservative as this is calculated without taking credit of any of the existing surrounding structures.

NRC RAI 7-5

The following editorial errors were identified. Provide corrected SAR pages for review. (CSDAB)

a. Table 7.4.14 should list the regulatory accident dose limit units as mrem and not mrem/yr. Accident dose limits in 10 CFR 72.106(b) are given in terms of dose and not dose per year.
b. The first paragraph in Section 7.0 should be modified to clearly indicate that off-normal conditions are limited per 10 CFR 72.104 together with normal conditions. The current paragraph text appears to incorrectly indicate that off-normal conditions are limited with accident conditions by 10 CFR 72.106 limits.
c. Ensure appropriate terms are used for the respective dose evaluations. The current text indicates that site boundary dose rates are used to show compliance with the intent of 10 CFR Part 72 dose limits. Dose limits in 10 CFR Part 72 are for the controlled area boundary, as defined in that Part. While compliance with the intent of Part 72 limits is attempted to demonstrate compliance with 10 CFR Part 50 and Part 20 limits, the term 'site boundary' is a 10 CFR Part 20 term. The relationship between the controlled area boundary and the site boundary can be elaborated to justify how demonstration of compliance with dose limits for the one equates to demonstration of compliance with the dose limits for the other.

This information is needed to ensure compliance with 10 CFR 20.1301(a), 10 CFR 50.36a and the intent of 10 CFR 72.104 and 72.106.

Response to RAI 7-5

The editorial errors in the licensing report, as noted in the RAI, have been addressed as follows.

a. The accident dose limit units have been corrected and mrem is specified instead of mrem/year in the appropriate Chapter 7 tables (Tables 7.4.18 to 7.4.20).

b. Chapter 7 has been extensively revised. In Section 7.1.3 it is identified that the off-normal condition dose limits are in accordance with 10 CFR 72.104.

In revised Chapter 7 the 10 CFR Part 72 term "controlled area boundary" is used instead of "site boundary" when demonstrating compliance to Part 72. Controlled area boundary and accompanying regulations are discussed in Section 3.1.2.

NRC RAI 7-6

Provide an occupational dose assessment that captures all elements of the spent fuel transfer operations and include appropriate justification of assumptions regarding personnel numbers and locations relative to the STC, time durations, applicable dose rates, and adequacy of assessment detail and description accuracy. Additionally, an inspection found that during lifts of the STC, there is a 13/16" gap between the lid and the flange; the assessment should consider this gap and STC operations should be modified, as appropriate, to account for this

gap. (CSDAB)

Accurate and adequate assessments of occupational dose are important to ensure appropriate considerations are taken in the development of detailed procedures and the planning of operations. Staff continues to have concerns regarding the occupational dose assessment provided in the licensing report. Staff's initial concerns (see first round RAI questions 7-17 and 7-18) were only partly addressed, and further questions have arisen due to changes made to the assessment and the much higher dose rate estimates for the loaded STC. First, it is not clear that the assessment includes all elements of the operations. A comparison with the assessments done for the HI-STORM 100 system's loading operations illustrates the level of detail and comprehensiveness that would be expected, allowing for differences between systems (e.g., bolted closure vs. welded closure). Additionally, new procedures have been introduced, such as the 24-hour pressure rise test. Second, there continue to be inconsistencies between the assessment's description of operation conditions and those analyzed in the shielding models and the descriptions in Chapter 10 of the licensing report. Third, dose rates for some operations appear to be inappropriate for the actual personnel locations and configurations as understood from the Chapter 10 descriptions. Examples include operations on the STC lid using dose rates from the HI-TRAC axial side. Finally, various changes to the numbers and locations of personnel relative to the operations have been made without explanation or basis. It is not clear how fewer people than previously stated are required for the same operations. It is also not clear how they are to perform their functions from greater distances. For example, operations for raising the STC from the SFP, such as surveying the STC lid dose rates and washing the STC and crane equipment with clean water appear to be performed from 10 meters distance from the STC. Also, some operations of a similar nature are done under different conditions (e.g., placing STC in HI-TRAC vs. moving STC from HI-TRAC into SFP).

This information is needed to confirm compliance with 10 CFR 20.1101(b) and the intent of 10 CFR 72.104 and 72.126.

Response to RAI 7-6

Occupational doses to the radiation workers have been reevaluated and are reported in the revised licensing report. The results of this reevaluation are presented in Table 7.4.22. Section 7.4.12 discusses the occupational dose calculation. All operations will be performed following Indian Point's radiation protection program which will ensure that ALARA objectives are met, e.g. lead blankets will be used for radiation protection as applicable. Moreover, the operators will not all be at the closest location for the entire duration with respect to the STC or HI-TRAC during a specific operation. The operators will perform their stipulated tasks during a specific operation and, on completion of that task, wait in a low dose waiting area. Therefore, an additional column is introduced in the occupational dose table (Table 7.4.22), which documents the duration at closest distance with respect to the STC or HI-TRAC. These durations at the closest distance are utilized for the occupational dose calculations.

The occupational dose to the workers has been reevaluated considering the following:

- All the dose rates from the bare STC are evaluated assuming a 13/16" gap between the lid and the flange. A steel ring is attached with the lid to minimize streaming through this gap. Additionally, the few localized relatively high dose locations surrounding STC, such

as on surface of the ribs, are clearly shown in the licensing report and the Indian Point radiation protection personnel will develop their ALARA plans accordingly.

- All elements of the operations as presented in Table 7.4.22 are covered with as much detail as practical. The occupational dose calculation steps are consistent with the operation description in Chapter 10 of the licensing report.

- All the required procedures performed by the worker in the radiation zone are accounted for in this new occupational dose evaluation, including the 24 hour pressure rise test.

- Appropriate dose rates are used for the dose to the worker calculations. For example the dose rate from the outer ring of the STC top lid (from Table 7.4.6) is used for all the operations on the STC lid. Note that dose rates are calculated without the HI-TRAC top lid for HI-TRAC related operations. Workers will perform the HI-TRAC top lid operations from the side of the HI-TRAC by extending their hands and other equipment. Nevertheless, for HI-TRAC lid operations dose rates from the outer ring of the STC lid is utilized.

- The locations and number of personnel involved have been revised as detailed in the licensing report. These inputs to the dose analysis are based on a careful evaluation of the expected operational procedures and other spent fuel transfer experiences.

- Remote operational tools will be employed as far as practicable to minimize the time that a worker is near the STC.

It can be concluded from Table 7.4.22, that the occupational dose associated with fuel transfer operations employing the STC/HI-TRAC is reasonable.

NRC RAI 7-7

Explain whether normal operations with the loaded STC by itself are done remotely, including how remote operations are performed, the kinds of equipment used, and the quality standards employed for such equipment. If normal conditions are dependent upon remote operations, an appropriate condition should be included in the proposed TS, Appendix C. (CSDAB)

With revised STC dose rates much higher than has been evaluated for normal spent fuel loading or unloading, for currently approved spent fuel storage systems and the revisions to worker positions relative to the STC outside the SFP and the HI-TRAC, it is not clear whether or not operations are expected to be done in a remote fashion such that under normal conditions personnel are not around the STC like they are for loading operations for approved spent fuel storage systems. If the evaluation and operations rely upon remote operations, the application should clearly indicate that is the case, providing a description of how remote operations are to be performed (e.g., optical guidance systems and remote crane maneuvering), an explanation of how some steps can be performed remotely, and the assurance of equipment reliability for such operations. Chapter 10 of the licensing report should also be modified to clearly indicate that operations are performed remotely. Additionally, a TS condition should be added to state that the STC is handled remotely when out of the SFP and the HI-TRAC in conjunction with

appropriate ALARA practices and that equipment assigned appropriate quality standards for remote handling operations will be used for such operations, with consideration for redundancy of such equipment.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.34, and the intent of 10 CFR 72.104, and 72.126.

Response to RAI 7-7

The estimated average dose rates at the side of the STC (right below the upper flange) are less than 4 rem/hr similar to the estimated average dose rates at the side of the Holtec HI-TRAC transfer cask (from HI-STORM FSAR) when used with MPC-32 (3.6 rem/hr). It is noted that, local dose rates at the rib locations on the sides of the STC are higher than the dose rates at the rib locations of the Holtec HI-TRAC transfer cask, however, these dose rates do not exceed those for other normal reactor maintenance operations.

In keeping with ALARA practices, operator dose will be minimized through a combination of maximizing distance from the source and limiting the operator time in the dose field. The highest dose rates that occur during the fuel transfer evolution take place when the STC is in open air when traveling between the spent fuel pool and the HI-TRAC. During these operations, the crane operator will use a wireless radio controlled pendant to operate the crane so that the operator can be at a distance from the STC while still maintaining visual contact. The operator will be on the side of the pool/truck bay opposite of the STC which is a minimum distance of about 8 meters. The crane spotter will use remote cameras to help guide the operation when inserting the STC into the HI-TRAC and into the spent fuel pool. He will be located in a position similar to that of the crane operator.

The bottom of the STC has a generous taper and the top of the STC centering assembly has a lead-in with a 45 degree angle and 2" width to facilitate aligning the STC with the opening in the HI-TRAC. There is no need for additional operations personnel to be near the HI-TRAC when the STC is being installed. During unloading, the clearance of the sides of the STC to the spent fuel pool racks, walls, and tool holders is at least 8", therefore the STC can be positioned in the pool and lowered into the water to reduce dose before any final adjustments in position are required. The positioning of the crane can be further enhanced using alignment aids such as laser pointers mounted to the crane bridge and/or trolley along with targets on the fuel building floor to repeatedly locate the STC in the HI-TRAC and spent fuel pool without any significant intervention or adjustment of the crane position.

The wireless control for the crane is a commercial item that is in use at most nuclear plants in the US. They are used at many sites that load the Holtec HI-STORM system. The system has been proven reliable and will be tested during the required dry runs. The most common cause for problems with the remote controls is low battery power and interference from other radio equipment in the area. Spare batteries will be maintained during loading and radio interference issues are controlled through the plant procedures related to RFI/EMI controls. There is no special testing or qualification needed for the wireless controls for the crane.

NRC RAI 7-8

Perform an evaluation to demonstrate compliance with 10 CFR 20.1301(b) for STC movement between the SFP and the HI-TRAC for both loading and unloading operations, implementing

additional controls to ensure compliance with this requirement, including appropriate TS, as necessary. (CSDAB)

It is not clear if the current evaluation addresses compliance with the limits of 10CFR 20.1301(b) for the condition of the STC by itself, including for normal conditions. Given the high dose rates of the STC and the relatively close proximity of other buildings where members of the public may be located (as seen in Figure 7.4.2 of the licensing report) while loading or unloading operations are ongoing, it is not clear whether additional controls are needed to ensure compliance with 10 CFR 20.1301(b). Any assumptions regarding shielding provided by building materials and structures should be appropriately justified. Consideration should also be given to the need for additional conditions in the proposed TS, as necessary, to ensure appropriate controls are instituted to ensure compliance with 10 CFR 20.1301(b).

This information is needed to confirm compliance with 10 CFR 20.1301(b).

Response to RAI 7-8

An evaluation has been performed to demonstrate compliance with 10 CFR 20.1301(b) for the STC movement between the SFP and the HI-TRAC and the results are presented in Tables 7.4.7 and 7.4.8 of the licensing report. The evaluation shows that a 60 m restricted area is required from the surface of the STC. Conservatively, the same 60 m distance around the FSBs at Unit 3 and 2 will be imposed as restricted area and will be controlled in accordance with the Technical Specifications during the STC movement from SFP to HI-TRAC. Note that the fuel storage building itself nor any other surrounding structures are credited in this calculation.

The proposed TS has been revised to include the restricted area for STC movement between the SFP and the HI-TRAC.

NRC RAI 7-9

Provide further justification regarding the assumed cobalt impurity levels. (CSDAB)

While staff accepts that the dose evaluations for the public and the controlled area boundary use fuel with decay times appropriate for the assumed cobalt impurity levels (as stated in response to first round RAI 7-14), this does not hold true for the occupational dose assessments that use fuel that, based upon the decay time and the burnup, would have been manufactured in a time when cobalt was not controlled to limit its amount in assembly hardware as it is for more recently fabricated fuel (post 1989). Additionally, it is still not clear that Non-Fuel Hardware (NFH) impurity levels may not be higher as well since the combination of proposed burnup and cooling time limits for NFH indicate that these would also have been fabricated during the period before cobalt reduction efforts were begun. Thus, the applicant should provide justification for the impurity level assumed for the fuel contents used in the occupational dose evaluations and the NFH (in all evaluations) or modify the evaluations to account for higher cobalt levels, on par with what has been identified in literature as the cobalt levels found in assembly hardware from that time (pre-1989).

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 10 CFR 72.104 and 72.106(b).

Response to RAI 7-9

The revised analyses utilize maximum cobalt impurity levels, which are obtained from the measurements of the composition of the assembly hardware used by the Indian Point unit 3 assembly manufacturers (Westinghouse). The maximum cobalt-59 content of the pre-1989 fuel assemblies in the unit 3 SFP is 1.2 gm/kg of steel, whereas the post-1989 assemblies contain 0.5 gm/kg (maximum) of cobalt-59. These cobalt-59 impurities are used in the shielding calculations. More specifically, 1.2 gm/kg is applied for assemblies with 20 years cooling (outer basket locations) and 0.5 gm/kg are used for assemblies with 10 years cooling (inner basket locations). Additionally, the unit 3 pool has 6 assemblies with inconel spacer grids. These are accounted for in the calculations by assuming all 4 inner region assemblies contain inconel spacer grids. 4.7 gm/kg of cobalt-59 is assumed for inconel.

Since relatively long cooling times (greater than 15 years) may be required for BPRAs and TPDs (especially for the TPDs with high burnups), the value of 1.2 gm/kg of cobalt-59 is used for steel in all BPRAs and TPDs. 4.7 gm/kg of cobalt-59 is applied for inconel in those devices. For RCCAs only inconel is considered with 4.7 gm/kg of cobalt impurity. NSAs are analyzed with 0.5 gm/kg of cobalt-59 in steel as 10 years of cooling time is assumed in the dose evaluations for those devices. However, for NSA inconel is again considered with 4.7 gm/kg of cobalt impurity.

All cobalt impurity levels used in the calculations, including assembly fittings and NFH, are discussed in Section 7.2.2 of the licensing report.

NRC RAI 7-10

Provide the basis for the division of off-normal conditions into two categories of time duration (8 hours and 30 days) and describe the kinds of events/conditions that fall into the two different categories, including adequate justification. (CSDAB)

It is not clear that the division of off-normal conditions into two categories of differing durations is appropriate or justified. It is also not clear what kinds of events or conditions would be considered to fall into one or the other of these categories nor why their classification as one or the other type of off-normal condition would be justified. Staff notes that assumptions regarding off-normal conditions may impact whether or not compliance with the intent of 10 CFR 72.104(a) can be demonstrated, even in cases where only a single transfer is considered as off-normal and the remaining transfers are normal; thus, evaluations to demonstrate this compliance should be modified, as necessary.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.104(a).

Response to RAI 7-10

In the previous submittal an 8 hours off-normal condition was considered for HI-TRAC movement between unit 3 and 2 with 10% fuel rod breaches. In the current submittal, the sealing system of the STC will be tested to leak tight criteria per ANSI 14.5. Therefore, an effluent contribution to the dose rates is no longer considered in the current revision of the licensing report, which eliminates the requirement of the off-normal condition with 10% rod breaches.

In the current revision two off-normal conditions are postulated, one for bare STC movement from the SFP to the HI-TRAC and the other one for the HI-TRAC movement between unit 3 and

2. Crane hang up (4 hrs) is considered as the off-normal condition for STC movement between the SFP and the HI-RAC. The postulated off-normal condition for the HI-TRAC is the transporter break down (30 days event).

NRC RAI 7-11

Justify the change in the hours used to determine the annual dose contribution to the controlled area boundary of the Independent Spent Fuel Storage Installation (ISFSI) and other site facilities or use the originally assumed hours. (CSDAB)

In its resubmittal of the licensing report along with the RAI responses, the applicant changed the number of hours per year used in the determination of the annual dose contribution from the ISFSI and other site facilities for evaluation of annual doses against the 72.104(a) limits. The hours were reduced to 192 from 500. While 192 hours is the total hours expected for the total number of spent fuel transfers anticipated in a given year (currently taken to be 24), the ISFSI and site facilities contribute to dose at the remaining times of the year when spent fuel transfers are not occurring. Based upon previous arguments, 500 hours seems to be a more justifiable time estimate to use for these facilities for 72.104(a) evaluations. As part of any justification, the applicant should also include dose evaluations for a controlled area boundary at a distance of 137 meters (based upon the distance assumed in the ISFSI 72.212 evaluation), calculating the ISFSI and site facilities' dose contributions assuming 500 hours per year and 24 spent fuel transfers. These evaluations would need to address both normal and off-normal conditions. Staff notes that the assumptions regarding exposure time may impact the ability to demonstrate compliance with the intent of 10 CFR 72.104(a), even in cases where only a single transfer is considered off-normal and the remaining transfers are normal; evaluations to demonstrate this compliance should be modified, as necessary.

This information is needed to confirm compliance with 10 CFR 50.34, 10 CFR 50.34a, and the intent of 10 CFR 72.104(a).

Response to RAI 7-11

In the previous submittal the time used to determine the annual dose contribution to the controlled area boundary of the ISFSI and other site facilities was 192 hours per year. In the current submittal Chapter 7 of the licensing report has been extensively revised and 500 hours per year is used together with a controlled area boundary at a distance of 137 meters from the edge of the ISFSI. This change is reflected in Tables 7.4.16, 7.4.17, 7.4.18, 7.4.19, and 7.4.20.

Note that in the current submittal, 16 transfer operations between unit 3 and 2 are considered as that is the maximum targeted number of transfers per year.

NRC RAI 7-12

Provide an evaluation of the impacts on dose rates around the loaded STC and the loaded HI-TRAC for contents including a neutron source assembly (NSA). (CSDAB)

The current evaluation relies upon the arguments used in the HI-STORM 100 FSAR for NSAs. However, the allowable loading configuration in a HI-STORM 100 is such that a basket cell with an NSA is always completely surrounded by basket cells loaded with fuel not containing NSAs on all sides. This is not the case in the STC basket. Furthermore, some NSAs have significantly long half-lives and source strengths similar to design-basis fuel assemblies. This

was the basis for the restriction of only a single NSA loaded in the very center of the MPC. In the STC, no basket cell where a NSA may be loaded is completely surrounded by basket cells without NSAs. Thus a more detailed evaluation of the impacts on dose rates around the loaded STC and the loaded HI-TRAC should be performed to show these impacts, including azimuthal variations to capture the areas around the STC and around the HI-TRAC where the outer basket cell contents do not shield the inner basket cells where NSAs are permitted. The evaluations should also address any potential impacts on the doses to personnel and members of the public.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 20.1301(a) and (b), 10 CFR 50.34, and the intent of 10 CFR 72.104, 72.106 and 72.126.

Response to RAI 7-12

An evaluation has been performed of the impacts on dose rates around the loaded STC and HI-TRAC for contents including a NSA. In the evaluation NSAs are restricted to one per cask and also restricted to the inner region. These restrictions are incorporated into proposed TS Table 3.1.2-2. In this evaluation the NSA source terms are described in Section 7.2.2 of the licensing report and the NSAs are evaluated at 360,000 MWD/MTU burnup and 10 years cooling time. This change has also been incorporated into proposed TS Table 3.1.2-2. The dose results on the surface, and 1 m from the STC and HI-TRAC are reported in Table 7.4.21. For side dose rates from both the STC and HI-TRAC, dose rates are reported at 0^0 (on the rib) and at 45^0. The 45^0 location is selected to show the dose rates on the surface and at 1 m, where the outer basket cell contents does not shield the inner basket cells. Table 7.4.21 demonstrates that in general one NSA and 11 BPRAs are bounded by 12 BPRAs for the side, and for the bottom the NSA is bounded by the RCCAs. Note that in some cases the dose rates from 1 NSA and 11 BPRAs combination are slightly higher than that of 12 BPRAs case. Further investigation has revealed that the difference in the radial surface dose rates (Surface 0^0 and 1 m away from surface 0^0 for the bare STC) are not statistically significant, i.e. results are within one standard deviation. Additionally, the top dose rates (bare STC) between the two cases (1 NSA and 11 BPRAs vs. 12 BPRAs) are also comparable. HI-TRAC dose rates with NSA at 45^0 (on surface) and for the top lid are marginally higher than that with the BPRAs. This marginal difference is due to the NSA neutron source term. However, it is important to note here that the HI-TRAC dose rates are calculated without considering borated water inside the STC. Therefore, BPRAs and RCCAs are used to report the maximum doses and to show compliance with the regulatory limits.

NRC RAI 7-13

Modify the accident dose evaluation to include the following configurations as a result of a tip over accident: (CSDAB)

a. dose resulting from exposure of the HI-TRAC base, side, and top, accounting for any areas of the STC basket that are no longer submerged in water as a result of the cavity water receding from the side of the STC now facing up, and
b. the STC off-center in the HI-TRAC as a result of the tip over accident and the crushing of the inner impact limiter/STC centering assembly, with the loaded HI-TRAC in the vertical orientation.

An evaluation of the tip over accident should address applicable shielding and dose rate conditions and evaluations. According to the structural evaluation, the STC centering assembly

acts as an impact limiter and crushes during the tip over accident; however, it is not clear how much crush will occur. The shielding evaluation should consider the case where the STC is no longer centered in the HI-TRAC, shifting the STC from center to the extent that the structural evaluation shows the centering assembly will crush. The assumption of loss of the water jacket should also be used in these evaluations as it is for the current accident evaluation.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.106.

Response to RAI 7-13

The accident dose evaluation has been modified to include the RAI identified configurations as a result of a tip over accident. These additional two conditions are:

1. Simultaneous loss of water from the HI-TRAC water jacket and from the annular region between the HI-TRAC and STC. Note that the loss of water from the annular region between the STC and the HI-TRAC was modeled instead of water receding from the side of the STC now facing up after a tip over, since the complete loss of water from the annular region would bound the condition of relocation of water in the STC from the tip over.

2. STC off-center as a result of tip over accident accompanying with simultaneous loss of water from water jacket and HI-TRAC annulus.

Note that as in all other HI-TRAC calculations, these two conditions are also modeled without the HI-TRAC top lid. The results are reported in Tables 7.4.12 and 7.4.13 of the licensing report. Additionally, these two accident conditions are evaluated to show compliance with 10 CFR 72.106 at the controlled area boundary. Results of those evaluations are presented in Tables 7.4.19 and 7.4.20.

NRC RAI 7-14

Provide an evaluation that demonstrates the bounding dose rates for a loaded STC containing the contents permitted by the proposed TS contents limits. (CSDAB)

A regionalized loading pattern is used for some dose evaluations in the application. Staff asked the applicant in the previous RAI (see question 7-15 of the first RAI) to justify the use of the selected source terms in the regionalized loading pattern and the bounding nature of the dose rates and dose estimates from these sources. The applicant discussed use of a uniform loading pattern to perform some of the evaluations, stating that it exceeds the limits of what is allowed by the proposed decay heat limits. The applicant's response does not address the issue, especially in light of the very high dose rates from the STC with the representative loading. Decay heat limits and dose rates do not correlate on a one-to-one basis given that different combinations of burnup, decay time and minimum enrichment can yield the same decay heat but quite different radiation source terms. For the proposed operations, the STC operates much like a transfer cask and can be considered a lightweight transfer cask, since it is the device that is used to load and unload fuel from/into the spent fuel pools. With the regionalized loading also resulting in very high dose rates on the STC, much higher than has been evaluated for all approved dry storage systems' transfer casks, it is important to understand the maximum dose rates that may be obtained for the allowable contents; even small relative dose rate variations

can mean large changes in dose rates. Understanding the maximum dose rates that can occur during transfer operations will enable proper ALARA and operations planning as well as demonstration that transfer operations with all proposed contents will meet the regulations at all stages of the operations. Dose evaluations should be modified as necessary to account for the bounding dose rates.

This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a) and (b), 10 CFR 50.90 and 50.34a(c) and the intent of 10 CFR 72.104.

Response to RAI 7-14

A revised dose evaluation has been performed that has determined the bounding dose rates for a loaded STC containing the contents permitted by newly proposed TS contents limits. The revised analysis recognizes that decay heat limits and dose rates do not correlate on a one-to-one basis given that different combinations of burnup, decay time and minimum enrichment can yield the same decay heat but quite different radiation source terms.

In the revised analysis five loading combinations which represent the majority of the current assemblies in the Indian Point unit 3 spent fuel pool were selected. The remainder of the assemblies in the spent fuel pool, and those to be added to the pool inventory during refueling, will also be covered by these selections after additional cooling in the SFP. These selections are reported in Table 7.1.1 of the licensing report and are also included in the TS. The dose rates are evaluated for all five loading patterns and are presented in Tables 7.4.1 and 7.4.9 for STC and HI-TRAC, respectively. Table 7.4.1 show that loading pattern 4 is bounding from shielding perspective. Hence, loading pattern 4 is utilized for all other bare STC dose calculations in Chapter 7. On the other hand, Table 7.4.9 establishes that in general the loading pattern 3 is bounding for the side, while loading patterns 4 is bounding for top and bottom for the HI-TRAC. However, it is observed that this trend is not universal and depends on the dose locations and transfer situations (normal or accident) [L.G]. In Chapter 7, dose rates for all the HI-TARC cases are reported only for the bounding loading patterns as determined.

NRC RAI 7-15

Refer to RAI 7-1 from the previous RAI letter. Evaluate the effect on dose and potential effect on canister leakage rate assuming a 10% fuel rod breach for all off-normal conditions. (TCB)

Section 7.4.5 "Effluent Dose Evaluation" of Report HI-2094289 identifies the first off-normal condition as a breakdown of the cask transporter without HI-TRAC recoverability for 30 days, but only assumes 1% fuel rod breach. The percent of spent fuel postulated to fail for off-normal conditions is 10% as identified in Table 5-2 of NUREG-1536 Rev. 1 "Standard Review Plan for Spent fuel Dry Storage Systems at a General License Facility". There appears to be no justification for only assuming 1% fuel rod breach. The 10% release fraction identified in the standard review plan is a bounding value for off-normal conditions and is not meant to be reduced based on postulated specific scenarios. Similarly, we would not expect the 100% fuel rod breach for accident conditions to be reduced based on a postulated specific accident, but utilized as a bounding value.

10 CFR Part 50, Appendix B, Criterion III, Design Control states in part, that measures shall be established to assure that applicable regulatory requirements and the design basis....are correctly translated into specifications, drawings, procedures, and instructions. Also, ASME Code, Section III, NCA-4000, Article 4134.3(a) references ASME NQA-1, where in Supplement

3S-1, Section 3.1 it states that design analyses such as physics, stress, hydraulic and accident, shall be performed in a planned, controlled and correct manner.

Response to RAI 7-15

The effect on dose and the potential effect on canister leakage rate assuming a 10% fuel rod breach for off-normal conditions no longer requires evaluation. A new approach has been taken and the sealing system of the STC will now be tested to the leak tight criteria per ANSI 14.5 as described in the response to RAI 8-1. The factory leak testing of the entire STC confinement boundary is also discussed in that response. Therefore, the effluent contribution to the dose rates is not considered in the latest revision of the licensing report.

NRC RAI 7-16

Refer to RAI 7-2 from the previous RAI letter. Justify or remove the statement at the end of the third paragraph in Section 7.4.5 “Effluent Dose Evaluation” of Report HI-2094289 that fines, volatiles and crud would remain entrapped within the water environment. If applicable, submit a revised section 7.4.5 as part of your RAI response. (TCB)

Since the STC, the confinement boundary, is tested to a finite leak rate (i.e. not leaktight) and can be pressurized (refer to Figure 5.3.2 “24 Hour Pressure Rise Under Design Basis Conditions,” a pathway exists for the potential release of radioactive material. No credit can be taken for the HI-TRAC since it is not part of the confinement boundary. Hence, making a statement that this contamination would remain within the water seems invalid.

10 CFR Part 50, Appendix B, Criterion III, Design Control states in part, that measures shall be established to assure that applicable regulatory requirements and the design basis....are correctly translated into specifications, drawings, procedures, and instructions. Also, ASME Code, Section III, NCA-4000, Article 4134.3(a) references ASME NQA-1, where in Supplement 3S-1, Section 3.1 it states that design analyses such as physics, stress, hydraulic and accident, shall be performed in a planned, controlled and correct manner.

Response to RAI 7-16

As discussed in the response to RAI 7-15 a new approach has been taken and the sealing system of the STC will now be tested to the leak tight criteria per ANSI 14.5 as described in the response to RAI 8-1. The factory leak testing of the entire STC confinement boundary is also discussed in that response. Therefore, the effluent contribution to the dose rates is not considered in the latest revision of the licensing report, which additionally eliminates the requirement for HI-TRAC credit as a part of confinement boundary. The above mentioned statement from Section 7.4.5 has been deleted from the revised licensing report.

NRC RAI 7-17

Refer to RAI 7-3 from the previous RAI letter. The response to NRC RAI 7-3 provided by letter dated October 5, 2010, included an unnumbered table on page 57 of Attachment 1 which summarizes inputs used to calculate the atmospheric dispersion factors (χ/Q values) used in the effluent dose evaluation. (AADB)

a. Please explain how the σ_y and σ_z values listed in the table on page 57 were derived from Figures 1 and 2 of Regulatory Guide (RG) 1.145, “Atmospheric Dispersion Models for Potential

Accident Consequence Assessments at Nuclear Power Plants." For example, were σ_y and σ_z computer generated or were they derived by approximation directly from the figures? Please provide additional detail, including computer generated summaries and/or annotated figures, which show how the calculations were made, particularly at 1 and 20 meters as these distances are less than the 100 meter minimum distance plotted in Figures 1 and 2.

b. Please explain the basis for why χ/Q values were calculated for a distance of 1 meter. Please provide additional detail supporting the use of the RG 1.145 methodology to determine χ/Q values considering that the intent of the guidance does not appear to be appropriate for this distance. RG 1.145, which provides guidance for calculation of χ/Q values applicable to typical exclusion area boundary and low population zone distances, implicitly assumes a minimum distance of 100 meters with regard to information provided in Figures 1 and 2. Further, NRC staff notes that other NRC documents such as RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," state that NRC guidance for the calculation of χ/Q values may not apply under certain conditions, such as at distances less than about 10 meters.

c. The application references Interim Staff Guidance (ISG)-5, "Confinement Evaluation," as the basis for the stability and wind speed inputs. ISG-5 states that the use of stability category D and a wind speed of 5 meters per second (m/s) are acceptable for the normal and off-normal case calculations, and stability category F and a wind speed of 1 m/s are acceptable for the accident case calculation. These are default values for dry cask storage systems. NRC staff notes that these values are based on a generic approximation of meteorological conditions for an average site in the United States. Please clarify whether the normal and off-normal cases are assumed to occur during 50 percentile and the accident case during 95 percentile meteorological conditions, respectively, as is typically assumed for other nuclear reactor effluent dose evaluations. Please confirm how the use of these default values is justified when compared with representative 50 and 95 percentile meteorological conditions at the Indian Point site.

d. Figure 7.4.2 of Holtec International Report HI-2094289 (ADAMS Accession Number ML103080113) is a site map showing the haul path and an exclusion area boundary (EAB), which the licensee has defined for use in the current license amendment request. This EAB does not appear to be the current Indian Point licensing basis EAB for either Unit 2 or Unit 3 defined by 10 CFR Part 50.2. Therefore, please explain the relationship of the EAB shown in Figure 7.4.2 and the current Indian Point licensing basis EABs. In addition, please provide the minimum distance between any point along the haul path and 1) the Indian Point current licensing basis EABs, and 2) the control room intakes.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61.

Response to RAI 7-17

As discussed in the response to RAI 7-15 a new approach has been taken and the sealing system of the STC will now be tested to the leak tight criteria per ANSI 14.5 as described in the response to RAI 8-1. The factory leak testing of the entire STC confinement boundary is also discussed in that response. Therefore, σ_y, σ_z, χ/Q, and wind conditions no longer require explanation or clarification as requested by RAI Parts, a, b, and c.

Part d: The "exclusion area boundary" that was defined in Figure 7.4.2 of the licensing report is

not related to Indian Point's exclusion area boundary. The exclusion area boundary defined in Figure 7.4.2 is temporary and only applies to STC/HI-TRAC movement between unit 3 and 2. To avoid any further confusion the term "exclusion area boundary" is removed from the report and the term "restricted area" is introduced. Note that this is a temporary restricted area which will remain in effect only during the STC/HI-TRAC transfer. Two restricted area are defined, the first one is for the STC movement between the pool and the HI-TRAC and vice versa and the second one is the HI-TRAC movement between unit 3 and 2. Note that they are independent of each other as the two operations are not occurring at the same time.

Exclusion zones for Indian Point are defined in Chapter 2 of the IP3 FSAR. The exclusion zone boundary for Indian Point Unit 3 is at a distance of 350 m from the containment building, whereas the exclusion zone boundary for Unit 2 is defined by 520 m distance from the containment building. As the effluent calculations are no longer relevant to the shielding chapter of the licensing report, the locations of the control room intakes are no longer required.

CHAPTER 8 – MATERIALS EVALUATION, ACCEPTANCE TESTS and MAINTENANCE PROGRAM

NRC RAI 8-1

Refer to RAI 8-1 from the previous RAI letter. Justify the lack of leak testing of the entire confinement boundary of the STC as well as describe what leak tests are done in the shop and the field. (TCB)

The *original* RAI 8-1 *included the following which was not addressed in the response or in the* referenced sections of Holtec Report HI-2094289: "Additionally, the entire confinement boundary should be leak tested in accordance with the guidance of ANSI 14.5-1997 to verify compliance with the design leak tightness as determined in RAI 7-1, above. This leak testing should be performed initially at the fabrication facility and periodically (within 12 months prior to each use) to ensure that the containment leak tightness has not deteriorated over time. The leak testing done at the time of loading fuel is usually less stringent and is done to ensure that the gaskets are properly seated and the containment has been assembled properly, and typically is checked to be at least 1E-3 ref-cm3/sec."

The applicant's response stated that "Section 8.4.4 had been revised to more clearly define the factory leakage test and the periodic leakage test of the lid gaskets that is performed during loading operations." However upon reviewing Section 8.4.4 no distinction is made between factory and periodic leak tests and no discussion is provided with regard to leak testing the entire confinement boundary.

This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and 10 CFR Part 50, Appendix B, Criterion XI- Test Control.

Response to RAI 8-1

Section 8.4.4 has been revised to discuss the factory leak testing of the STC confinement boundary. The following text has been added:

> The *entire* STC *confinement boundary shall be leak tested at the factory prior to initial* use to demonstrate that the leakage rate meets the criteria for "leak-tight" as defined in ANSI N14.5. The confinement boundary material and welds will be tested using the gas

filled envelope (gas detector) method or other applicable method per ANSI N14.5. The seals on the STC lid and lid cover plates shall be leak tested using the evacuated envelop (gas detector) method, or other suitable method as described in ANSI N14.5. The leakage rate acceptance criterion is "leak-tight" as defined in ANSI N14.5.

Section 8.5.2 and Table 8.5.1 have been revised to specify that the seals on the STC confinement boundary will undergo a periodic leakage test (within 12 months of fuel transfer) to demonstrate that the confinement boundary seals will maintain the required leakage rate of "leak-tight" and that the confinement capabilities have not deteriorated over an extended period of use. They will also be revised to specify that the seals on the STC confinement boundary will be subject to a pre-transfer leakage test (analogous to the "pre-shipment" test in ANSI N14.5) to confirm that the confinement system has been properly assembled for fuel transfer. The STC seals will be leak tested to demonstrate that there is no leakage when tested to a sensitivity of 1 x 10^{-3} ref. cc/sec in accordance with the ANSI N14.5 requirements. In addition, text has been added to Section 8.5.2 to define the leakage testing requirements following any maintenance activity that may affect the function of the confinement boundary. Retesting of the body of the STC is only required if maintenance activities have been performed that may affect the confinement boundary (e.g., weld repairs to confinement boundary welds). The following revised text has been added to section 8.5.2:

The seals on the STC lid and lid cover plates shall be tested at a frequency defined in Table 8.5.1. The seals shall undergo a periodic leakage test to confirm that the confinement capabilities have not deteriorated over an extended period of use. The periodic leakage test of the seals shall be performed using the evacuated envelop (gas detector) method, or other suitable method as described in ANSI N14.5. The leakage rate acceptance criterion is "leak-tight" as defined in ANSI N14.5. The seals shall undergo a pre-transfer leakage test to confirm that the confinement system has been properly assembled for fuel transfer. The pre-transfer leakage test of the seals shall be performed using the gas pressure rise method, or other suitable method as described in ANSI N14.5. The acceptance criteria is no detected leakage when tested to a sensitivity of 1 x 10^{-3} ref. cc/sec. Failure to achieve a leakage rate below the required value shall be cause for seal replacement, seating surface repair, or other repair and retest of the seal joint as applicable.

The STC confinement boundary components shall be leakage tested following any maintenance repair which may affect the confinement boundary function to demonstrate that the leakage rate is "leak-tight" as defined in ANSI N14.5. The confinement boundary material and welds will be tested using the gas filled envelope (gas detector) method or other applicable method per ANSI N14.5. The seals and sealing surfaces on the STC lid and lid cover plates shall be leak tested using the evacuated envelop-gas detector method, or other suitable method as described in ANSI N14.5. The leakage rate acceptance criterion is "leak-tight" as defined in ANSI N14.5.

Table 8.5.1 entries have been revised as follows:

Task	Frequency
Periodic Leakage Test of STC (lid and lid cover plates) seals	Within 12 months prior to each fuel transfer. Acceptance criteria is "leak-tight" as defined in ANSI N14.5.
Pre-transfer Leakage Test of STC (lid and	Following each fuel loading, prior to fuel

lid cover plates) seals	transfer. Acceptance criteria for each seal is no detected leakage when tested to a sensitivity of 1.0 x 10^{-3} ref. cc/sec.
Maintenance Leakage Test of STC (lid and lid cover plates) seals	Following seal replacement. Acceptance criterion is "leak-tight" as defined in ANSI N14.5.
Leakage Test of HI-TRAC top lid seals.	Following each fuel loading, prior to fuel transfer. Acceptance criteria is no detected leakage when tested to a sensitivity of 1.0 x 10^{-3} ref. cc/sec.

NRC RAI 8-2

Refer to RAI 8-3 from the previous RAI letter. Justify the response provided in amended Section 8.4.3. If applicable submit a revised Section 8.4.3 as part of your RAI response. The staff finds that the Code required pressure test (125% of design pressure) alone is not sufficient to pressure test the bottom flange area of the HI-TRAC since it neglects the weight of the fully loaded STC which sets on the bottom HI-TRAC flange. The applicant's submittal states that only the 125% of design pressure is required for the HI-TRAC pressure test, which is not correct. Also, indicate in this section the total test pressure needed to meet the Code required test including the weight of a fully loaded STC which weighs about 40 tons. As previously stated, a dead load equivalent to a fully loaded STC may be placed inside the HI-TRAC for the pressure test, with the weight of the HI-TRAC supported only by its trunnions, in lieu of increasing the pressure above 125% of the design pressure. (TCB)

The applicant's response to the previous RAI stated that changes were made to address this issue to Sections 10.1.2 "STC Preparation and Setup- HI-TRAC Inspections and Checklist" and 8.5.2 "Leakage Tests". However, the staff could find no such changes made relating to this issue. From reading the revised SAR sections, the staff is concerned that the applicant may not fully appreciate the purpose of the pressure test. The pressure test is a structural integrity test performed before use to assure proper fabrication of the HI-TRAC. It is not a leakage test, per se, but uses the absence of leaks as its acceptance criteria (as mandated by the Code) to ensure the as-constructed vessel's suitability for the loadings. Therefore more than just the gasketed area of the pool lid needs to be checked for leaks.

The NRC staff also needs confirmation of the structural integrity testing planned for the STC. If it will only be by pressure test, demonstrate that the pressure test exceeds all deadweight loads with appropriate margin and that the STC will be supported only by its crane attachment points during the test.

10 CFR Part 50, Appendix B, Criterion XI- Test Control requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Response to RAI 8-2

Text clarifying the requirements to account for the weight of the loaded STC on the HI-TRAC Pool Lid joint was added in the previous licensing report (Revision 3) in subsection 10.1.2, step

9 referring the user back to section 8.5.2 which describes the required maintenance leakage testing for use of the HI-TRAC. However, the text describing the requirements for testing was added to the maintenance leakage testing in subsection 8.5.2 and not to the initial acceptance testing described in subsection 8.4.3. The following text has been added to 8.4.3 to make the description complete in response to the RAI.

> When performing the test, the HI-TRAC shall be supported by the trunnions. The HI-TRAC shall be loaded with a full weight STC or STC mock-up and filled with water to fully load the pool lid bolting and bottom flange welds. As an alternative to using the full weight STC, the test pressure may be increased to account for the missing weight. The acceptance criteria for the pressure test shall include no visible water leakage from the pool seal and drain plug as well as the rest of the HI-TRAC water boundary. All joints, connections, and regions of high stress such as regions around openings and thickness transition sections shall be examined for leakage.

The STC pressure boundary test does not require that any dead loads are included in the testing. The STC is lifted by the crane attached to the lid with the fuel and STC weight being transferred from the STC trunnions to the lid through the STC Lifting Device. The lid bolts are not engaged during lifting such that the STC will only be lifted with the internal cavity vented to atmosphere. Therefore, a combination of internal pressure and dead weight of the fuel will not be present in combination. The internal cavity of the STC will only be pressurized with the STC resting on the bottom of the HI-TRAC and dead weights due to lifting are not a concern. The structural effects on the pressure boundary that result from lifting the STC by the trunnions will be determined by the 300% load test of the trunnions. However, as a defense in depth, the STC hydrostatic pressure test will be conducted with the STC supported by either the top lid or the lifting trunnions to maximize the loads on the confinement boundary welds. Text has been added to subsection 8.4.3 to identify this commitment.

NRC RAI 8-3

Make the following modifications to provide additional clarity and consistency. (CSDAB)

a. Modify Section 8.4.1 to also state that the STC will be assembled in accordance with and verified to meet the TS requirements for the STC design. In addition to the licensing drawings, the TS contain design requirements with which the fabricated STC must comply.
b. Modify Section 8.4.5 to also state that the lead sheet will be layered so that the minimum total thickness meets the TS requirements for the STC design. In addition to the licensing drawings, the TS contain a design requirement on the minimum thickness of the STC lead shielding.
c. Modify Section 8.4.5 to include more of the response to RAI question 8-11 from the previous RAI letter, particularly that the layering of lead sheets where each layer is made of multiple sections will be done so that section edges in adjacent layers are offset to eliminate potential streaming paths. This aspect of fabrication is important with regard to radiation protection, and based on the response to RAI question 8-11, should have been included in the referenced section of the licensing report.
d. Describe the acceptance testing that will be used to ensure that areas packed with lead wool will perform in a manner comparable with the lead sheet for shielding purposes.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.44(c)(4) and 72.126(a).

Response to RAI 8-3

a. Section 8.4.1 has been modified to address the comment.

The following text has been added to the first sentence of Section 8.4.1:

"The STC shall be assembled in accordance with the licensing drawings supplied in Section 1.5 and the applicable STC Technical Specifications."

b. Section 8.4.5 has been modified to address the comment.

The following text has been added to Section 8.4.5:

"The lead sheet will be layered for a minimum total thickness as specified on the licensing drawings found in Section 1.5 and the applicable STC Technical Specifications."

c. Section 8.4.5 has been modified to address the comment.

The following text has been added to Section 8.4.5:

"If multiple sections are used to make a layer, they are butted up tight against one another and any gaps are filled with lead wool as described above. If multiple sections are used to make the layers, the joints between the sections are staggered to eliminate any potential streaming paths."

d. Section 8.4.5 has been modified to address the comment.

The lead sheets were installed in the STC such that there will not be any gaps between lead sheets or the lead and the steel cavity which could lead to significant streaming. Any gaps that were observed during the lead installation were filled with lead wool which is compressed in place to fill the void. Nonetheless, as an additional defense in depth, the STC will be subjected to a gamma scan to demonstrate that the lead cavity is free of voids that would lead to significant streaming

The following text has been added to Section 8.4.5 to address the testing of the lead:

"The effectiveness of the lead installation in the STC body shall be verified after fabrication by performing a gamma scan on the accessible surfaces of the canister in the lead shielding region. The purpose of the gamma scan test is to demonstrate that the lead shielding is free from voids that may result in streaming paths through the lead. Measurements shall be taken on a 6-inch by 6-inch (nominal) grid pattern over the surfaces to be scanned. Any gamma dose rates that vary significantly from the average gamma dose measurements, accounting for the presence of the STC ribs as applicable, shall be evaluated by Holtec to determine the effect on the dose calculations provided for the STC. Should the measured calculations using the measured gamma dose rates show that the calculated dose rates will exceed the dose rates used to license the STC, corrective actions should be taken, if practicable, and the testing re-performed until successful results are achieved. If physical corrective actions are not practicable, the degraded condition may be dispositioned with a written evaluation in accordance with applicable procedures to determine the acceptability of the STC for service. Gamma scanning shall be performed in accordance with written and approved procedures. Measurements shall be documented and shall become part of the quality documentation

package."

NRC RAI 8-4

Refer to RAI 8-8 from the previous RAI letter. Identify and reconcile the discrepancies in applicable rules for construction between ASME Code, Section III, Division 1, Subsection ND and Subsection NC for the construction of the Shielded Transfer Canister (STC) to ensure that the STC is constructed to acceptable quality standards. (SMMB)

Spent fuel canisters (i.e. the confinement boundary) are normally constructed to ASME Code Subsection NB or NC (reference NUREG-1536 section 3.4.1). The applicant is proposing to construct the STC to Subsection ND, as indicated in Section 1.3.1 of HI-2094289, Rev. 3. This approach does not appear to provide the same degree of quality for a spent fuel storage or transfer canister.

This information is required for compliance with 10 CFR 50, Appendix A, GDC 61, and 10 CFR Part 72.122(a).

Response to RAI 8-4

The STC has been designed and manufactured to quality standards that are commensurate with those of an MPC even though the STC is used only for "short term operations" (in the parlance of 10CFR Part 72) and is not subject to the thermal and pressure transients that typically act on an MPC during its on-site storage and off-site transport. It is true that the reference code for the STC is ASME Section III, Division 1, Subsection ND, but, as summarized below, the design and manufacturing of the equipment far exceeded the requirements of "ND", as well as "NC" of the Code, in many significant respects. The reasons for selecting Subsection ND as the reference code for the STC, as opposed to Subsections NB or NC, are discussed in the response to RAI 8-8 from the October 5, 2010 Entergy letter (ADAMS Accession No. ML102910511). The key reasons are:

- the STC serves a similar role to that of the HI-TRAC in that they are both fuel transfer devices (not long term storage devices); since the code of reference for the HI-TRAC is ASME Section III, Subsection NF for Class 3 structures, the pressure vessel counterpart to "NF Class 3" – Subsection ND – is used as the reference code for the STC.
- cyclic fatigue, a core concern of Subsection NB and Subsection NC (NC-3200), is not a credible source of failure for the STC because significant thermal transients are essentially absent from the STC and the number of mechanical loading cycles is relatively small.

In what follows, the discrepancies in applicable rules for construction between ASME Subsection ND and Subsection NC for the construction of the STC are identified and evaluated.

First it is noted that Articles NC-3000 and ND-3000 are essentially the same, except for the following:

a) Subsection NC provides an alternate set of "design-by-analysis" rules (NC-3200), which uses allowable stress intensities (versus the "design-by-formula" method of NC-3300/ND-3300 which use allowable stresses);

b) NC-3300 expects the pressure vessel Category A, B and C welds to be 100 percent radiographed.

The allowable stress intensities of NC-3200 are greater than the allowable stresses of NC-3300/ND-3300 and, therefore, enable the designer to reduce the thickness of the vessel. The STC, however, is conservatively designed to meet the lower allowable stress limits of NC-3300/ND-3300 with significant additional margins of safety. A photo of the fabricated STC is shown in Figure 8-4.1 at the end of this RAI response.

Listed below are the principal design and manufacturing features of the STC that substantiate the degree of quality for the STC confinement boundary.

1) The primary membrane stress in the STC shell due to <u>accident</u> internal pressure plus normal handling is only 2,054 psi as compared to the <u>Level A</u> allowable membrane stress limit of 20,000 psi per NC-3300/ND-3300. Hence, the STC shell has a factor of safety of nearly 10 against primary stress failure. Under the rules of NC-3200, the Level A allowable membrane stress intensity is 22,400 psi versus a calculated value of 2,144 psi for the same load conditions. Thus, the factor of safety associated with the STC shell is even larger according to the alternate rules of NC-3200.

2) The junction of the base plate and the STC shell has been made into a butt welded joint which exceeds the requirement of "NC" to allow the joint to be more easily examined. Indeed, the MPCs used in the industry use a corner joint (rather than the more robust and more easily examined butt weld joint). In this respect the STC exceeds the state-of-the-art MPC designs used in the industry.

3) The STC shell to the top flange joint, designated in the Code (NC and other code subsections) as a Category C joint and permitted to be made as a corner joint, has been made as a butt welded joint. In this respect also, the STC exceeds the MPC with respect to structural and confinement robustness.

4) To provide an additional degree of quality relative to the radiography requirements of ND, the STC shell thickness is sized to provide safety margins that are nearly an order of magnitude above code allowable for Subsection NC and ND and will be subjected to extensive radiography beyond the requirements of Subsection ND, but less than the 100% radiography required by Subsection NC. The STC confinement boundary welds were subject to radiographic examination exceeding the minimum NDE requirements of ND-5200 (i.e., spot radiography). Specifically, more than 30 radiographs were taken for the STC confinement boundary to establish the soundness of its weld seams All confinement boundary welds (confinement shell longitudinal and circumferential welds, shell to top flange weld, and shell to base plate weld) have been subjected to RT examination over all or a portion of their length.

5) To provide added confidence, the STC has been helium leak tested to the leak tight criteria of ANSI N14.5 in the manner of a transport cask for off-site shipment. This test was performed after the hydrostatic pressure test of the STC pressure boundary (minimum test pressure of 125% of design pressure) to provide additional confirmation that that no thru-wall structural defects exist in the confinement boundary.

6) The STC has been designed and manufactured as a Safety Related component under Holtec's Quality Assurance program (equivalent to Important-to-Safety Category A under Part 72) which is on par with the MPC and is the highest safety class that can be assigned to a confinement vessel. No exceptions have been taken to the Safety Related component requirements under 10CFR50 Appendix B.

Of course, the STC, like the MPC, cannot be stamped as a Code vessel because of its fundamental anatomical differences from a classical pressure vessel. However, as the above design features make it clear, the STC has been engineered and manufactured to quality standards that emulate "NC" of Code, and exceed its provisions in several material respects. Hence, the degree of quality for the STC is no less than that of a spent fuel storage or transfer canister constructed to ASME Subsection NC.

Finally, as asked in the RAI, the table provides the exceptions to "NC" that are applicable to the STC confinement boundary. This table has been prepared in the same format and manner as that used for the MPC in the part 72 dockets 72-1008 and 72-1014. Based upon the information provided above as well as the following table, Entergy concludes that the STC has been designed, fabricated, and inspected to a degree of quality that exceeds both Subsection NC and ND, and in most respects, that of the Holtec MPC used for both storage and transport of spent fuel.

LIST OF ASME CODE EXCEPTIONS FOR STC CONFINEMENT BOUNDARY

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures	Comparable to MPC under docket 72-1014	Meets NC	Meets ND
STC Confinement Boundary	Subsection NCA	General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.	Because the STC is not an ASME Code stamped vessel, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the STC licensing report, HI-2094289 includes the design criteria, service conditions, and load combinations for the design and operation of the STC as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All safety related activities are governed by the NRC-approved Holtec QA program. Because the STC is not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections ND, of the Code, as applicable, shall be interpreted to apply to the Entergy (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the Entergy and its vendors assigned to oversee and inspect the manufacturing process.	Yes	No	No

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures	Comparable to MPC under docket 72-1014	Meets NC	Meets ND
STC Confinement Boundary	NC/ND-1000	Statement of requirements for Code stamping of components.	Cask confinement boundary is designed, and will be fabricated in accordance with ASME Code, Section III, Subsection ND to the maximum practical extent, but Code stamping is not required.	Yes	No	No
STC Confinement Boundary	NC/ND-2000	Requires materials to be supplied by ASME-approved material supplier.	Holtec approved suppliers will supply materials with CMTRs per ND-2000.	Yes	No	No
STC Confinement Boundary	NC/ND-2300	Provides impact testing requirements for materials.	The STC confinement boundary materials have been impact tested with satisfactory results per ND-2300. The test results also exceed the required C_v values per NC-2300, except for the lateral expansion of the STC bolting material (SA-564 630 H1100). It is noted, however, that the impact testing of the STC bolting material has been conducted at 0°F, which is substantially below the Lowest Service Metal Temperature of the STC closure lid studs. The minimum metal temperature of the STC closure lid studs is bounded by the freezing temperature of the water inside the HI-TRAC. A more realistic temperature estimate for the STC closure lid studs during fuel transfer is at least 100°F. The increased service temperature of the STC closure lid studs, coupled with the low magnitude of stress in the studs (SF > 9), virtually eliminates the risk of brittle fracture.	No	No	Yes

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures	Comparable to MPC under docket 72-1014	Meets NC	Meets ND
STC Confinement Boundary	NC-3300/NC-5200	Requires full radiography of Category A and B welded joints, and Category C full penetration butt welded joints.	STC confinement boundary welds have been examined in accordance with ND-5200, which has provisions for partial and spot radiography. The use of ND-5200 is justified based on the large factor of safety associated with the STC confinement shell (SF > 9). As a compensatory measure, the STC confinement boundary has been helium leak tested to the leak tight criteria of ANSI N14.5.	No	No	Yes
STC Confinement Boundary	NC/ND-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the STC-to-HI-TRAC) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including intercomponent annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components	Yes	No	No
STC Confinement Boundary	NC/ND-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of cask vessel is as a radionuclide confinement boundary under normal and hypothetical accident conditions. Cask is designed to withstand maximum internal pressure and maximum accident temperatures.	Yes	No	No

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures	Comparable to MPC under docket 72-1014	Meets NC	Meets ND
STC Confinement Boundary	NC/ND-8000	States requirement for name, stamping and reports per NCA-8000	STC to be marked and identified in accordance with the drawing. Code stamping is not required. QA data package prepared in accordance with Holtec's approved QA program.	Yes	No	No

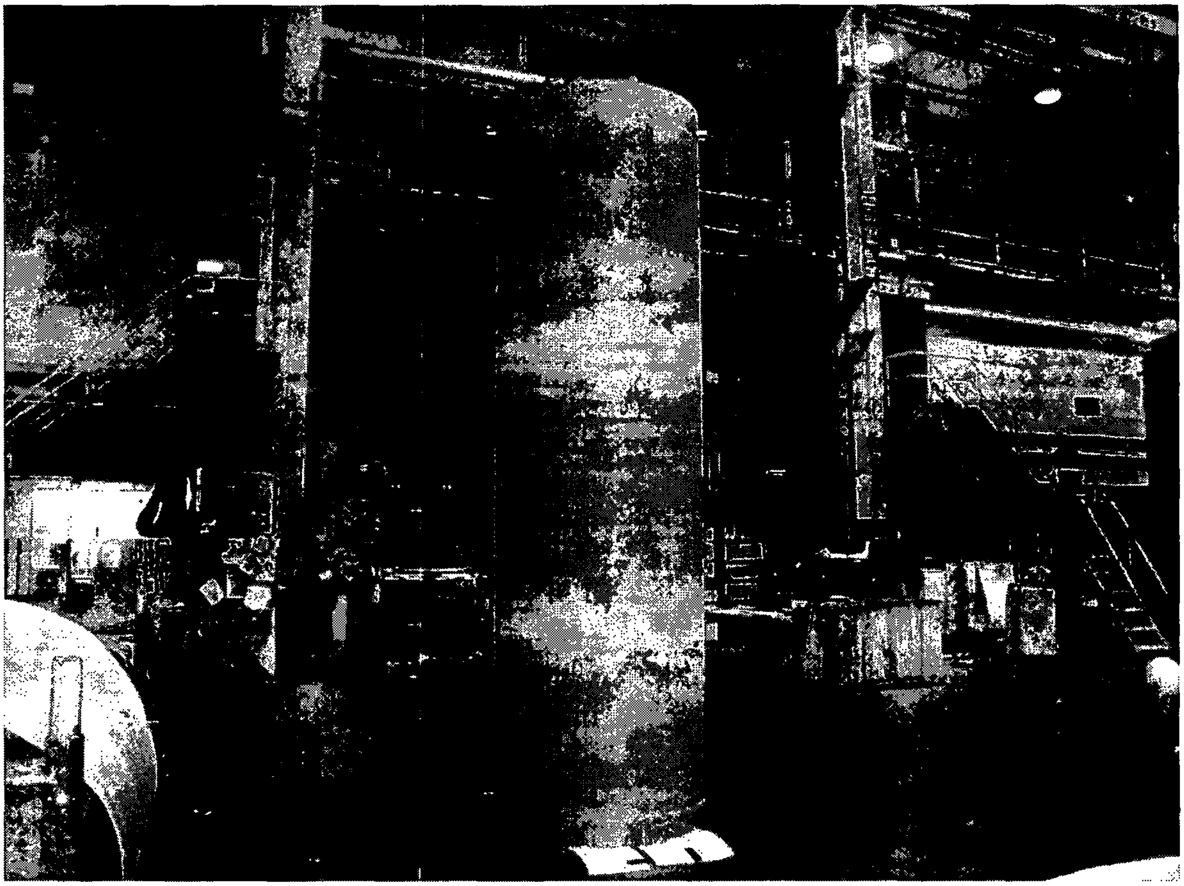

Figure 8-4.1 Completed Shielded Transfer Canister at Holtec Manufacturing Division

CHAPTER 10 - OPERATING PROCEDURES

NRC RAI 10-1

Regarding the operations descriptions, provide the following: (CSDAB)

a. Justification for removing water from the STC/HI-TRAC annulus at the currently described step in the unloading procedures. Considering the step in the loading procedures in which the annulus water level is raised (just after the STC lid bolting is tightened to the required levels) and the operations descriptions following the water drain down for the unloading operations, it seems more appropriate and in keeping with ALARA to not drain down the annulus water until just prior to loosening the STC lid bolts.
b. Modification of the operations descriptions to include the steps for installation and removal of the Bottom Missile Shield (BMS). The responses to the first round RAI questions 1-2 and 7-6, indicate that the BMS is to be installed with the HI-TRAC empty; however, Chapter 10 descriptions need to be updated to reflect

these operations and not just a check that the BMS is installed at the time the loaded HI-TRAC is to be moved.

c. Modification of the operations descriptions to account for the 24-hour pressure rise check. Other evaluations, such as occupational dose estimates, should be updated as necessary.
d. Modification of Section 10.5.5 of the report to include descriptions of steps to be taken in the event gas sampling indicates damage to assemblies. These steps would include such things as whether or not fuel is off-loaded and how it is handled.
e. Inclusion of the step for filling the HI-TRAC neutron shield.
f. Descriptions of how the presence of a walkway crossing over the haul path is addressed. Clarify whether and how the EAB also covers this walkway.
g. Explanation of the kinds of delays envisioned for Section 10.5.2 and the configurations that are considered (e.g., whether the STC is always in the HI-TRAC or the STC may be outside of the HI-TRAC). Evaluations of these configurations, as necessary, should also be provided.

This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a) and (b), 10 CFR 50.34, and the intent of 10 CFR 72.104.

Response to RAI 10-1

In the revised licensing report, the following changes have been made to the Chapter 10 operating procedures:

a. Section 10.4.1 has been modified such that the step for removing water from the STC/HI-TRAC annulus (10.4.1.10) is now located immediately before the step where the lid bolting is loosened during the unloading process.
b. Section 10.1.2 has been modified to include a step for installation of the BMS on the HI-TRAC (10.1.2.10). The BMS remains attached to the HI-TRAC and is only removed at the end of the loading campaign or if the pool lid seal needs repair or replacement.
c. Old Step 10.2.3.38, New Step 10.2.3.40 has been modified to include directions for monitoring the pressure rise in the STC. Monitoring of the STC pressure will be performed either remotely or from a low dose area and will not materially impact the estimated occupation dose totals.
d. Section 10.5.5 has been modified to provide additional guidance for unloading of fuel assemblies suspected of potential damage from an accident condition.
e. Section 10.1.3 has been modified to add a step to add water to the HI-TRAC neutron shield (10.1.3.5).
f. Steps 10.3.1.4 and 10.3.1.8 have been modified to include provisions for control of access to walkways which may cross over the haul path.
g. The types of delays envisioned for requiring water inventory control are as follows:

    - Failure of the hydraulic torque system used for the lid bolts that prevents the STC or HI-TRAC lids hardware from being properly torqued.
    - Crane malfunction which prevents the HI-TRAC lid from being installed in the HI-TRAC.
    - Leak testing system malfunction that prevents the STC lid seals from being leak tested prior to final closure for transfer.

For each of these delays, the STC will be located in the HI-TRAC. No additional evaluations or analysis are required for these configurations as they are all bounded by existing analysis of the system provided that the water inventory control is maintained.

NRC RAI 10-2

Regarding the dose rate measurements for the loaded STC and HI-TRAC, define and include the criterion/criteria used to determine when dose rates that exceed expected values may be acceptable and justify waiting to perform dose rate measurements on the HI-TRAC until the currently proposed step. (CSDAB)

The currently proposed operations descriptions for the dose rate measurements for the STC lid and the HI-TRAC side include performance of an evaluation to determine if higher dose rates are acceptable and fuel transfer can continue. However, it is not clear what criterion or criteria are used to make that determination. Such criteria should be defined and provided as part of the operations descriptions related to the measurement procedures. For example, a criterion used for determining the acceptability of the higher dose rates on the transfer cask or storage overpack in the TS Radiation Protection Program for the HI-STORM 100 system is a determination that the as-loaded MPC, considering its contents, the number of casks at the ISFSI, etc. will not cause the limits of 72.104 to be exceeded. Similarly, an appropriate criterion, or criteria, is needed for the currently proposed operations. Additionally, the basis for delaying the measurements on the HI-TRAC side until step 55 of Section 10.2.3, versus performing the measurements very shortly after the STC is placed in the HI-TRAC (e.g., after step 23 or 28), is not clear. If a problem arises that necessitates corrective actions, the various operations to prepare the STC and HI-TRAC for transfer will have to be undone with the current sequence of operations. This seems unnecessary and as well as to not meet the intent of ALARA. The dose rate measurements on the HI-TRAC side should be performed shortly after the STC is placed in the HI-TRAC.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.34 and the intent of 10 CFR 72.104.

Response to RAI 10-2

The steps describing the measurement of the dose rates at the side of the HI-TRAC have been moved up to immediately after the STC has been placed into the HI-TRAC to avoid the potential for additional dose accumulation due to a loading error. The dose measurement steps have been revised and refer to the Technical Specification for the expected dose rate limits and the requirements for determining whether any higher than expected dose rates are acceptable. The dose rate limits and the actions required to evaluate the dose readings and their acceptability for continued transfer of fuel between units are defined in Appendix C, Part II, Subsection 5.4 of the Technical Specification.

Specifically, Step 18 in Section 10.2.3 has been revised to read:

Perform a radiological survey of the STC lid and compare to the expected dose rates as referenced in Technical Specification Appendix C, Part II, Subsection 5.4, to ensure

there has not been a fuel mis-load. Dose rate measurements shall be taken at the locations described in the Technical Specification.

Step 19.c has been revised to read:

> Perform a written evaluation to determine (1) why the surface dose rate limits were exceeded, and (2) if the higher dose rate values are acceptable, fuel transfer can continue in accordance with controls expressed in Technical Specification Appendix C, Part II, Subsection 5.4.

Step 55 has been moved to Steps 23 and 24 and the text has been revised to read:

(Step 23) Perform surface dose rate measurements for the HI-TRAC and compare to the expected dose rates as referenced in Technical Specification Appendix C, Part II, Subsection 5.4. Dose rate measurements shall be taken at the locations described in the Technical Specification. Compare the measured dose rates with calculated dose rates for the design basis fuel to ensure they are less than expected.

(Step 24) If dose rates exceed expectations, perform the following:

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

b. Perform a written evaluation to determine (1) why the surface dose rate limits were exceeded, and (2) if the higher dose rate values are acceptable fuel transfer can continue in accordance with controls expressed in Technical Specification Appendix C, Part II, Subsection 5.4.

c. If the higher dose rate values are not acceptable, the STC will be returned to the spent fuel pool and a reload of the STC will be performed.

NRC RAI 10-3

In Table 10.0.1 of the SAR, “Operational Considerations,” add a row for “Crane Hang-up or Loss of Power.” This event is discussed on page 10-26, but is not listed in the table. Step 10.5.1.1.b should start with a requirement to perform radiation surveys to establish stay times for personnel, due to the high dose rates when the STC is suspended from the crane. (LPL1-1)

Response to RAI 10-3

An entry has been entered into Table 10.0.1 for Crane Hang-up or Loss of Power. Step 10.5.1.1.b has been modified to include instructions to perform radiation surveys and establish radiological controls in the area around the STC.

NRC RAI 10-4

In Section 10.5.2 of the SAR, it states that water shall be circulated through the STC daily to insure that the STC internal cavity is filled. If the 24 hour pressure rise test is in progress, circulating water or venting will violate the test conditions. Add an exception to water circulation and venting for the pressure rise test. (LPL1-1)

Response to RAI 10-4

An exception has been added to Step 1 of Section 10.5.2 of the licensing report to exempt the water circulation requirements during the execution of the 24 hour pressure rise test.

Specifically, Step 1 of Section 10.5.2 now includes the following sentence at the end:

The above requirements are not applicable during the 24 hour pressure rise test.

TECHNICAL SPECIFICATIONS:

NRC RAI TS-1

Add a TS condition that the restricted area boundary for the transfer operation is a minimum of 20 meters (or the distance used in the evaluations, as modified in response to this RAI) from the haul path. (CSDAB)

The radiation protection evaluation relies upon a set minimum distance to separate members of the public on site from the transfer operations to show compliance with 10 CFR 20.1301(b). This distance is a significant parameter in the evaluation and should be appropriately controlled. The applicant refers to this distance, or the boundary at this distance, as the exclusion (area) boundary (see Section 7.4.6 of Report HI-2094289). This terminology does not seem to be correct for this particular activity. The terminology should be made consistent with 10 CFR Part 20, using the terminology defined in that part of the regulations for actions performed and controls that are set for purposes of radiation protection.

This information is needed to confirm compliance with 10 CFR 20.1301(b).

Response to RAI TS-1

A TS condition has been added to proposed TS Section 4.1.4 to state that a restricted area boundary will be established at a minimum distance of 20 meters from the haul path.

The minimum distance of 20 meters from the haul path is supported by the analyses described in Section 7.4.7 of the Licensing Report where compliance with 10 CFR 20.1301(b) is demonstrated. The term exclusion (area) boundary is no longer used in this context in the licensing report (HI-2094289) having been replaced by the 10 CFR 20.1301(b) terminology of restricted area.

Also note that in response to RAI 7-8 a TS condition has been added to proposed TS Section 4.1.4 that says "During the movement of the STC from the SFP to the HI-TRAC

and from the HI-TRAC to the SFP a restricted area boundary shall be established at a minimum distance of 60 meters from the fuel storage building when the STC contains one or more fuel assemblies."

NRC RAI TS-2

Provide the following clarifications and modifications of the proposed TSs, Appendix C, Parts I and II. (CSDAB)

a. Clarify the apparent inconsistency between TS, Appendix C, Part II, 4.1.4.6 and SAR Section 10.2.3.37, modifying the appropriate location to give the correct tolerances on the STC water level.
b. Include the STC licensing report (SAR), Chapter 10 as part of the basis in Appendix C, Part I, Section 2.1 and SAR Chapter 8 as part of the basis in Appendix C, Part I, Section 2.2. These sections of the SAR form the basis for the operations, acceptance tests, and maintenance program directly related to the STC and the transfer operations.
c. Modify LCO 3.1.2.b (Appendix C, Part II) to include an item 4 that duplicates item 5 of LCO 3.1.2.a and adds that rod control cluster assemblies (RCCAs) and NSAs cannot be loaded in the configuration of LCO 3.1.2.b. Appropriate Non-Fuel Hardware (NFH) loading restrictions, supported by the licensing report evaluations, are needed for both loading configurations.
d. Clarify the meaning of Note 3 to LCO 3.1.4 (Appendix C, Part II) with regard to defining a given assembly's burnup, especially with respect to the proposed limits in LCO 3.1.2. It is not clear from this note that assemblies with burnup/exposure greater than 55 GWD/MTU may not be loaded in the STC (since the note recalculates the burnup if the assembly had hafnium inserts). The maximum allowable burnup in the TS should be supported by all the appropriate STC licensing report evaluations (e.g., shielding, etc.).
e. Add the minimum specifications of the HI-TRAC neutron shielding to the HI-TRAC description in TS, Appendix C, Part I, Section 1.0. The neutron shielding features are also an important aspect of the HI-TRAC.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104 and 72.126.

Response to RAI TS-2

The following clarifications and modifications of the proposed TSs, Appendix C, Parts I and II are included in the proposed TS accompanying this response:

a. The inconsistency between TS, Appendix C, Part II, 4.1.4.6 and SAR Section 10.2.3.37, has been resolved. The proposed TS, now LCO 3.1.3, has been revised to state that the STC water level shall be 9.0 +0.5/-1.5 inches below the bottom of the STC lid. The associated step in the SAR, now 10.2.3.39, also states this water level requirement.

b. The STC licensing report (SAR), Chapter 10 has been listed as part of the basis in Appendix C, Part I, Section 2.1 and SAR Chapter 8 has been listed as part of the basis in Appendix C, Part I, Section 2.2.

c. LCO 3.1.2.b (Appendix C, Part II) has been modified to include an item 3 that duplicates the equivalent item of LCO 3.1.2.a (now item 4) and also adds that Rod Control Cluster Assemblies (RCCAs), Neutron Source Assemblies (NSAs), and Hafnium suppressors cannot be loaded in the configuration of LCO 3.1.2.b.

   Appropriate Non-Fuel Hardware (NFH) loading restrictions, supported by the licensing report evaluations of Chapters 4 and 7, have been added to proposed LCO 3.1.2, and are included in Table 3.1.2-2 "NON FUEL HARDWARE Post Irradiation Cooling Times and Allowable Average Burnup." Table 3.1.2-2 places loading restrictions on BPRAs, WABAs, TPDs, RCCAs, Hafnium Suppressors. and NSAs.

d. Note 3 to LCO 3.1.5 (Appendix C, Part II) has been clarified with regard to defining a given assembly's burnup, with respect to the proposed limits in LCO 3.1.2. This note only applies to the unloading of the fuel from the STC into the Unit 2 spent fuel pool when considering burnup for the application of burnup credit. The revised Note 3 now says, "For fuel assemblies exposed to Hafnium inserts during irradiation the burnup of the assembly used to classify the fuel for unloading into the IP2 spent fuel pit shall be the burnup prior to the exposure to the Hafnium insert."

   Note that the fuel to be loaded into the STC will be limited to the maximum burnups specified in LCO 3.1.2. As discussed in the response to TS-6 the proposed TS limit the radiation source term of the allowable STC contents by limiting, in part, the maximum allowable burnup.

e. The minimum specifications of the HI-TRAC neutron shielding have been added to the HI-TRAC description in TS, Appendix C, Part I, Section 1.0.

NRC RAI TS-3

Modify the proposed TS, Appendix C, Part I to: (CSDAB)

a. Include recovery from off-normal conditions (such as crane hang-up) in the operations listed in Section 2.1 and
b. Include manual crane operation and crane recovery/repair as part of Section 2.3.

The dose rates from the bare STC (analytical) are significant (even for the representative loading), much more so than has been seen for nearly all spent fuel loading operations to date. Therefore, staff is particularly concerned about a potential off-normal event such as the hang-up of the crane or a malfunction of the equipment allowing personnel to remain at significant distances from the STC during operations with the STC outside the pool and the HI-TRAC. Crane hang-ups are not uncommon, especially when the cranes are loaded with weights approaching their capacity. Thus, operations to recover from a crane hang-up with a loaded STC could provide significant occupational exposures. Procedures for such scenarios should therefore be developed beforehand and appropriate training provided. Inclusion of manual crane operation in the dry run will assure the licensee can effectively operate the crane manually in a high dose rate environment as well as inform the licensee's predictions of potential worker dose in the event of a crane hang-up or other off-normal event.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104 and 72.126.

Response to RAI TS-3

In recognition of the importance of recovery from a crane hang up, that could include manual crane operation, procedures will be developed and training will be provided to support such a recovery. In addition, demonstration of manual crane operation will be included in the dry run to be performed prior to any fuel transfer operations. Accordingly, the proposed TS, Appendix C, Part I has been modified to:

a. Include recovery from all off-normal conditions in the operations listed in Section 2.1 Operating Procedures, and
b. Include manual crane operation and crane recovery/repair as part of Section 2.3 Pre-Operational Testing and Training Exercise.

NRC RAI TS-4

Provide a TS dose rate limit and measurement requirement for the top lid of the STC and side of the HI-TRAC. (CSDAB)

Requirements that establish dose rate limits and the necessary measurements are a feature of the TS associated with 10 CFR Part 72 dry storage loading operations (including the HI-STORM 100), which are similar to the loading operations performed for the proposed wet transfer system. TS dose rate limits and measurements provide assurance of correct contents loading, ensure operations and ALARA planning is still adequate/appropriate for a given loading operation, and ensure that conditions outside those assumed for the design and operations are identified and properly handled to assure protection of personnel and members of the public.

Considering that the dose rates for even the proposed contents are very substantial and that conditions of the kind noted herein could make them even more substantial, a TS should be established that provides dose rate limits for the STC lid and the HI-TRAC side, an appropriate measurement scheme for each limit, and the appropriate corrective actions for instances where the limits are exceeded. The limits should be derived from the analysis for the representative loading and should capture areas of the STC and HI-TRAC of significance to occupational and public dose. Given the similarities of the proposed system to dry storage transfer systems, staff anticipates that the limits and measurements will be established in a similar fashion as for dry storage transfer systems, with appropriate consideration for differences versus those systems and the applicant's analyses serving as the basis.

If it is proposed that dose rates that are higher than the proposed limits may be evaluated and considered acceptable under appropriate circumstances, the process and criteria for finding higher dose rates acceptable and allowing continuance of operations should be included in the TS and appropriately justified. For example, for dry storage, higher dose rates trigger a verification of correct loading and, if the loading is correct, a determination of whether or not 10 CFR 72.104 limits can be met for the loaded MPC. Since this is a wet transfer between pools, a criterion (such as ensuring compliance with 10 CFR 20.1301(b)) in addition to 72.104 limit compliance, considering the number of transfers to be performed in a given year, may be appropriate.

This information is needed to confirm compliance with 10 CFR 50.34, 10 CFR 20.1101 and 20.1301(a) and (b), and the intent of 10 CFR 72.104.

Response to RAI TS-4

TS dose rate limits and measurement requirements for the top lid of the STC and side of the HI-TRAC have been developed and are included in proposed TS 5.4 Radiation Protection Program. The limits have been derived from the analysis for the representative loading with the highest expected dose rates and capture areas of the STC and HI-TRAC of significance to occupational and public dose. These limits and measurements have been established in a similar fashion as for dry storage systems.

It is proposed that dose rates that are higher than the proposed limits may be evaluated and considered acceptable under appropriate circumstances. The process and criteria for finding higher dose rates acceptable and allowing continuance of operations are included in proposed TS 5.4 and are appropriately justified. Higher dose rates trigger a verification of correct loading (TS 5.4.4.a.) and, if the loading is correct, a determination of whether or not the transfer can proceed without exceeding the dose limits of 10 CFR 72.104 or 10 CFR 20.1301 (TS 5.4.4.b.).

The fuel transfer dose is listed separately in the site boundary dose report prepared in accordance with 10 CFR 72.104. The dose is considered part of the plant's operational contribution to the site boundary dose. Currently the site boundary dose report assumes that there are 16 inter-unit fuel transfers per year with a bounding dose for each transfer. If dose rate measurements taken in accordance with Section 5.4 are exceeded an evaluation is expected which will consider the assumptions already in the site boundary dose report to ensure that the yearly limit will not be exceeded. Similarly, the dose rate limits in 10 CFR 20.1301 will also be considered and an evaluation performed to ensure that occupational and public doses will not be exceeded.

NRC RAI TS-5

Modify the fuel specifications in proposed TS, Appendix C, Part II, Table 4.1.1-1 to accurately reflect the contents (as evaluated in the amendment application) that will be loaded into the STC, including the following changes. (CSDAB)

a. Include the fuel cladding material. The shielding evaluation only supports zirconium-based cladding; thus, this should be the listed cladding.
b. Change the fuel rod clad I.D. to be a maximum value. It is currently shown as a minimum value, which appears to be incorrect. A minimum cladding thickness should be established, which only can be done with the clad I.D. set at a maximum and the clad O.D. set as a minimum.
c. The correct maximum active fuel length should be given. The shielding evaluation, for example, indicates the maximum active fuel length is 144 inches.
d. The correct fuel assembly maximum length should be given. The current length (176.8 inches) given in the table does not physically fit in the STC, which the licensing drawings indicate has a cavity length of only 168 15/16 inches. Further, the inclusion of NFH extends the needed STC cavity length to be able to contain the proposed contents.

e. If the active length and assembly length are accurate, then the STC design should be revised and the evaluations modified accordingly to support the new design and the proposed contents.

This information is needed to confirm compliance with 10 CFR 50.34 and the intent of 10 CFR 72.104.

Response to RAI TS-5

The fuel specifications in proposed TS, Appendix C, Part II, Table 4.1.1-1 have been modified to accurately reflect the contents (as evaluated in the amendment application) that will be loaded into the STC, including the following changes:

a. The fuel cladding material.
b. The fuel rod clad I.D. to be a maximum value.
c. The correct maximum active fuel length is given. The fuel along with NFH will physically fit in the STC. Fuel spacers are typically used in MPCs for dry storage and transportation to keep the active length of the fuel aligned with the neutron absorber panels in the basket during horizontal handling and eventual transport of the MPC. In the horizontal position it may be possible for the fuel to move up towards the top of the MPC and outside of the envelope of the neutron absorber panels. Fuel spacers are not required for the STC since there are no instances where the STC will be in horizontal position.
d. The correct fuel assembly maximum length is given.
e. No design changes and no additional evaluations are necessary based on responses to c. and d. above.

NRC RAI TS-6

Propose TS limits on maximum burnup, minimum enrichment and minimum decay/cooling times to limit the radiation source term of the contents and provide supporting quantitative evaluations to justify the proposed limits. Otherwise, provide additional justification, including appropriate quantitative evaluations, that the currently proposed TS with respect to allowable STC contents are sufficient to limit the radiation source of those contents. (CSDAB)

The currently proposed operation is similar to loading operations for dry storage systems. TS for these systems, including the HI-STORM 100, define the allowable contents in terms of the maximum burnup, minimum enrichment and minimum decay time of the assemblies. That being the case, the applicant has proposed a very different approach to limit the contents. The applicant has provided some justification in response to the first round RAI question 10-5. However, no quantitative evaluation was provided to support the statements made in that response. While the applicant has included a source term that significantly exceeds the decay heat limits for comparison as part of the licensing report, this does not indicate the variation in dose rates that will exist between assemblies of different burnup, decay time, and minimum enrichment that have the same decay heat. Adequate justification would include evaluations that show how the dose rates vary for an adequate variety of assembly burnup, enrichment and decay time combinations that result in the same decay heats for the different STC basket regions.

Given that radiation source and decay heat do not correlate on a one-to-one basis and the significance of the STC dose rates for the 'representative' loading, it is important to ensure the TS properly define the allowable radiation source in the STC. A decay heat limit alone would allow for radiation sources of varying strengths that could result in dose rates that may be different enough to necessitate non-trivial changes to operation procedures and controls for the purposes of occupational and public radiation protection and significantly impact off-normal conditions and accident conditions exposures as well. Additionally, it is not clear how using a decay heat limit alone would be practically implemented, including independent verifications, considering that decay heat is a calculated value that is derived from an assembly's enrichment, cooling/decay time and burnup.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.90 and 50.34a(c) and the intent of 10 CFR 72.104, 72.106 and 72.44(c)(1).

Response to RAI TS-6

TS limits on maximum burnup, minimum enrichment and minimum decay/cooling times are proposed to limit the radiation source term of the allowable STC contents.

The analyzed loading configurations are provided in the Table TS-6-1 below (Table 7.1.1 of the licensing report), and bound the majority of the current Unit 3 spent fuel pool inventory. The remaining assemblies will only be transferred when their cooling times meet the requirements of Table TS-6-1. As determined in Section 7.4 of the licensing report, loading pattern 3 or 4 results in the bounding dose rates depending on the dose locations and whether the STC is outside or inside the HI-TRAC.

The source terms were applied in a regionalized loading scheme to the 12 fuel assembly locations available in the STC. The regionalized loading pattern was utilized to facilitate the transfer of hotter fuel in the spent nuclear fuel storage pool by taking advantage of self-shielding effects. As can be seen in Table TS-6-1 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 times are assigned to the four inner fuel assembly locations.

Therefore, it is proposed to revise LCO 3.1.2 to include the limits on maximum burnup, minimum enrichment and minimum decay/cooling times specified in Table TS-6-1. This Table has been added to LCO 3.1.2 as Table 3.1.2-3.

Table TS-6-1

Allowable STC Loading Configurations

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

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

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

NRC RAI TS-7

Justify the lack of a TS surface contamination LCO for the STC. (CSDAB)

Operations include the immersion of the STC in the spent fuel pool (SFP). While the STC is washed down prior to and as it is removed from the SFP, this is to minimize contamination. This is the same kind of practice taken with the transfer cask for spent fuel loading operations for dry storage and there is normally a TS LCO limiting surface contamination (see TS 3.2.2 for the HI-STORM 100 system). The bases for that LCO state the LCO "allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination." While the HI-TRAC for the

currently proposed operations does not enter the SFP, the STC acts as a transfer cask for the loading and unloading operations in its use out of the HI-TRAC. Given this consideration and the leak rates allowed for the HI-TRAC, a contamination LCO may be appropriate.

This information is needed to confirm compliance with 10 CFR 20.1101(b), 10 CFR 50.34, and the intent of 10 CFR 72.44(c)(1), 72.104, and 72.126(a).

Response to RAI TS-7

In the HI-STORM 100 System LCO 3.2.2 addresses transfer cask surface contamination. LCO 3.2.2 notes that the LCO is not applicable to the transfer cask if it remains in the fuel storage building. In this situation the outside surface of the transfer cask would not be exposed to the environment. An equivalent to LCO 3.2.2, though, is not applicable to the STC as the STC is not a final dry fuel storage canister, nor is it subject to weathering. By virtue of its intended use, the STC outside surface would not be exposed to the environment even after immersion in either spent fuel pool. When the STC leaves a building it is confined within the HI-TRAC.

As noted in the RAI, the basis for the HI-STORM 100 System LCO 3.2.2 is that the LCO "allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination." As also noted in the RAI, the STC is washed down as it is removed from the SFP, to minimize contamination. Entergy has not proposed an LCO equivalent to the HI-STORM 100 System LCO 3.2.2 due to ALARA concerns that arise from performing the decontamination activity together with the fact that the prevention of the spread of contamination and the reduction in personnel dose are concerns applicable to the immediate transfer activity and not to long term storage. Radiological controls will remain in effect during the transfer of fuel. Large area swipes will be taken of the STC and HI-TRAC surfaces in accordance with procedural radiological controls specifically developed for this activity. The intent of these actions, monitoring and controls is to reduce and detect loose contamination and particles and allow timely and appropriate personnel protection action. The level of effort of these contamination and particle surveys will be in accordance with operational ALARA principles.

Contamination in the form of particles could be transferred from the surface of the STC to the water in the annulus space inside the HI-TRAC; however, the HI-TRAC pressure boundary will be tested to ensure that it is water tight to prevent the loss of water from the annular region during fuel transfer operations. This testing is described in Section 8.4 of the licensing report. Following completion of an inter-unit fuel transfer campaign Entergy expects to have the HI-TRAC restored to such a condition to support all dry transfer operations and conditions of that mode of operation.

NRC RAI TS-8

Revise the proposed TS 3.1.3 to ensure a correct measurement of the pressure rise in the STC after it is loaded with spent fuel. (LPL1-1)

Measurement of the pressure change in the STC cavity after the STC is loaded is used to verify that the decay heat load is within design parameters. The staff's understanding

of the pressure rise monitored by proposed LCO 3.1.3 is that the pressure rise is the difference between the lowest observed absolute pressure in the STC and the absolute pressure in the STC at the end of the 24 hour period. That description of the pressure rise should be added to TS Bases 3.1.3. Also, since the staff expects the pressure in the STC to initially decrease after the water level is established, SR 3.1.3.1 should be revised to say, "Once upon establishing required water level AND hourly thereafter." Taking at least 25 data points over the 24 hour period will allow more accurate checking of the pressure change. Also, SR 3.1.3.2 should be added to specify the pressure instrumentation to be used. Wording similar to the following is needed: "Verify that two channels of pressure instrumentation with a range of at least 1 psia to 75 psia, and calibrated to within 2% accuracy within the past 12 months, are installed on the STC." The Frequency of SR 3.1.3.2 could be "During performance of surveillance 3.1.3.1."

This information is required for compliance with 10 CFR 50, Appendix A, GDC 61.

Response to RAI TS-8

The proposed TS (now LCO 3.1.4) has been revised to ensure a correct measurement of the pressure rise in the STC after it is loaded with spent fuel. As discussed in the responses to RAIs 1-1, 5-5, and 5-6, it is proposed to detect a significant fuel misload by measuring the rate of pressure rise over a 24 hour period. LCO 3.1.4 has been revised to say, "The pressure rise in the STC cavity shall be ≤ 0.2 psi/hr averaged over a rolling 4 hour period."

A rate of STC pressure rise above 0.2 psi/hr during any rolling 4 hour period would be an indication that the design basis of the STC is not being met. This indication could be due to one or more of the following conditions:

a. STC water level not within limit,
b. one or more fuel assemblies misloads,
c. presence of air in the STC cavity.

Should the acceptance criterion of 0.2 psi/hr not be met the STC would be depressurized and actions taken to verify: the STC water level, that a fuel misload had not occurred, and that air had been effectively precluded from entering the STC. Once the cause of the non conforming condition has been identified and corrective actions taken, SR 3.1.4.1 would need to be re-performed satisfactorily prior to STC transfer operations.

Measurement of the rate of pressure rise in the STC cavity after the STC is loaded is used to verify that fuel transfer can proceed within the established thermal design basis. It is proposed to take 25 data points over the 24 hour period. SR 3.1.4.1 has been revised to say, "Verify by direct measurement that the rate of STC cavity pressure rise is within limit." A Note has also been added to the SR that says, "Pressure measurements shall be taken once upon establishing required water level AND hourly thereafter for 24 hours. Pressure may initially drop during pressure stabilization."

SR 3.1.4.2 has been added to specify the pressure instrumentation to be used and that an ASME Code compliant pressure relief valve or rupture disc must be installed during the test. The proposed SR says, "Verify that an ASME code compliant pressure relief valve or rupture disc and two channels of pressure instrumentation with a range of at

least 0.1 psia to 15 psia and calibrated to within 1% accuracy within the past 12 months are installed on the STC." The frequency of SR 3.1.4.2 is "During performance of SR 3.1.4.1."

Note that two sets of pressure instrumentation are required for the operation of STC as shown in Table 10.1.1. The use of each is described in Chapter 10. The pressure rise gauges are as specified above and are used during the LCO Surveillance. The other set of pressure gauges also referred to as the "coarse" gauges, are used along with the ASME code compliant relief valve or rupture disk during leak testing and other operations involving the STC to ensure no over-pressurization of the STC occurs.

NRC RAI TS-9

Throughout the TS, "non fuel hardware" is a defined term and should therefore be capitalized. (LPL1-1)

Response to RAI TS-9

It is recognized that "non fuel hardware" is a defined term and the proposed TS have been revised accordingly.

NRC RAI TS-10

SR 3.1.1.1 says to "Verify the STC boron concentration is within limit using two independent measurements." Please explain the independent measurements to be used. If they are not truly independent (such as titration and neutron absorption), then it may be better to describe them as "two separate measurements." (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-10

The STC boron concentration will be verified using two separate measurements and proposed SR 3.1.1.1 has been revised to say, "Verify the boron concentration is within limit using two separate measurements." Note that LCO 3.1.1 has been revised to include the boron concentration of the water in the spent fuel pit in addition to the STC.

NRC RAI TS-11

TS 4.1.4.3 says "LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains irradiated fuel only." Consider revising this to say "LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel." This is more precise, since there typically are items other than irradiated fuel in the spent fuel pit. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-11

The spent fuel pit typically does contain items other than irradiated fuel, therefore, proposed TS 4.1.4.3 has been revised to say, "LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel assemblies."

NRC RAI TS-12

TS 4.1.4.8 says "TRANSFER OPERATIONS shall only be conducted when the HI-TRAC water level is within +0/-1 inch of the top of the STC lid and the water level has been independently verified." Consider revising this to say "TRANSFER OPERATIONS shall only be conducted when the HI-TRAC water level is within +0/-1 inch of the top of the STC lid prior to installing the HI-TRAC lid and the water level has been independently verified." It is obvious that as the water heats up in the HI-TRAC it will expand and will no longer be within the specified range. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-12

The HI-TRAC water level is established and independently verified prior to installing the HI-TRAC lid, therefore, the proposed TS, which is now 4.1.4.9, has been revised to say, "Prior to installing the HI-TRAC lid the HI-TRAC water level shall be verified by two separate inspections to be within +0/-1 inch of the top of the STC lid."

NRC RAI TS-13

TS 5.2(iii) says "Four coupons will be tested at the end of each inter-unit fuel transfer campaign." Since the duration of a campaign is not defined, please propose a time frame instead, such as every 2 years. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-13

Proposed TS 5.2(iii) has been revised to say, "Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained."

NRC RAI TS-14

The Note in SR 3.1.1. says that the surveillance is only required to be performed if the STC is submerged in water. Since the STC is submerged in water while it is in the HI-TRAC, the boron concentration would have to be checked every 48 hours. If the STC had to be left in the HI-TRAC for an extended period of time, for example, due to an equipment malfunction, it will be very difficult to get a sample for boron analysis. Recommend revising the wording to something similar to "This surveillance is only required to be performed if the STC is submerged in water with the STC lid not fastened...". (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-14

Originally proposed SR 3.1.1 is modified by a Note indicating that the surveillance is only required to be performed if the STC is submerged in water or if water is to be added to, or recirculated through, the STC. These are the only times when a change in STC boron concentration could potentially occur. In order to preserve the assumptions of the criticality analysis, the Note further requires that water added to, or recirculated through, the STC must meet the boron concentration requirements of LCO 3.1.1. This Note does not apply to the addition of steam to the STC as discussed in the licensing report.

In recognition of the fact that the STC could be submerged in water in the HI-TRAC for an extended period of time without potential for a change in boron concentration, the Note in proposed SR 3.1.1 has been revised to say, "This surveillance is only required to be performed if the STC is submerged in water in the spent fuel pool or if water is added to, or recirculated through, the STC when the STC is in the HI-TRAC. Any added water must meet the boron concentration requirement of LCO 3.1.1."

NRC RAI TS-15

LCO 3.1.1 would allow the licensee to sample the boron in the STC to verify at least 2000 ppm, and then lower the STC into the IP3 spent fuel pit with the water in the spent fuel pit at 1000 ppm boron. This allows the possibility of diluting the STC boron. Either revise Appendix A, LCO 3.7.15, to require 2000 ppm in the IP3 spent fuel pit whenever fuel assemblies are in it, or add an LCO to Appendix C to verify 2000 ppm in the IP3 spent fuel pit prior to placing the STC in the spent fuel pit. Revise SAR section 1.4, and other sections as necessary, to reflect this change. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-15

The intent of LCO 3.1.1 and the associated SR was not to allow sampling of the STC boron concentration prior to lowering the STC into the IP3 spent fuel pit. Rather, the intent was to demonstrate that the STC boron concentration, when the STC is submerged in the IP3 spent fuel pit, is greater than or equal to 2000 ppm, prior to loading fuel into the STC.

In order to clarify this intent the Note to SR 3.1.1.1 has been revised to read, "This surveillance is only required to be performed if the STC is submerged in water in the spent fuel pool or if water is added to, or recirculated through, the STC when the STC is in the HI-TRAC. Any added water must meet the boron concentration requirement of LCO 3.1.1." The specified frequency of SR 3.1.1.1 is "Once, within 4 hours prior to entering the applicability of this LCO and once per 48 hours thereafter." This frequency, together with the LCO, ensures that the boron concentration of the water in the STC in the spent fuel pool is verified to be greater than or equal to 2000 ppm, prior to loading fuel into the STC.

In order to ensure that the boron concentration of the water in the spent fuel pit is not reduced below that assumed in the STC criticality analysis LCO 3.1.1 has been revised

to say, "The boron concentration of the water in the Spent Fuel Pit and the STC shall be ≥ 2000 ppm." In addition, it is proposed to revise Unit 3 LCO 3.7.15, "Spent Fuel Pit Boron Concentration" by adding the following Note: "During inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration.""

Sections 10.2 and 10.4 Inter-Unit Transfer Operations of the licensing report has been revised to clarify when and where boron concentration measurements are taken.

NRC RAI TS-16

TS 4.1.2.1 is insufficient to specify the criticality controls, as it does not specify the location or size of the Metamic panels. One method would be to incorporate a drawing by reference which shows that information. Also, TS 4.1.2.1.g refers to B-10 loading in the B_4C as greater than or equal to 18.4 %. This can be misinterpreted, as 18.4 weight percent of the B_4C is not B-10. Please revise it to specify the B-10 density in the B_4C in terms of an areal density (grams per square centimeter). (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-16

The parameter most important for the criticality control function of the neutron absorber is the so-called B-10 areal density, i.e. the amount of B-10 per unit area of the absorber panel (usually specified as gm B-10/cm^2). While this parameter can be measured in the final product (via neutron attenuation testing), it is not a direct input into the manufacturing process. However, the value is the mathematical product of three input and process parameters, namely the B_4C weight percent of the material, the percent B-10 in the Boron in the B_4C, and the thickness of the panel, together with an appropriate proportionality constant (See Section 8.2 of the licensing report).

To provide a robust and conservative acceptance criteria approach, each of these three parameters is controlled independently in the manufacturing process and each parameter must independently meet a specified minimum required value.

This approach essentially guarantees that the panels exceed the required areal density. Since the approach uses a worst-case combination of the minimum value for each parameter, no statistical evaluation or criteria is required. The specific requirements are therefore:

- All lots of B_4C will contain boron with an isotopic B-10 content of at least 18.4%.
- The B_4C content in METAMIC® shall be greater than or equal to 31.5 and less than or equal to 33.0 weight percent. (on drawing)
- The Metamic panel thickness must be no less than the minimum thickness specified, 0.102 in. (on drawing)

These three measurable parameters can be verified by the documentation packages and QA records for the panels used in the STC. This assures that the minimum B-10 areal density of 0.031 g/cm^2 is achieved. The above three parameters are given in the STC basket drawing 6015 along with the size and location of the panels on the basket. Proposed TS 4.1.2.1.i has been added to reference drawing 6015 for the size and location of the panels and says "The size and location of the neutron absorber panels

shall be in accordance with drawing 6015 of the Licensing Report (Holtec International Report HI-2094289)."

Proposed TS 4.1.2.1.g has been modified to state more clearly that the B_4C will contain boron with an isotopic B-10 content of at least 18.4% and now says "The B_4C in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%".

NRC RAI TS-17

One of the operating conditions of the STC is the possibility of leaving some fuel assembly locations open in order to allow the loading of fuel which does not meet the minimum burnup requirements in the remaining locations. This is, therefore, an initial condition of a design-basis accident (fuel misload). This should be controlled by a TS. Please revise the Note for LCO 3.1.2 to say that if one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, and 4 must be empty, with cell blockers installed that prevent inserting fuel assemblies. Also propose a new SR 3.1.2.2 that verifies by visual inspection that cell blockers are installed on cells 1, 2, 3, and 4 prior to placing a Type 1 fuel assembly in the STC. Add the description of the cell blockers to the SAR. (LPL1-1)

This information is required for compliance with 10 CFR Part 50, Appendix A, GDC 61.

Response to RAI TS-17

In order to allow the loading of fuel assemblies that do not meet minimum burnup requirements (Type 1 assemblies) some STC fuel assembly locations are left open as specified in proposed LCO 3.1.2. Therefore, in order to preclude the physical placement of fuel assemblies in these open locations, a cell blocker will be installed as necessary. The Note for LCO 3.1.2 has been revised to say, "If one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, and 4 must be empty, with a cell blocker installed that prevents inserting fuel assemblies."

A new SR 3.1.2.2 is also proposed that says, "Verify by visual inspection that a cell blocker which prevents inserting fuel assemblies into cells 1, 2, 3, and 4 of the STC is installed." The frequency of the proposed SR is "Prior to placing a Type 1 fuel assembly in the STC."

A figure depicting a typical cell blocker has been added to Chapter 10 of the Licensing Report, Figure 10.2.