ATTACHMENT 1 TO NL-11-118

Response to Request for Additional Information
(HOLTEC NON PROPRIETARY)

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

REQUEST FOR ADDITIONAL INFORMATION (RAI)
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. ML091940176 (proprietary), ML091940177 (non-proprietary) and ML091940178 (non-proprietary), as supplemented by letters dated September 28, 2009, (ADAMS Accession Nos. ML092950460 (proprietary) and ML093020080 (non-proprietary)), October 5, 2010 (ADAMS Accession Nos. ML 102910508 (proprietary), ML103080112 (non-proprietary), and ML103080113 (non-proprietary)) and July 28, 2011 (ADAMS Accession Nos. ML 112200258 (proprietary), and ML112430437 (non-proprietary)), 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 4, ADAMS Accession No. ML11243A220 (non-proprietary). The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has the following questions:

CHAPTER 4 -CRITICALITY EVALUATION (CSDAB and SRXB)

NRC RAI 4-1

In Section 6.2.8 of the SAR it describes the fuel assemblies sliding to the STC lid in the tipover analysis. As this moves the fuel assemblies away from the neutron absorber in the cells, the NRC staff needs further information on this configuration. The NRC staff is concerned that a criticality event could add more heat than the analyzed heat load, resulting in pressurization and failure of the confinement boundary, as well as the hazard of a criticality event to workers in the area. Provide a criticality analysis for this configuration or redesign the STC internals (perhaps use a spacer) to ensure the fuel assemblies will not slide towards the STC lid during the postulated tip-over accident. If a criticality analysis is provided, it should take into consideration the fact that the upper part of the fuel assembly may be more reactive than the average of the entire fuel assembly.

This information is required to demonstrate that the system can withstand the postulated accidents and successfully maintain subcriticality, in compliance with General Design Criteria (GDC) 61, GDC 62, and the intent of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 72.122.

Note that additional Chapter 4 RAIs will follow in a subsequent letter.

Response to RAI 4-1

In order to address the criticality concerns related to a tip-over event additional criticality calculations have been performed. The following conservative assumptions and conditions were applied in the analyses:

- The maximum misalignment between the poison in the basket and the active region would exist if the basket remains on the base plate of the STC, while the fuel moves toward the lid. In this situation, a misalignment of about 6 inches could exist, considering all tolerances in

such a way that they would increase the misalignment. Conservatively, a maximum misalignment of 8 inches is used in the calculations.

- Only the center 4 assemblies can slide fully towards the lid, since the shield-ring that is attached to the inside of the lid essentially prevents any significant sliding of the outer 8 assemblies. Nevertheless, the analyses assume that all 12 assemblies in Configuration 1 and all 8 assemblies in Configuration 2 slide fully towards the lid. This is a substantial conservatism, and was predominantly chosen to simplify the modeling of this condition.

- As in the design basis calculations, any steel of the basket above or below the poisoned area is neglected and replaced by water. This means that over the entire height of the misalignment, only fuel and water is present, while in reality there would still be some basket steel present between the assemblies in that area.

- As pointed out in the RAI, for Configuration 1, it is important to consider the lower burnup of the fuel in the misaligned area compared to the assembly average burnup. To ensure the maximum reactivity effect of the misalignment is considered, calculations are performed for both Configuration 1A and 1B, and several enrichments including the maximum and minimum enrichments analyzed for the normal conditions.

- The tipover accident of the STC is independent from the misloading accident condition, which was the only accident condition previously analyzed for its impact on criticality safety. Consequently, it is not necessary to consider both the misloading and the misalignment accidents concurrently. The approach in the analysis is therefore to use the soluble boron requirement from the misloading accident, and then show that the maximum k-eff from the misalignment is below that from the misloading conditions, i.e. to show that the misalignment condition is bounded by the previously analyzed accident condition.

The results (maximum k-eff) of the analyses performed under the conditions outlined above are summarized in the following table. In all cases, the maximum k-eff is below that for the previously analyzed bounding accident condition of 0.9450. The misalignment from a tip-over event is therefore bounded by the previously analyzed conditions. Therefore, no increased heat generation is expected during the tip-over, and the previously determined soluble boron level of 1025 ppm for accident conditions remains unchanged.

Enrichment, wt%	Maximum k-eff under Tip-Over Conditions, with 1025 ppm Soluble Boron		
	Configuration 2	Configuration 1A	Configuration 1B
2.0	N/A	0.8271	0.8281
4.0	N/A	0.9233	0.9173
5.0 (Configuration 2 and 1A) 4.5 (Configuration 1B)	0.8441	0.9340	0.9231

Chapter 4 of the licensing report will be revised to include the results of this analysis.

CHAPTER 7 -SHIELDING DESIGN AND ALARA CONSIDERATIONS

NRC RAI 7-1

Provide a dose rate evaluation for neutron source assemblies (NSAs) with the appropriate cobalt impurity level, modifying other aspects of the shielding evaluation as necessary. (CSDAB)

In response to RAI 7-9 in a letter dated July 28, 2011, the licensee provided cobalt impurity levels for the assembly hardware and non-fuel hardware that are based on hardware age and vendor data. The impurity level for NSAs doesn't appear to be consistent with the age and vendor data. Given the cooling time and the allowable burnup of the NSAs, this hardware would appear to have been manufactured at a time of higher impurity levels. Thus, the shielding evaluation should address NSAs with the higher impurity level or further justification should be provided to support using the lower impurity level for NSAs.

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 7-1

The NSA related analyses have been revised utilizing the maximum cobalt impurity level (1.2 gm/kg) from the pre 1989 era. The pre 1989 impurity level is obtained from the measurements of the composition of the assembly hardware used by the Indian Point unit 3 assembly manufacturer (Westinghouse). Note that the minimum cooling time requirement for the NSA is also revised to 20 years of cooling, while the maximum achievable burnup (360,000 MWD/MTU) remains unchanged. Section 7.2.2 of the licensing report will be revised to reflect the updated cobalt impurity level for NSA. Additionally, Section 7.4.11 of the licensing report and Appendix C of HI-2084109 (shielding calculation package) will also be revised. The proposed TS (Table 3.1.2-2) has been revised to incorporate the increased NSA minimum cooling time requirement. Section 7.4.11 of the licensing report will be revised as follows:

7.4.11 <u>Dose Rates from STC and HI-TRAC with NSA</u>

The dose rates comparison between one NSA (inner basket location) and 11 BPRAs vs. 12 BPRAs on the surface, and 1 m from the STC and HI-TRAC are reported in Table 7.4.21. One NSA (inner basket location), 3 RCCAs (inner basket locations) and 8 BPRAs (outer basket location) are compared with 4 RCCAs (inner basket location) and 8 BPRAs (outer basket location) for bottom dose rates. Note that loading pattern 4 from Table 7.1.1 is used for this comparison purpose. 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 do not shield the inner basket cells. Table 7.4.21 demonstrates in general that NSA is bounded by BPRAs for the side, and for the bottom 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 the 12 BPRAs case. Further investigation has shown that the difference for 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. The HI-TRAC dose rate with NSA at 45^0 (on surface) is 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 and therefore

higher than would otherwise be the case. Therefore, BPRAs and RCCAs are used to report the maximum doses and to show compliance with the regulatory limits.

TABLE 7.4.21

DOSE RATES AROUND STC AND HI-TRAC WITH NSA AND BPRA

	STC		**HI-TRAC**	
Dose Rate Location	**Total with 1 NSA and 11 BPRAs (mrem/hr)**	**Total with 12 BPRAs (mrem/hr)**	**Total with 1 NSA and 11 BPRAs (mrem/hr)**	**Total with 12 BPRAs (mrem/hr)**
	Radial Surface			
Surface (averaged)	3248.8	3256.9	1.4	1.4
Surface 0°	11450.7	11373.9	1.1	1.1
Surface 45°	1816.5	1891.2	0.9	0.8
1 m away from surface (averaged)	1034.9	1036.8	0.7	0.7
1 m away from surface 0°	1405.2	1385.0	0.7	0.7
1 m away from surface 45°	786.2	869.5	0.7	0.7
	Top Lid			
Surface	1324.1	1360.3	1880.2	1887.2
1 m away from surface	383.8	389.7	701.3	703.9
	Bottom Lid			
	Total with 1 NSA, 3 RCCAs and 8 BPRAs (mrem/hr)	**Total with 4 RCCAs and 8 BPRAs (mrem/hr)**	**Total with 1 NSA, 3 RCCAs and 8 BPRAs (mrem/hr)**	**Total with 4 RCCAs and 8 BPRAs (mrem/hr)**
Surface	18865.0	21188.0	12.6	12.7
1 m away from surface	4387.2	4704.7	16.4	16.9

NRC RAI 7-2

Modify the shielding accident evaluations to address dose at distance from the HI-TRAC/STC system in a tipover that includes any uncovering of fuel in the STC. (CSDAB)

In response to RAI 7-13 in a letter dated July 28, 2011, the licensee modified the accident dose analysis to include aspects of the tipover scenario. However, the response does not appear to have addressed all the concerns expressed in that RAI. Given the void space in the STC, there may be a possibility for some of the fuel to become uncovered with the STC on its side. This condition could have significant dose consequences and should be evaluated. Any assumptions used in the analysis should be adequately justified.

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

Response to RAI 7-2

The accident Condition 3 in the licensing report, which previously was STC off-center within the HI-TRAC due to tip over resulting in crushing of the centering assembly accompanying with the simultaneous loss of water from HI-TRAC jacket and HI-TRAC annulus, has been revised to address the possibility that some of the fuel would be uncovered with the STC on its side. The revised accident Condition 3 now very conservatively assumes the non-credible condition where there is absolutely no water in the HI-TRAC and STC system, that is, all the fuel assemblies inside the STC are exposed (uncovered). Table 7.4.20 of the licensing report, which presents the dose at the controlled area boundary for accident Condition 3, will be revised as presented below:

TABLE 7.4.20

DOSE (WITH BPRA) TO AN INDIVIDUAL AT CONTROLLED AREA BOUNDARY FOR ACCIDENT CONDITION 3

HI-TRAC Dose Rate (mrem/500 hrs)	**Dose rate from ISFSI (approx. 137 m from the edge of the ISFSI in the west direction) (mrem/500 hrs)[1]**	**Site (e.g. operating plant facilities and other site sources such as the temporary low level storage building) (mrem/500 hrs)[2]**	**Total (mrem/ 500 hrs)**	**Regulatory Limit (mrem)**
10CFR72.106(b) – Accident (1 cask)				
43.0	13.4	0.43	56.83	5000

1. The closest controlled area boundary location (with bounding dose rates from the ISFSI) is on the west side towards the Hudson River. Since there are no permanent occupants in the west direction (due to the Hudson River) 500 hours per year is used as the occupancy time.
2. 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installation Utilizing the Holtec International HI-STORM 100 Cask System, Rev 7, Entergy Nuclear

NRC RAI 7-3

Provide the neutron energy spectrum on the outer surface of the STC and the transfer cask at the measurement points specified in Technical Specification (TS) 5.4.6 that were used to determine the neutron dose rates at those points. (CSDAB)

TS 5.4.2 establishes requirements that the neutron plus gamma measured dose rates not to exceed 1400 mrem/hr on the top of the STC (with the lid in place) and 5 mrem/hr on the side of the transfer cask. In order to measure the neutron dose rate, the neutron energy spectrum must be approximated to ensure the detection equipment is appropriately calibrated for the neutron spectrum at the measurement points. Therefore, please provide the MCNP-based neutron energy spectrum at the measurement points specified in TS 5.4.6.

This information is needed to confirm compliance with 10 CFR 50.34, 10 CFR 20.1301(a) and (b), 20.1501(b), and the intent of 10 CFR 72.104.

Response to RAI 7-3

Neutron energy spectrums for the dose measurement locations as specified in proposed TS 5.4.6a have been evaluated (see response to RAI TS-1) for the STC lid (Location 1) and the HI-TRAC side. Also evaluated were the neutron dose rates at locations approximately 180 degrees apart on the periphery of the STC lid avoiding the areas around the inlet and outlet ports (Locations 2 and 3 in the Tables below). The results are discussed in Appendix F of HI-2084109 (shielding calculation package). The neutron spectrums for these measurement locations are presented in the following tables. Loading pattern 3 from Table 7.1.1 of the licensing chapter was used for this evaluation. Note that the other TS 5.4.6a STC lid locations were not evaluated. These locations will be evaluated and the results will be provided to IPEC Radiation Protection to confirm that the neutron detection equipment is appropriately calibrated.

The results are summarized in the following tables.

Location	Energy Range						
	0.0 – 0.5 eV	0.5 – 1.0 eV	1.0 eV – 1.0 keV	1.0 keV – 0.1 MeV	0.1 – 1.0 MeV	1.0 – 20.0 MeV	Total
	Neutron Flux (neutrons/cm^2-sec) without NSA contribution						
STC lid (location 1)	0.9	0.4	5.3	5.6	13.6	3.7	29.4
STC lid (location 2)	0.6	0.3	6.0	6.7	8.4	2.3	24.3
STC lid (location 3)	0.8	0.1	2.9	5.7	11.0	2.6	23.1
HI-TRAC side	1.92	0.03	0.34	0.39	0.65	1.56	4.89

Location	Energy Range						
	0.0 – 0.5 eV	0.5 – 1.0 eV	1.0 eV – 1.0 keV	1.0 keV – 0.1 MeV	0.1 – 1.0 MeV	1.0 – 20.0 MeV	Total
	Neutron Flux (neutrons/cm^2-sec) with NSA contribution						
STC lid (location 1)	0.9	0.4	5.4	5.7	14.0	3.8	30.2
STC lid (location 2)	0.6	0.3	6.0	7.0	8.5	2.7	25.1
STC lid (location 3)	0.8	0.1	3.0	5.9	11.0	2.7	23.5
HI-TRAC side	1.93	0.03	0.34	0.40	0.66	1.57	4.93

CHAPTER 8 - MATERIALS EVALUATION, ACCEPTANCE TESTS and MAINTENANCE PROGRAM

Refer to RAI 8-4 from the licensee's letter dated July 28, 2011. Provide justification to reconcile the following discrepancies in applicable rules for construction between the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Division 1, Subsection ND and Subsection NC for the construction of the STC to ensure that the STC is constructed to acceptable quality standards:

NRC RAI 8-1

The lateral expansion test results for the STC bolting material met the requirements for ASME Code Subsection ND, however, it did not meet the requirements stated in Subsection NC. Substantiate the justification provided previously by providing a numeric estimate of the lateral expansion value at the lowest service temperature. Include the basis and calculation for both the lowest service temperature and the estimated lateral expansion value. Alternatively, use bolting material meeting the acceptance requirements of ASME Code Subsection NC. (SMMB)

Response to RAI 8-1

In order to ensure that the bolting material meets the requirements for both ASME Code Subsections NC and ND, the bolting material has been changed to SB-637 N07718. Since SB-637 N07718 is a non ferrous material, no impact testing is required per Paragraphs NC-2311(a) (7) and ND-2311(a) (7). The licensing drawing (6013), Licensing Report (HI-2094289) and the Structural Calculation package (HI-2084118) will be revised to capture the change in the bolting material. Entergy commits that the existing STC lid bolts will be replaced by new SB-637-N07718 lid bolts prior to the first inter-unit transfer of fuel.

NRC RAI 8-2

ASME Code Subsection NC requires full radiography of the weld joints. However, select radiography is required by ASME Code Subsection ND. Substantiate the justification provided previously for the limited radiographic testing (RT). Provide evidence that the portions of the confinement boundary welds that were not subject to RT will perform as expected. Include a discussion of weld repairs that resulted from RT over adjacent and similar joint configurations. Alternatively, perform full radiography of the weld joints as required by ASME Code Subsection NC. (SMMB)

Spent fuel canisters are normally constructed to ASME Code Subsection NB or NC. The licensee is proposing to construct the STC to ASME Code Subsection ND and has provided a table outlining the discrepancies between Subsection NC and ND. However, discrepancies regarding both brittle material behavior and weld quality were not adequately addressed.

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

Response to RAI 8-2

In the response to RAI 8-4 from the July 28, 2011 Entergy letter (ADAMS Accession No. 112200258) and to its predecessor RAI 8-8 dated October 5, 2010 (ADAMS Accession No.

ML102910511), the safety case for invoking ASME Section III Class 3 for the STC was presented. In particular, in the above RAI responses, the requirements of Subsection ND with respect to the criteria that quantify the margin of safety in the STC against loss of confinement integrity were set down and shown to be met by large margins. In the supplemental response provided below, the residual deficiency in the previously provided safety justification with respect to "both brittle material behavior and weld quality", noted in this RAI, is addressed.

First, it should be emphasized that the confinement welds in the STC meet Section III class 2 (NC) in all respects including welding procedure specification, procedure qualification and welder training protocols. In other words, the welds were deposited in the STC vessel in full compliance with "NC" in every respect. The only respect in which the as-deposited welds don't meet NC verbatim is the extent of radiography performed. Even in respect of radiography, the amount of NDE performed on the STC welds is 35 times that required for "ND" and nearly 80% of that required for NC.

From the stand point of safety, this slight deficit in the extent of radiography coverage and the associated reduction in the safety margin is more than compensated by the fact that the computed margin of safety in the STC confinement shell is in excess of 9. This large margin of safety (greater than 9) is computed by making an additional overarching assumption that does not credit the lead and steel that stiffen the confinement shell. Such a large margin of safety can be intuitively accepted to eliminate the risk of a flaw propagation in the (theoretically speaking) incompletely radiographed regions of the confinement boundary, namely the shell-to-base plate and shell-to-flange butt welded joints (i.e., those which have not been 100% radiographed).

To provide a quantitative proof of the absence of risk of crack propagation in the confinement boundary, the methodology (linear fracture mechanics) used in docket number 72-1014 to establish a similar margin in the MPC lid-to-shell weld (which is not radiographed or UT'd) is used.

For purposes of defining the margin of safety against crack propagation, we make the assumption that the reference temperature is -40 deg. F, an exceedingly conservative assumption because the operating conditions limit the environmental temperature to 0 deg. F in Chapter 3 of the Licensing Report. For reference, the STC confinement boundary welds, which are depicted in Figure 8-2.1, include the STC shell-to-base plate weld, the STC shell-to-upper flange weld, and the longitudinal and circumferential welds within the STC shell. Figure 8-2.1 also provides the weld identification numbers for each of the welds. The RT examinations performed for these welds are summarized in Table 8-2.1.

From Table 8-2.1, four of the five STC confinement boundary welds have been subjected to 100% radiography as ASME Code Subsection NC would require. Overall 79% of the total weld length associated with the STC confinement boundary welds have been examined by RT. The only portions of the STC confinement boundary welds that have not been fully radiographed with acceptable results are the shell-to-upper flange weld, a 1" segment of the shell-to-base plate weld that has a known indication and was not repaired, and a 1" segment of the shell-to-base plate weld that was repaired after RT, but was not re-examined. A non-conformance report was generated for the two 1" segments on the shell-to-base plate weld, and the discrepant conditions were evaluated against ASME Code Subsection ND requirements and accepted as-is (Ref. Holtec SMDR # 1775-2008).

In summary, the STC confinement welds have been subject to an extensive amount of RT examination, which far exceeds the minimum requirements of ASME Code Subsection ND.

In fact, the longitudinal and circumferential welds within the STC shell have been fully radiographed such that they meet the inspection requirements of ASME Code Subsections NC and ND. It is only the shell-to-upper flange weld and limited portions of the shell-to-base plate weld that have not been subject to 100% RT per ASME Code Subsection NC.

Because the shell-to-upper flange and shell-to-base plate welds are full penetration butt welds it is readily deduced that the most likely fracture failure mode for these welds is through tensile failure, formally known as the Mode I crack propagation in the fracture mechanics literature. We should, therefore, assume a flaw shape and configuration which would synergize with the Mode I failure model. The shape of the weld joint and the nature of the applied loading (axial) indicate that the flaw should be assumed to be rectangular with sharp corners, oriented with its long side perpendicular to the STC shell axis (see Figure 8-2.2). Further, we assume that the flaw extends 360 degrees circumferentially, i.e., it is seamless. Such an adversely oriented cylindrical flaw, albeit entirely hypothetical, helps maximize the potential for crack growth. Finally, we will assume that the flaw is 50% of the thickness of the shell, i.e., 2a = 0.5" in Figure 8-2.2.

Under normal conditions, the only sources of stress in the weld are the dead weight of the STC (excluding the base plate) and the operating stress from internal pressure. The state of stress is more severe during lifting (when the STC is lifted using a connection to the top lid) when the entire dead weight of the STC shell, base plate, SNF, and fuel basket must be supported by the shell welds. The combined effects of normal handling plus accident internal pressure are examined in Appendix B of [1], which gives σ = 2,054 psi for the 1" thick shell weld. We will now proceed to utilize concepts from fracture mechanics to determine the consequences of σ on the assumed flaw.

A critical flaw size is assumed to exist if the Stress Intensity Factor (SIF) equals the materials' fracture toughness.

Fracture toughness of carbon steel (SA-516 Gr. 70) is obtained from the Certified Material Test Report (CMTR) for the STC confinement shell. The lowest measured value of Charpy absorbed energy, C, at -40°F was 28 ft-lb.

The Charpy value C is related to the fracture toughness K by a relationship of the form [2].

$$\left(\frac{K}{\sigma_y}\right)^2 = 5\left(\frac{C}{\sigma_y} - 0.05\right)$$

where:

K is fracture toughness in ksi $\sqrt{inch}$

σ_y is yield stress, ksi

C is Charpy energy in ft-lb

Using a conservative value of C = 28, σ_y = 38 (yield strength), we obtain

$$K = 70.4\ ksi\ \sqrt{inch}$$

To determine the "Stress Intensity Factor", we utilize the Mode I solution for a center cracked plate of width 2b [3]. The value of b in our case is the half-thickness of the STC confinement shell, i.e., 0.5 inch.

Let us assume that the crack is 0.5 inch long, i.e., a = 0.25" in Figure 8-2.2. The stress intensity factor K_I under σ in this configuration is given by

$$K_I = \sigma\sqrt{\pi a}F(x)$$

where:

$$F(x) = \{1 - 0.025x^2 + 0.06x^4\}\sqrt{\sec\frac{\pi x}{2}}$$

$$x = \frac{a}{b} = 0.5$$

σ = 2.05 ksi

By substituting for x, we obtain; F(x) = 1.186. Then, for σ = 2.05 ksi, we have K_I = 2.15 << K.

Thus, the fracture toughness of carbon steel is considerably greater than the stress intensity factor corresponding to a 50% thru-thickness crack (360°) oriented to maximize the potential for Mode I (tensile) failure at the (unattainably) low temperature of -40 deg F.
This result provides the safety justification that there is no credible risk that an undetected flaw in either the shell-to-upper flange weld or shell-to-base plate weld would propagate under the most severe STC loading condition.

References:

[1] HI-2084118, Shielded Transfer Canister Structural Calculation Package, Revision 4.

[2] "Stress, Strain, and Structural Matrices", W.D. Pilkey, Wiley.

[3] "The Stress Analysis of Cracks Handbook", H. Tada et. al., 3rd Edition.

Table 8-2.1: Summary of RT Examinations for STC Confinement Boundary Welds

Weld ID No. (see Figure 8-2.1)	Total Length of Weld (inches)	Length of Weld Subjected to RT (inches)	Did Weld Repairs Result from RT?	Remarks
1.1	81	81	Yes	100% radiographed; initial RT results led to one 4-inch long weld repair; RT re-performed following repair with acceptable results
1.2	81	81	Yes	100% radiographed; initial RT results led to 2 weld repairs totaling 6-1/4" in length; RT re-performed following repairs with acceptable results
2	138	138	Yes	100% radiographed; initial RT results led to 6 weld repairs totaling 108" in length; RT re-performed following repairs with acceptable results
11	138	17	No	spot radiography performed; RT results were acceptable
13	138	138	Yes	100% radiographed; initial RT results led to 12 weld repairs totaling 44-5/8" in length; RT re-performed following repairs with acceptable results; IPEC later identified two issues with RT results/weld repairs (a 1" long indication was identified on one RT film that was initially accepted without repair, RT was not re-performed for one weld repair)

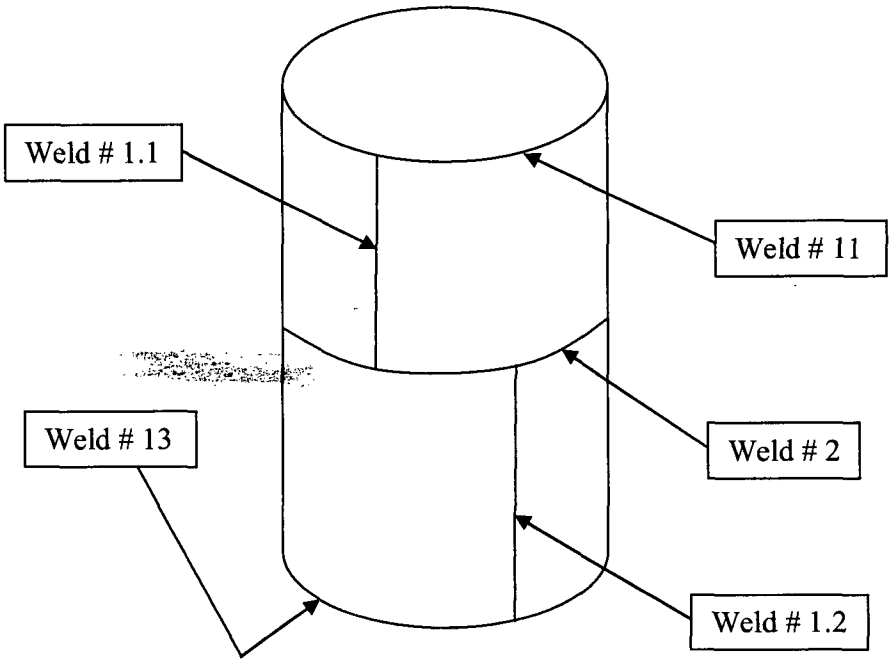

Figure 8-2.1: STC Confinement Boundary Welds

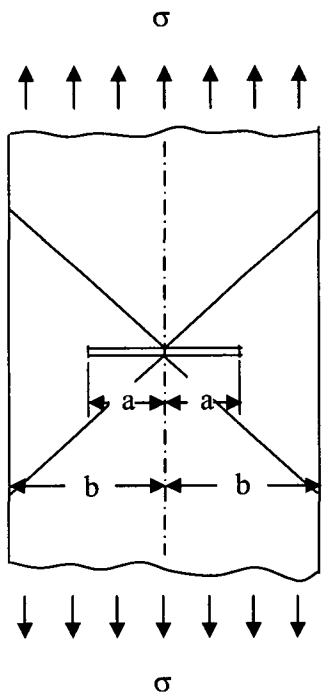

Figure 8-2.2: Center Crack in STC Shell Weld

NRC RAI 8-3

Modify the acceptance tests descriptions to address the following items: (CSDAB)

a. State in the first paragraph in SAR Section 8.4.1 that visual inspections and measurements will ensure the STC dimensions conform to the TS.
b. Define the term "packaging" used in SAR Section 8.4.1.
c. Define "significant" as used for the lead shielding acceptance testing in SAR Section 8.4.5.

RAI 8-3 in the NRC letter dated March 16, 2011, requested a change to Section 8.4.1 of the SAR to indicate that the STC dimensions would be inspected to ensure compliance with the TS. This aspect of that RAI was not completely addressed. The same section of the SAR also uses the term "packaging." This term is not defined anywhere in the SAR and it is unclear as to what constitutes the "packaging" that will be inspected. The STC lead shielding is found acceptable, per Section 8.4.5 of the SAR, if the gamma scan measurements don't vary significantly from the average gamma dose rate measurements, accounting for the presence of the STC ribs. The applicant should describe what constitutes a significant variation, especially in light of the dose rates at and near the STC ribs being rapidly varying from 11.4 to 4.4 rem/hr on contact while the average dose rate is about 3.3 rem/hr on contact. If comparison is made to some standard, such as a test block, then the acceptance criterion should be clarified to indicate this comparison and what constitutes acceptable shielding performance.

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

The acceptance test descriptions in the SAR will be revised as follows:

a) A sentence will be added to section 8.4.1 to state that visual inspections and measurements will ensure the STC dimensions conform to the Technical Specifications.
b) The word "packaging" will be replaced by the term "STC" in section 8.4.1.
c) The following description of acceptance testing will be added to section 8.4.5, "The shielding in the sides of the STC is composed of a steel-lead-steel composite in all areas except where the ribs are located. The purpose of the gamma scan is to ensure that there are not areas where there are significant gaps in the lead shielding. The absence of even a small percentage of the lead thickness would create a significant increase in the gamma radiation that travels through the wall of the STC. Therefore a shield block constructed of a steel-lead-steel composite using the minimum thickness of each of the layers has been used to define the minimum acceptable shielding levels. In the areas near the ribs, the dose rate will be higher due to the streaming through the rib. In these areas, the gamma source is first used to measure the gamma transmission through the rib itself. The transmission level through the rib is then used as the limit for the acceptable gamma count in the areas adjacent to the ribs. Any significant gaps in the lead, which would be filled with air, will lead to an increase in dose rate that would exceed that which passes through the steel rib itself and would be considered unacceptable. All dose rates measurements adjacent to the rib area have been confirmed to be less than the dose rate measurements through the rib itself and all other dose rate measurements have been demonstrated to be less than that of the test block confirming that the lead installation in the STC is free of voids."

CHAPTER 10 -OPERATING PROCEDURES

In a letter dated July 28, 2011, Entergy responded to NRC RAI TS-8 by stating that Surveillance Requirement (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 states, "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."

In a clarifying response, Entergy described that the STC lid is fitted with process connections which included valve quick-disconnect nipples (Snap-Tite model SHVN-12-12) and that, prior to installing the nuts that hold the STC lid in place, hoses with the mating valved quick-connect couplings are attached to the lid which ties them into the relief valves, pressure gauges, and further downstream, isolation valves used for the test.

NRC RAI 10-1

The proposed arrangement has an isolation valve (quick disconnect) in the pressure relief line flow path. This configuration is inconsistent with the ASME Code. Describe how direct communication will be verified between the STC and the pressure relief valves and pressure instrumentation. (SBPB)

This information is needed to confirm compliance with GDC 61, GDC 63, and 10 CFR Part 50, Appendix B.

Response to RAI 10-1

Steps will be added to the Operations chapter (Chapter 10, Section 10.2.3) to direct that borated water (minimum 2,000 ppm) be circulated through the STC prior to bolting the lid down. The water will be supplied through the vent connection and will exit at the drain connection (See Figure 10.1 of the licensing report). The water flow through the STC will ensure that there is an open pathway between the STC internals and the relief valve (i.e., the quick connect fitting valves are open). After the flow check has demonstrated that the flow path is open, the STC lid will be bolted down to create a closed system for establishing the steam vapor space.

NRC RAI 10-2

The pressure rise test is monitoring for a very small increase in pressure. However, any small leakage at fitting and pipe connections would adversely influence the test results. Describe how leakage will be prevented at the various valves and fittings to assure the pressure rise test results will not be invalidated. (SBPB)

This information is needed to confirm compliance with GDC 61, GDC 63, and 10 CFR Part 50, Appendix B.

Response to RAI 10-2

The piping and connections used for the pressure rise test will be leak tested as part of the equipment check-out, prior to each fuel transfer operation. A test procedure will be

developed to demonstrate that any leakage from the instrument tree valves and fittings would not have a significant impact on the pressure rise test results. The amount of air in-leakage that would be equivalent to the 0.2 psi/hr (proposed TS LCO 3.1.4) pressure rise in the STC vapor space has been determined. An acceptance criterion of 10% of that leakage rate has been selected as a reasonable limit for any air infiltration from all of the valves, relief valves, instruments, and piping joints for the vacuum pressure rise test. A factory leak test of the system utilizing this methodology has been performed successfully. In the event that connections are broken due to repair or replacement of components during the loading campaign, the piping and connections will also be re-tested. When the steam generator system is used to establish the steam vapor space in the STC, the process connections will be monitored for signs of leakage at the quick connect fittings and any leaks will be repaired prior to performing the pressure rise test.

NRC RAI 10-3

Clarify the language in Chapter 10 of the SAR with regard to filling of the HI-TRAC neutron shield jacket and performance of the dose rate measurements as described below. (CSDAB)

The condition given in Section 10.1.3, Step 5 for filling the HI-TRAC neutron shield jacket is not clear. It should be modified to indicate that the shield jacket is filled, if not already filled. Additionally, the language regarding the dose rate measurements given in Section 10.2.3, Steps 18, 19, 23, and 24 should be clarified to indicate that the measurements are compared against dose rate limits and not expected dose rates. Also, the second aspect of the written evaluation described in Steps 19.c and 24.c is not clear. It seems this part should be written as: "... (2) if the higher dose rates are acceptable. If the higher dose rates are acceptable, fuel transfer can continue ..."

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 10-3

Chapter 10 of the SAR will be revised as follows:

- Section 10.1.3, Step 5 will be revised to state that the neutron shield jacket is filled if it was not already filled.
- Section 10.2.3, Steps 18 and 19 will be revised to state that the measured dose rate is compared against a calculated dose rate based on the STC being full of water to provide preliminary assurance that the STC contents will meet the Tech Spec Limits.
- Section 10.2.3, Steps 23 and 24 will be revised to state that the dose rate measurements are compared against the dose rate limits of the Technical Specifications.
- Section 10.2.3, new steps 41 and 42 will be added to perform dose rate measurements at the STC lid. The dose rate measurements are compared against the dose rate limits of the Technical Specifications. If the dose rate measurements exceed the limits the requirements of Technical Specification Appendix C, Part II, Subsection 5.4 will be followed.
- Section 10.2.3, Steps 19c and 24c will be revised to state "If the higher dose rates are acceptable per the required evaluations, fuel transfer can continue..."

TECHNICAL SPECIFICATIONS

NRC RAI TS-1

Modify proposed TS Appendix C, Part II, 5.4.2 and 5.4.6 to provide dose rate limits and associated measurements that are appropriate for the allowable loading configurations and are supported by the evaluations in the licensing report (SAR). (CSDAB)

In a letter dated July 28, 2011, in response to RAI TS-4 the applicant proposed dose rate limits and associated measurements for the transfer operations. The dose rate limits are for the STC lid and the side of the HI-TRAC. However, the proposed limit and measurements for the STC lid do not appear to be adequate for the operations. The limit for the STC lid and its measurement were derived from a configuration with the STC fully loaded and the air gap present in the STC. The description in the operations section of the SAR has the measurement performed with the STC filled with borated water. Also, given that the inner basket cells may be kept empty to comply with criticality requirements, it is not clear that the single measurement location at the STC lid radial center is sufficient. Additional measurement locations should be specified that include areas over the outer basket cells.

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

Response to RAI TS-1

The measurements taken in accordance with Section 5.4 of Technical Specification (TS) Appendix C are in place to ensure that the calculated contribution of the inter-unit transfer to the site boundary dose is below what is assumed in the site boundary dose report required for an Independent Spent Fuel Storage Installation (ISFSI) per 10CFR 72.104. The dose rate limits established for the top of the STC lid, measured per 5.4.6.a, and the side surface of the HI-TRAC, measured per 5.4.6.b, are based on a bounding configuration of fuel, including non-fuel hardware as allowed per the TS. This will bound any allowed configuration of assemblies and non-fuel hardware in the STC including the configuration where only eight cells are loaded. This is because the outer region loading requirements are more restrictive, requiring fuel with longer cooling times and disallowing certain non-fuel hardware. Therefore, TS 5.4.2 is considered sufficient to ensure the intent of 10CFR 72.104 is met.

In response to the RAI request for additional measurement locations on the STC lid Entergy proposes to take two additional verification measurements approximately 180 degrees apart and 12 to 18 inches from the center of the lid, avoiding the areas around the inlet and outlet ports, to ensure that the lid surface dose rate in this location also does not exceed the proposed TS limit. All measurements required by the TS shall be taken when the STC is in the HI-TRAC after the steam space is established and prior to HI-TRAC lid installation.

Proposed TS Section 5.4.6.has been revised to read:

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

As illustrated below, the dose rate measurements will be taken at locations 1, 2, and 3.

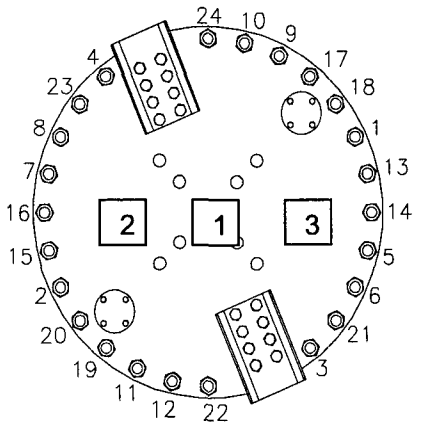

Figure TS-1.1

NRC RAI TS-2

Modify the following items in TS Appendix C, Part I (CSDAB):

a. Change the minimum outer STC shell thickness to 3/4 inches in Section 1.0.
b. Ensure the minimum dimensions given for the HI-TRAC are consistent between the TS and the licensing drawing.
c. Modify 2.3.m to read "Manual crane operations for bare STC movements."

The shielding evaluations for the bare STC used 3/4 inches as the outer shell thickness. Dose rates for the minimum thickness would be significantly higher and impact the evaluations in the Licensing Report. Since the STC has already been built and the as-built shell dimension is at 3/4 inches no design change would be necessary for the actual STC. Additionally, consideration should be given to ensuring consistency between the licensing drawing and the TS. The HI-TRAC dimensions should also be consistent between the TS and the licensing drawing. It is not clear that this is the case. Changing 2.3.m to the suggested text captures more generically the off-normal operations, which would include a crane hang-up and any other operational event that requires personnel to be in close proximity to the STC when not in the HI-TRAC.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(a) and (c), 72.104, 72.106, and 72.126.

Response to RAI TS-2

TS Appendix C, Part I has been modified as follows:

a. The minimum outer STC shell thickness has been changed to ¾ inches in Section 1.0.

   The licensing drawing (6013) provides the STC general arrangement that was used to fabricate the STC and is therefore not subject to revision. The dose analyses assumed, in some cases, more restrictive dimensions than shown on drawing 6013.

These more restrictive dimensions are included in TS Appendix C, Part I to provide assurance that should a replacement STC be required it would be fabricated to dimensions that preserve the dose analysis assumptions.

b. The minimum "water jacket" thickness has been changed to 5".

c. TS 2.3.m has been revised to read "Manual crane operations for bare STC movements including demonstration of recovery from a crane hang-up with the STC suspended from the crane."

NRC RAI TS-3

Modify the following aspects of TS Appendix C, Part II, LCO 3.1.2 and Table 4.1.1-1 (CSDAB):

a. Include "and/or NON-FUEL HARDWARE" at the end of the NOTE and in SR 3.1.2.2 of the LCO (SR 3.1.2.2: "... prevents inserting fuel assemblies and/or NON-FUEL HARDWARE into cells 1, 2, 3, and 4 of the STC is installed.")
b. Change Table 3.1.2-2 to show the unanalyzed cooling times for the TPDs and NSAs as N/A, not allowed. Define N/A at the bottom of the Table. Capitalize non-fuel hardware and spell out ITTR in note (a).
c. Confirm the Hafnium Suppressors burnup and cooling time specifications are correct in Table 3.1.2-2.
d. Change note b to Table 3.1.2-2 to indicate that interpolation is not acceptable/applicable for TPDs for burnups greater than 90 GWd/MTU and cooling times greater than 15 years (similar to what was done for HI-STORM 100) as well as for NSAs, RCCAs and Hafnium suppressors.
e. Change LCO 3.1.2.a.1 and b.1 to state initial average enrichment, or change footnote (a) to Table 3.1.2-3 to indicate that initial enrichment is assembly average enrichment and the specification is a limit on the minimum average initial enrichment.
f. In Table 3.1.2-1, the last entry in the column for Configuration B is a dash. The action to take is not clear. Recommend replacing the dash with note (e), "Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel."
g. Clarify Table 4.1.1-1 to indicate that the guide tube material is also ZR.
h. Footnote (b) to Table 4.1.1-1 should be deleted as it is not necessary.

These changes are necessary to keep the proposed TS consistent with the evaluations performed in the SAR and to clarify the conditions and limits on allowable contents. With regard to the Hafnium Suppressors, it is not clear, based on staff's understanding of these suppressors and their specifications given in the HI-STORM 100 Certificate of Compliance TS, that the specifications listed in the proposed TS are appropriate.

This information is needed to confirm compliance with 10 CFR Part 50 and the intent of 10 CFR 72.44(c), 72.104, 72.106, and 72.126.

Response to RAI TS-3

TS Appendix C, Part II, LCO 3.1.2 and Table 4.1.1-1 have been modified as follows:

a. LCO 3.1.2 has been revised to add "and/or NON-FUEL HARDWARE" at the end of the NOTE and SR 3.1.2.2 has been revised to read "... prevents inserting fuel assemblies and/or NON-FUEL HARDWARE into cells 1, 2, 3, and 4 of the STC is installed."

b. Table 3.1.2-2 has been revised to show the unanalyzed cooling times for the TPDs and NSAs as N/A, and N/A has been defined in new Note (c) as "N/A means not authorized for loading at this cooling time." In Note (a) Non-fuel hardware has been capitalized and ITTR has been spelt out as Instrument Tube Tie Rods.
c. The hafnium suppressors at IP3 do not have cumulative burnups that exceed 30,000 MWD/MTU therefore Entergy is proposing to add another column to TS Table 3.1.2-2 specifically for the hafnium suppressors and remove them from the RCCA column where the allowable burnup for these devices is up to 630,000 MWD/MTU. Hafnium suppressors are limited to loading in the four inner cells and the burnup is limited to 30,000 MWD/MTU with cooling time greater than 8 years. The dose at the bottom of the STC from the RCCAs bounds the hafnium suppressors due to the much higher allowable burnup of the RCCAs and the resulting higher gamma source from the activated cladding on the RCCAs. The dose at the top of the STC from BPRAs bounds the hafnium suppressors due to their equivalent physical characteristics and allowable burnup and cooling time combinations. This discussion will be incorporated into the revised licensing report.
d. Note (b) to Table 3.1.2-2 has been revised to indicate that interpolation is only permitted for BPRAs, WABAs and TPDs and that interpolation is not permitted for TPDs for burnups greater than 90 GWd/MTU and cooling times greater than 15 years.
e. Footnote (a) to Table 3.1.2-3 has been revised to indicate that initial enrichment is assembly average enrichment.
f. Table 3.1.2-1 has been revised by replacing the dash in the last entry in the column for Configuration B by "Note (e)". New Note (e) has been added and reads "Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel."
g. Table 4.1.1-1 has been revised to indicate that the guide tube/instrument tube material is ZR.
h. Footnote (b) to Table 4.1.1-1 has been deleted.

In addition to the above changes Configuration 6 and Note (c) have been added to TS Table 3.1.2-3. This configuration has been added to address the original Unit 3 fuel design (Westinghouse Low Parasitic (LOPAR) fuel) in which the middle five spacers were made of Inconel. LOPAR fuel was phased out by Cycle 7 being replaced by assemblies whose middle five spacers were made of Zircaloy. The evaluations that support adding these changes to Table 3.1.2-3 are included in the Shielded Transfer Canister Shielding Calculation in Enclosure 1 and will be incorporated into the revised licensing report.

NRC RAI TS-4

Remove from the proposed TS Appendix C, Part II the specification regarding the minimum restricted area size for the transfer operations (4.1.4.5 and 4.1.4.12). Instead, provide appropriate discussions in an RAI response and in the SAR regarding the radiological controls that will be exercised during the various stages of the operations. (CSDAB)

Based upon the further evaluations of the transfer operations, including design and operations changes, provided in response to the second round of RAIs and further discussions with the licensee, the NRC staff no longer finds a need for the TS to specify minimum restricted area sizes. Instead, the licensee should describe in the SAR, to an appropriate level of detail, the types of radiological controls to be used for the various stages of the operations to ensure compliance with 10 CFR 20.1301 (b), providing some tie-in (or context) to the evaluation provided in the application. The applicant should also ensure that

text in the application that referred to these technical specifications for minimum restricted area size are modified accordingly.

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

Response to RAI TS-4

The specifications regarding the minimum restricted area size for the transfer operations in proposed TS Appendix C, Part II (4.1.4.5 and 4.1.4.12) have been deleted. Licensing Report sections 7.4.5, 7.4.9, and 7.4.13 will be revised to reflect this deletion.

The TS minimum restricted area sizes were originally proposed to ensure radiation dose limits for individual members of the public would be in compliance with 10 CFR 20.1301. These specifications were based on dose calculations reported in sections 7.4.5 and 7.4.7 of the Licensing Report. These dose calculations remain applicable and continue to demonstrate compliance with 10 CFR 20.1301(b) provided the appropriate access controls measures for members of the public are in place. These measures are described below.

Radiological controls for all stages of inter-unit fuel transfer will be in accordance with the Entergy Corporation and Indian Point Energy Center Radiological Protection Programs. The major stages of the transfer operations are: loading the STC in the Unit 3 SFP, transferring the loaded STC from the Unit 3 SFP to the HI-TRAC, preparing the STC and HI-TRAC for transfer including performing the required TS Surveillances, transferring the loaded STC/HI-TRAC from the Unit 3 FSB to the Unit 2 FSB, preparing the STC and HI-TRAC for STC transfer to the Unit 2 SFP, transferring the STC to the Unit 2 SFP, and unloading the STC in the Unit 2 SFP. During each of these stages access control to radiologically controlled areas will be in accordance with Entergy procedure EN-RP-101. This procedure provides detailed guidance for entry requirements and associated controls for radiologically controlled areas, radiation areas, high radiation areas, contamination areas, etc. In accordance with EN-RP-101 the following access control measures will apply:

a) <u>Loading the STC in the Unit 3 SFP and unloading the STC in the Unit 2 SFP and transfer of a loaded STC from the unit 3 spent fuel pool (SFP) to the HI-TRAC and the subsequent transfer of the STC from the HI-TRAC to the unit 2 SFP including preparation activities.</u>

   Radiation Protection personnel will control access to the FSB to allow only those personnel involved in the work activities access except in the event that a member of the public has a demonstrable need for FSB access. In that case access may be permitted and the limits of 10 CFR 20.1301(b) would continue to apply.

   Radiation Protection personnel will also post and control access to affected areas outside the normal Radiologically Controlled Area (RCA) in accordance with the fleet access control procedure EN-RP-101. This procedure allows for the establishment of a restricted area which is an area to which access is limited for the purpose of protecting individuals (including members of the public) against undue risks from exposure to radiation and radioactive materials. The need for the establishment of a restricted area outside the normal RCA will be determined based on anticipated and actual STC and HI-TRAC dose rate measurements. The extent of any restricted area, if any, would be based on considerations similar to those documented in section 7.4.5 of the Licensing Report.

b) Transfer of the loaded HI-TRAC/STC from unit 3 FSB to unit 2 FSB

Prior to the transfer Radiation Protection Personnel will perform a survey of the loaded HI-TRAC/STC in accordance with TS 5.4.2, TS 5.4.3, and TS 5.4.6. If the TS 5.4.2 dose rate limits are not met then transfer operations shall not occur until appropriate corrective action is taken in accordance with TS 5.4.4.

Radiation Protection Personnel will also post and control access in accordance with Indian Point site procedure for the transfer of radioactive material and the fleet access control procedure EN-RP-101 as in a) above. The extent of any restricted area, if any, would be based on considerations similar to those documented in section 7.4.7 of the Licensing Report. Radiation Protection personnel will provide a continuous escort to ensure all personnel are aware of, and are protected from, any radiological hazard.

These radiological controls will be added to Chapter 7 of the SAR.

The access control measures identified in a) and b) above will ensure compliance with 10 CFR 20.1301(b).

NRC RAI TS-5

TS 3.1.2 refers to Hafnium suppressors while TS 3.1.5 refers to Hafnium inserts. Select a standard term and ensure only that term is used in the TS. (LPL 1-1)

Response to RAI TS-5

The Unit 3 reload designs use the term "Hafnium Flux Suppressors". To provide consistency the term "Hafnium Flux Suppressors" will be used and proposed TS 1.1 Definitions, TS 3.1.2, TS Table 3.1.2-2, and TS 3.1.5 have been revised accordingly.

HOLTEC INTERNATIONAL

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October 28, 2011

Mr. Roger Waters
Licensing Specialist
Indian Point Energy Center
450 Broadway
GSB Second Floor Licensing
Buchanan, NY 10511-0249

Document ID: 1775043

Subject: Information to Support Licensing Submittal on Inter-Unit Fuel Transfer

Dear Mr. Waters:

Holtec is pleased to approve the release of the following information to the United States Nuclear Regulatory Commission (USNRC):

Attachment A - Responses to NRC RAI dated 9/28/2011 (Non- Proprietary)
Attachment B - HI-2084109R9, "Shielded Transfer Canister Shielding Calculation" (Proprietary)
Attachment C- HI-2084118R5, "Shielded Transfer Canister Structural Calculation" (Proprietary)
Attachment D- HI-2084176R6, "Shielded Transfer Canister Criticality Calculation" (Proprietary)

We require that you include this letter along with the attached affidavit pursuant to 10CFR2.390 when submitting Attachments B,C and D to the USNRC.

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The responses to NRC questions have been authored and reviewed by the following individuals:

Response Number	Author	Reviewer
4-1	Stefan Anton	Tao He
Chapter 7	Kaushik Banerjee for Kaushik Banerjee Veena Gubbi	Lysa Sevastyuk for Lysa Sevastyuk
8-1	Chuck Bullard	Veena Gubbi
8-2	Chuck Bullard	John Griffiths for John Griffiths
8-3	John Griffiths for John Griffiths	Veena Gubbi
Chapter 10	John Griffiths for John Griffiths Veena Gubbi	Tammy Morin
Technical Specification	Tammy Morin	Veena Gubbi

Please do not hesitate to contact me at 856-797-0900 x 3703 if you have any questions.

Sincerely,

Veena Gubbi
Adjunct Project Manager
Holtec International