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NL-09-141

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: **Indian Point Nuclear Power Plant Units 2 and 3**
**<u>Non-proprietary response to request for supplemental information regarding the spent fuel transfer license amendment request (TAC Nos. ME1671, ME1672, and L24299)</u>**
Indian Point Units 2 & 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

References:

1) Entergy letter NL-09-100, 09/28/09, "Indian Point Nuclear Power Plant Units 2 and 3 – Response to request for supplemental information regarding the spent fuel transfer license amendment request (TAC Nos. ME1671, ME1672, and L24299)"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc (Entergy) submitted a response to an NRC request for supplemental information regarding a proposed license amendment concerning the inter unit transfer of fuel (Reference 1). A proprietary version of the response was included in that submittal. A non-proprietary version of the response has been prepared and is attached.

In accordance with 10 CFR 50.91, a copy of this submittal, with the attachment is being provided to the designated New York State official.

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

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge. Executed on 10-26-09.

Sincerely,

JE Pollock

JEP/rw

A001
NRR

Attachment: 1. Non-proprietary response to request for supplemental information regarding the spent fuel transfer license amendment request

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

ATTACHMENT 1 TO NL-09-141

# Non-proprietary response to request for supplemental information regarding the spent fuel transfer license amendment request

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

***NRC STAFF ACCEPTANCE REVIEW COMMENTS***
***REGARDING LICENSE AMENDMENT REQUEST FOR 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 No. ML091940176, Entergy Nuclear Operations, Inc. (Entergy) submitted a license amendment request (LAR) for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3). The proposed amendment would license a new shielded transfer canister (STC) and would allow spent fuel to be transferred from the IP3 spent fuel pool to the IP2 spent fuel pool in preparation for further transfer to the Independent Spent Fuel Storage Installation (ISFSI) already located on the site. The NRC staff has reviewed the application for acceptance and concluded that it did not provide technical information in sufficient detail to enable the NRC staff to commence its detailed review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment.

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

*A. Information Needed to Complete the Acceptance Review*

*1. Safety Functions of Major Components*

*NRC Review Comment 1.a*

***The LAR should completely delineate the performance requirements for the STC and HI-TRAC cask and include a failure modes and effects analysis to demonstrate that evaluations have been performed to show that safety functions will be accomplished for design basis events and other credible failures. At a minimum, the performance requirements should describe how each major component contributes to the safety functions described in Title 10 of the Code of Federal Regulations (10 CFR) Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC 61 and GDC 62 (i.e., shielding, confinement, decay heat removal, and criticality prevention). Refer to NUREG-0800, Standard Review Plan, for additional guidance, especially Chapter 15. For example, the application should contain a sufficiently broad spectrum of accidents and initiating events including the hazard and events further addressed below. Also, for accident analysis acceptance criteria, the design should be robust enough that all the postulated accidents produce about the same level of risk.***

**Response to 1.a**

(a) To comply with the Staff's request, the component description (Sections 1.3), fuel transfer operations (Section 1.4), applicable loadings (Section 3.2), acceptance criteria (Chapter 2), and analyses to quantify margins of safety under all applicable loadings (Chapters 6, 7) in the Licensing Report (Holtec Report No. HI-2094289) are herewith supplemented by Table 1.1, which provides the performance requirements of the sub-components of the STC and HI-TRAC in accordance with GDC-61 and 62. Table 1.2 provides the failure mode effects analysis for possible equipment failures associated with the fuel transfer. The potential failure modes presented in Table 1.2 are either:

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

Table 1.3 identifies the accidents or initiating conditions and discusses the resultant effects.

The information presented in this response is also supplemented by the "Compliance Matrix" provided in Response 8.c.

All chapter, section, figure or table references in this response correspond to the Licensing Report unless otherwise stated.

| Table 1.1 Performance Requirements of the HI-TRAC and STC | | | |
|---|---|---|---|
| No. | Equipment | GDC Criterion | Performance Requirements |
| 1 | STC pressure vessel | 61 (1), (2), (3) | Maintain structural integrity<br>Provide shielding<br>Maintains water in the STC cavity<br>Protects fuel |
| 2 | STC basket | 62 | The stainless steel basket maintains the fuel in a sub-critical geometry.<br>The metamic neutron absorber panels maintain the sub-criticality of the fuel. |
| 3 | STC seal, and vent and drain valves | 61 (1), (3), (5) | Maintains water in STC |
| 4 | HI-TRAC | 61 (1), (2), (3), (5) | Maintain structural integrity<br>Provide shielding<br>Maintains water in the annulus space<br>Protects fuel |
| 5 | HI-TRAC Water Jacket | 61 (2), (4) | Provides shielding<br>Provides heat transfer mechanism |
| 6 | HI-TRAC pool lid seal and drain plug | 61 (1), (3), (5) | Maintains water in HI-TRAC |
| 7 | HI-TRAC top lid seal and recessed vent port | 61 (1), (3), (5) | Maintains water in HI-TRAC |

| Table 1.2 Failure Modes and Effects Analysis | | | | |
|---|---|---|---|---|
| No | Equipment | Mode of Failure | Effect | Mechanism to prevent failure/analysis of event |
| 1 | STC Seal | Loss of seal integrity | Loss of pressure in STC possibly resulting in boiling | Testing of seal prior to transfer in accordance with ANSI N14.5 as discussed in Chapter 10 ensures functionality of the STC seal. |
| 2 | STC vent and drain valves | Release from vent and drain valves | Loss of pressure in STC possibly resulting in boiling | During the testing of the STC seal, while the STC cavity is pressurized, the valves will be closed and monitored to ensure they are not leaking. |
| 3 | STC drain tube | Mechanical failure | Improper air space in STC established resulting increased pressure in STC | Inspect drain line as part of lid inspection prior to installation for any blockage. |
| 4 | STC lid plugs | Loss of pressure integrity | Loss of pressure in STC possibly resulting in boiling | Plugs will meet the requirements of ASME Code Section III, Subsection ND.<br>During the testing of the STC seal, while the STC cavity is pressurized, the plugs will be tested in accordance with ANSI N14.5. |
| 5 | HI-TRAC pool lid seal and drain plug | Release of seal/plug resulting in loss of water in annulus space | Reduction in heat transfer and shielding | The pool lid seal is designated as safety related and will be purchased in accordance with an approved specification and supplied by an approved vendor.<br>Plugs will meet the requirements of ASME Code Section III, Subsection ND.<br>Prior to its initial use the HI-TRAC is tested in accordance with ASME Code ND-6000 as discussed in Chapter 8 to ensure acceptability of the HI-TRAC pool lid seal and drain plug.<br>Prior to each campaign the empty HI-TRAC will be tested in accordance with ANSI N14.5 and the seal and plug will be visually inspected for water leakage.<br>Prior to each transfer the HI-TRAC will be tested in accordance with ANSI N14.5 and the seal and plug will be visually inspected for water leakage. |

| Table 1.2 Failure Modes and Effects Analysis | | | | |
|---|---|---|---|---|
| No | Equipment | Mode of Failure | Effect | Mechanism to prevent failure/analysis of event |
| 6 | HI-TRAC top lid seal and vent port | Loss of seal integrity or any release from vent port | Loss of pressure in HI-TRAC possibly resulting in boiling in the annular space | Testing of seal prior to transfer in accordance with ANSI N14.5 as discussed in Chapter 10 ensures functionality of the HI-TRAC seal.<br>During the testing of the HI-TRAC seal, while the HI-TRAC cavity is pressurized, the vent port will be closed and monitored to ensure no pressure loss. |
| 7 | HI-TRAC water jacket relief device | Inadvertent opening of the relief device. | Loss of water shielding in HI-TRAC<br>Increase STC and HI-TRAC cavity pressure | The dose increase as a result of the loss of the water jacket shielding is negligible as discussed in Chapter 7.<br>STC and HI-TRAC cavity pressure remains below the design pressure limits as discussed in Chapter 5. |
| 8 | HI-TRAC Water jacket relief device | Failure to open at required pressure | Over-pressurization of water jacket with subsequent loss of water shielding in HI-TRAC | Bounded by inadvertent opening of the water jacket relief device. |
| 9 | VCT breakdown | Mechanical failure | Increase to site boundary dose | The dose increase to the site boundary is negligible as discussed in Chapter 7. |

| Table 1.2 Failure Modes and Effects Analysis | | | | |
|---|---|---|---|---|
| No | Equipment | Mode of Failure | Effect | Mechanism to prevent failure/analysis of event |
| 10 | Crane | Mechanical or electrical failure | Loss of water in STC, increased occupational dose | The crane will always have the ability to operate manually to lower the loaded STC and also move it side to side. If the STC is above the pool it can be manually lowered back into the pool. If the STC has already been lowered below the pool elevation, it can be manually aligned over the HI-TRAC and lowered into it. These operations are estimated to take no more than 10 hours to complete.<br><br>A Computational Fluid Dynamics simulation of the STC suspended in air inside the fuel handling building was performed.<br><br>The following conditions apply to the thermal analysis:<br>▪ STC contains the design basis decay heat 9.6 kW<br>▪ The STC is full of water<br>▪ STC lid is in place but bolts are not torqued<br>▪ Assumed air temperature in the fuel building: 100°F<br><br>The computed maximum water temperature in the STC with the above conditions is 195.8 °F; no localized boiling occurs. Therefore, the STC can remain suspended from the crane for an *indefinite* length of time, while maintaining the water temperature below boiling without any mechanical aid for heat removal.<br><br>ALARA principles will be employed until the STC is moved to the SFP or the HI-TRAC. |
| 11 | Temperature probes | Loss of signal | Unable to monitor temperature from a temperature probe | Temperature probes are inserted in specific locations around the basket (see Figure 2.1 in response 2.b). If there is a loss of signal in any temperature probe it will be replaced. |

| Table 1.3 Accident/Initiating Events and the Resultant Effects | | | |
|---|---|---|---|
| No. | Accident/Initiating Events | Effect | Discussion |
| 1. | Misloading of a fuel assembly which does not meet the design basis decay heat limits. | Increase in the total STC heat load possibly causing steaming and loss of water from the cavity resulting in uncovering the fuel and increased pressure inside the STC cavity.<br><br>Increase dose rate from STC. | Redundant administrative procedures in place will make the occurrence of a fuel misload unlikely.<br><br>The shortest time between reactor shutdown for a refueling outage and the fuel transfer campaign will be 3 months. A fuel assembly which has been recently discharged from the last core would have a visually detectable difference in appearance from a longer cooled fuel assembly.<br><br>A radiation survey will be performed on STC lid when removing it from SFP. If dose exceeds the expected dose rates, the STC will be lowered back into the SFP. The fuel assemblies will be re-verified to ensure a misload has not occurred.<br><br>Once the STC is placed in the HI-TRAC the water temperature is monitored for an extended period of time to ensure that heat load is below the design basis heat load before fuel transfer. |
| 2. | Crane malfunction while moving STC from SFP to HI-TRAC with a misload fuel assembly which does not meet the design basis decay heat limits. | Increase in the total STC heat load possibly causing steaming and loss of water from the cavity resulting in uncovering the fuel and increased pressure inside the STC cavity.<br><br>Increase dose rate from STC. | The probability of a fuel misload accident (based on discussion above) and a crane failure accident occurring concurrently is negligible. However, a defense-in-depth evaluation was performed. [PROPRIETARY TEXT REMOVED]<br><br>The crane will have the ability to operate manually to lower the loaded STC and also move it side to side. If the STC is above the pool it can be manually lowered back into the pool. If the STC has already been lowered below the pool elevation, it can be manually aligned over the HI-TRAC and lowered into it. These operations will be proceduralzed and are estimated to take no more than 10 hours to complete.<br><br>Therefore there is sufficient time to navigate the crane manually to bring the STC into the SFP or HI-TRAC where cooling and/or water addition will maintain the water in the STC cavity. |

| Table 1.3 Accident/Initiating Events and the Resultant Effects | | | |
|---|---|---|---|
| No. | Accident/Initiating Events | Effect | Discussion |
| 3. | Drop of loaded HI-TRAC | Release of HI-TRAC pool lid seal causing loss of water from the HI-TRAC cavity annulus which results in reduction of shielding and heat transfer. | The maximum height above a supporting surface which the loaded HI-TRAC can be lifted without the redundant drop protection is limited [PROPRIETARY TEXT REMOVED]. The lift height of the loaded HI-TRAC will be controlled as it is raised on the VCT to ensure this limit is not exceeded. Once the locking pins are engaged, attaching the HI-TRAC to the VCT and providing redundant drop protection, the lift height/carry is no longer limited and a drop accident during this part of the transfer is not credible.<br><br>A drop of the loaded HI-TRAC from the maximum lift height limit is presented in Load Case 5 in Chapter 6. |
| 4. | Earthquake | HI-TRAC tipover during the transfer operation causing loss of water from the HI-TRAC cavity annulus which results in reduction of shielding and heat transfer. | The stability of the HI-TRAC has been analyzed while it is connected to the VCT and resting on the fuel handling building floor under earthquake loadings. The accident analyses show that there will be no tipover of the HI-TRAC resulting in the loss of the annular water at any time during the transfer operation. (Section 6.2.6). |
| 5. | Environmental loadings | HI-TRAC tipover during the transfer operation causing loss of water from the HI-TRAC cavity annulus which results in reduction of shielding and heat transfer. | The loadings from an extreme environmental phenomena, such as high winds, tornado, and tornado-borne missiles, as specified for the 48 contiguous states in Reg. Guide 1.76, ANSI 57.9, and ASCE 7-88, are considered in the certification of HI-TRAC 100D in Docket No. 72-1014. These loadings bound the environmental loadings at IPEC. |
| 6. | Flood | HI-TRAC tipover during the transfer operation causing loss of water from the HI-TRAC cavity annulus which results in reduction of shielding and heat transfer. | The IP-2 and IP-3 FSB truck bay and transport haul path are well above the lowest building elevation and would require a rise in the river of over 55 feet to cause flooding; therefore the affect of the flood on the VCT is not considered credible and is not specifically analyzed. |
| 7. | Roadway collapse | HI-TRAC tipover during the transfer operation causing loss of water from the HI-TRAC cavity annulus which results in reduction of shielding and heat transfer. | Roadway has been evaluated to ensure no collapse under the pressure of the loaded VCT. (See response to NRC question 4). |

| Table 1.3 Accident/Initiating Events and the Resultant Effects | | | |
|---|---|---|---|
| No. | Accident/Initiating Events | Effect | Discussion |
| 8. | Misloading of a fresh fuel assembly | Criticality accident | A misloading condition where a fuel assembly not meeting the criticality requirements of the basket is analyzed. [PROPRIETARY TEXT REMOVED]<br>Additional calculations were therefore performed which credit the presence of soluble boron in the water. Results for those misloading conditions are summarized in Tables 4.7.11 and 4.7.12. With the soluble boron levels of at least 600 ppm the maximum $k_{eff}$ is below the limit of 0.95. |
| 9. | Fire | Reduction in heat transfer and increase in STC and HI-TRAC cavity pressure | Administrative controls will be implemented prior to each inter-unit transfer campaign to ensure there are no permanent or transient sources of fire in the vicinity of the transport path that create a condition outside the design basis fire analysis and of the HI-TRAC/STC assemblage.<br>An evaluation of the hazards along the haul path has been performed and is provided in Attachment A. [PROPRIETARY TEXT REMOVED]<br>The results show the fuel cladding temperature is within the SFST-ISG-11 limits; the maximum temperature of all materials are within design limits; The maximum STC and HI-TRAC pressures are within design limits. See Chapter 5. |
| 10. | Lightning | Ignition of the VCT fuel tank causing a design basis fire resulting in a reduction in heat transfer and increase in STC and HI-TRAC cavity pressure | [PROPRIETARY TEXT REMOVED]<br>The results show the fuel cladding temperature is within the SFST-ISG-11 limits; the maximum temperature of all materials are within design limits; The maximum STC and HI-TRAC pressures are within design limits. See Chapter 5. |

| Table 1.3 Accident/Initiating Events and the Resultant Effects | | | |
|---|---|---|---|
| No. | Accident/Initiating Events | Effect | Discussion |
| 11. | Tornado Missile | Loss of water in water jacket resulting in a reduction in shielding and heat transfer. Increase STC and HI-TRAC cavity pressure<br><br>Loss of water in the HI-TRAC annulus space. | The dose increase as a result of the loss of the water jacket shielding is negligible as discussed in Chapter 7. STC and HI-TRAC cavity pressure remains below the design pressure limits as discussed in Chapter 5.<br><br>The bottom flange of the HI-TRAC may be susceptible to tornado missiles therefore [PROPRIETARY TEXT REMOVED]<br><br>This is discussed further in the response to 1.c.<br><br>A tornado missile which strikes the top lid bolting may result in a loss of seal integrity; however the HI-TRAC remains in a vertical orientation and no loss of water will occur. Loss of pressure in the HI-TRAC annulus space with design basis heat load in the STC will have a negligible effect on the temperatures in the system.<br><br>The test port in the top lid of the HI-TRAC will [PROPRIETARY TEXT REMOVED] impervious to the tornado missile. |
| 12. | Improper Air Space in STC | Over-pressurization of the STC cavity. | The STC is an ASME Code compliant pressure vessel.<br><br>Redundant measures are taken to ensure the vapor space is correctly established. [PROPRIETARY TEXT REMOVED]<br><br>This establishes the height of the air space. Water will be collected or measured as it is removed from the STC and verified.<br><br>The pressure sensitivity to the height of the vapor space is presented in response to 2.c. |
| 13. | Improper Air Space in HI-TRAC | Over-pressurization of the HI-TRAC cavity. | The HI-TRAC is an ASME Code compliant pressure vessel.<br><br>Visual verification shall be made prior to installing the HI-TRAC top lid to ensure the correct water height is established.<br><br>The pressure sensitivity to the height of the vapor space is presented in response to 2.c. |

*NRC Review Comment 1.b*

***Section 1.3.2 of Enclosure 1 to the LAR states that the function of the HI-TRAC is to retain its contents under normal, off-normal, and accident conditions. That section also states that the HI-TRAC is designed to:***

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

***However, the LAR does not indicate if the annular water volume inside the HI-TRAC performs shielding or heat transfer safety functions. Consequently, the LAR did not include a thermal or shielding evaluation of the effect of the loss of the annular water volume in the HI-TRAC. Only the loss of the external jacket water was considered.***

**Response to 1.b**

The water in the HI-TRAC annulus renders both a heat transfer and shielding function.

Because of its heat transfer safety function, ensuring that the annulus water is not lost is a central objective in the system design. Loss of HI-TRAC annulus water accident scenarios have been evaluated in Tables 1.2 and 1.3 in this document, and is further discussed below.

Consistent with this objective, the design features engineered into the HI-TRAC cask are intended to ensure that the cask's cavity will serve as a high integrity water container. In particular, the cask flange-to-bottom lid joint and the drain plug have been qualified to Section III Subsection ND (Pressure Vessel) Code to a Design Pressure of 30 psig. Furthermore, the joint and the drain plug is hydrotested to 40 psig, which is considerably greater than the normal operating pressure (30 psig). Even in the case of a misloaded batch of fuel with a high heat load [PROPRIETARY TEXT REMOVED], the internal pressure in the HI-TRAC cavity (see Table 2.2) will not exceed the pressure at which the joint is hydrotested.

The large thermal inertia of the system ensures that the steady state pressures cited above will likely never be reached.

In summary, there is no credible mechanism for a loss of annulus water.

*NRC Review Comment 1.c*

***The referenced tornado missile analysis is incomplete because the safety functions performed by the HI-TRAC are different. Specifically, the referenced tornado analysis evaluated the HI-TRAC for penetration and deformation of the HI-TRAC shell and lid to demonstrate that the canister would not be penetrated and the canister would be retrievable; the analysis does not demonstrate that the HI-TRAC bolted lid connections would retain the annular water volume following a missile impact.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34 and GDCs 2, 61, and 62.***

**Response to 1.c**

The tornado missile strike directed to the HI-TRAC bottom flange joint is considered in Case #11 in Table 1.3 in this document.

A new ancillary [PROPRIETARY TEXT REMOVED] has been designed [PROPRIETARY TEXT REMOVED] to protect [PROPRIETARY TEXT REMOVED] from impulsive or impactive loads due to an incident tornado missile. [PROPRIETARY TEXT REMOVED]

For the bolted lid connections of the HI-TRAC top lid; after an accident involving a tornado missile the top lid will continue to remain in place. In the event some damage is done to the bolts the water will remain in the HI-TRAC. Additionally, it has been shown that the HI-TRAC always remains vertical, even after a tornado missile impact.

*2. Thermal (Heat Removal) and Containment (HI-TRAC and STC) Analyses*

*NRC Review Comment 2.a*

***Control of the STC internal temperature and pressure is based predominantly on factors managed by administrative controls (procedures) rather than design measures. Procedures control the heat load within the STC, the water volume within the STC, and the water volume in the annular space between the STC and the HI-TRAC. The heat transfer to the environment is significantly affected by the water volumes in both the STC and the HI-TRAC. The pressure response is highly dependent on the administratively controlled establishment of an air volume within the STC and HI-TRAC. In a water-solid condition within either the STC or HI-TRAC, the pressure change resulting from a small change in temperature could be significant.***

***Given the dependence on administrative controls, the licensee has not evaluated the temperature and pressure response resulting from: (1) the thermal misloading of a fuel assembly; (2) the failure to establish adequate water volumes in the STC and HI-TRAC, or (3) the failure to establish an adequate air volume in the STC and HI-TRAC. As further specified below, this information is needed to evaluate the potential***

***consequences and the prerequisite reliability of the administrative controls that is needed in preventing or mitigating these events. (Also, see item 1a above)***

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

**Response to 2.a**

The failure modes with a potential safety consequence have been considered in Table 1.2.

Specifically, the thermal and pressure responses resulting from the three operational errors cited above have been addressed as follows:

1. Thermal misloading of Fuel Assembly: Response 2.b, Tables 2.1 and 2.2 and Case 1 in Table 1.3

2. Failure to establish an adequate water volume in the STC and HI-TRAC: Response 2.c and Cases 12 and 13 in Table 1.3.

3. Failure to establish an adequate air volume in the STC and HI-TRAC: Response 2.c and Cases 12 and 13 in Table 1.3.

*NRC Review Comment 2.b*

***The application needs to contain a thermal analysis for misload of fuel assemblies that exceed specified decay heat limits. Misloads are credible events based on industry operational events in spent fuel pools and dry storage casks. The impact of a thermal misload on the proposed STC containment pressure boundaries and potential consequences are not analyzed.***

***The application indicates misloads will be detected by measuring and identifying "large differences in temperature" between STC cells. However, the application did not provide any information on the equipment, techniques, sensitivity, nor testing and validation of this method, which is needed to assure reliable identification of fuel assemblies, which have a potential wide range of decay heat.***

***The thermal misload analyses needs to consider the type of misloads that are credible, including the maximum possible decay heat, which could result in immediate heat-up problems during STS closure and handling operations; and moderate heat loads above the limits which could result in delayed heat-up problems during HI-TRAC transfer. The misload analyses should determine thermal safety margins with respect to the specified design pressure limits, and specify the potential consequence of exceeding the pressure limit. If a method for detecting thermal misloads is credited to mitigate the consequences, then the application should provide sufficient information on the equipment and validation of the method to demonstrate the reliability in preventing misloads is sufficient.***

***This information is needed for staff's review to ensure compliance with the criteria contained in GDC-61 and the intent of 10 CFR 72.128.***

**Response to 2.b**

For a postulated scenario wherein the STC contains fuel assemblies with a decay heat load [PROPRIETARY TEXT REMOVED] higher than the design maximum heat load), Computational Fluid Dynamics (CFD) simulation [PROPRIETARY TEXT REMOVED] shows that the STC and HI-TRAC cavity pressures are [PROPRIETARY TEXT REMOVED] below their permissible internal pressure [PROPRIETARY TEXT REMOVED]. Thus, an error resulting in a hypothetical fuel misload, resulting in a significant increase in the STC heat load, will not lead to overpressure in the STC or the HI-TRAC.

Table 2.1: Effect of Increased Heat Load on STC's Pressure and Temperature

| Case | Heat Load | STC Water, Peak Temp., °C | STC Water, Bulk Temp., °C | Cavity Pressure, psig |
|---|---|---|---|---|
| Design Basis Heat Load, Q | 9.6kW | 105 | 103 | 34 |
| [PROPRIETARY TEXT REMOVED] | | | | |

Table 2.2: Effect of Increased Heat Load on HI-TRAC's Pressure and Temperatures

| Case | Heat Load | HI-TRAC Water, Peak Temp., °C | HI-TRAC Water, Bulk Temp., °C | Cavity Pressure, psig |
|---|---|---|---|---|
| Design Basis Heat Load, Q | 9.6kW | 94 | 90 | 17.5 |
| [PROPRIETARY TEXT REMOVED] | | | | |

Additionally, the STC loading procedures will include steps to detect fuel misload. [PROPRIETARY TEXT REMOVED] This measurement will be performed after the STC lid has been installed and the STC is placed inside the HI-TRAC.

[PROPRIETARY TEXT REMOVED]

As an example, a transient thermal analysis of the STC [PROPRIETARY TEXT REMOVED] has been performed. During fuel loading, the temperature within the STC will rise with time. [PROPRIETARY TEXT REMOVED] If the measured temperature rise is at or below the predicted values [PROPRIETARY TEXT REMOVED] it is assured that the decay heat and the STC pressure are below the design limits.

The following operational step or equivalent shall be added to the operational procedures to ensure the placement of the temperature probes in the STC.

> Ensure calibrated temperature probes have been located at the designated locations. Measure and record the time and temperature every 30 minutes for each location over a period of no less than 3 hours. Compare the measured temperature rise from each temperature probe reading with the limiting temperature rise presented in Table 2.3. Ensure that the measured temperature rise from all the temperature probe readings is less than the limiting predicted temperature rise prior to continuing the loading evolution. If the measured temperature rise exceeds the predicted temperature rise in Table 2.3 for any of the probes, the STC will be returned to the spent fuel pool.

*NRC Review Comment 2.c*

***The design needs to provide a positive means of establishing the required air volume in the STC and the air volume in the HI-TRAC. Failure to establish an adequate water volume in the HI-TRAC or an adequate air gap in either the STC or HI-TRAC appears to be credible because any one of these results would require the incorrect performance of just one procedural step. Alternatively, the application needs to analyze failure to maintain an appropriate air volume as a credible accident. The use of an air volume appears to be critical in ensuring thermal-hydraulic performance during normal and accident conditions. The application indicates that performance of a pump will determine the air volume in the STC by loss of suction. However, based on operational experience, it appears that administrative errors, human factors, or undetected equipment malfunction in the pump system could result in not establishing the required air volume in the containment system.***

***This information is needed in order for the staff to proceed to perform a technical review of the spent fuel transfer system in accordance with GDC 61 and the intent of 10 CFR 72.4, 72.122, and 72.128.***

**Response to 2.c**

To provide additional assurance regarding establishment of the target vapor space in the STC cavity, the operational steps will be changed to reflect water removal from the STC using a gas blow down in place of a pump down for the main water removal. The change in method will eliminate potential problems associated with poor pump suction. Pressurizing the STC internals with air or nitrogen will force water out through the drain connection. [PROPRIETARY TEXT REMOVED] As a second confirmation that the proper amount of air space is developed during the blow-down, the amount of water that is removed from the cask will be measured.

The following operational changes shall be performed.

For the STC air space:

A water totalizer or collection tank shall be used to measure the amount of water removed prior to placing the STC in the HI-TRAC. Instead of using a pumping system to pump out the water, the blow down operation is performed by compressed air or nitrogen as follows:

- Connect a pressure source of compressed dry air or nitrogen regulated to $\leq$ 30 psig and blow down the water out through the drain to create the required air space.

- Confirm that the amount of water removed during blow down is sufficient[PROPRIETARY TEXT REMOVED].

As discussed above, redundant operational measures are deployed to ensure that the vapor space in the STC (achieved by expelling water at the assembly station) is properly controlled. Therefore, the existence of an inaccurate vapor space in the STC is not credible. However, even if a deviation were to occur, the effect is of no safety consequence, as established by the parametric CFD studies summarized in the Table 2.4.

It is noted from Table 2.4 that the total pressure in the vapor space increases as the initial space volume is decreased. The increase is largely due to the compression of the air due to volumetric expansion of water with increased temperature. The partial pressure due to the heat-up of water is subject to minimal change (the pressure must remain in equilibrium with the saturation temperature of water). The results tabulated in Table 2.4 also shows that even with a 20% inadvertent reduction in the STC cavity air space, the cavity pressure remains well below the normal pressure limit of 50 psig.

<table>
<tr><th colspan="5">Table 2.4: Effect of Reduced Vapor Space on the STC Cavity Pressure<br>(STC in the HI-TRAC Scenario during normal on-site transfer)<br>(Q = 9.6kW)</th></tr>
<tr><th></th><th>Case</th><th>Vapor Space Height, in</th><th>Vapor Space Volume, in$^3$</th><th>Water Temperature<br>STC Vapor Space Pressure, psig</th></tr>
<tr><td>1.</td><td>Design Nominal</td><td colspan="2" rowspan="3">[PROPRIETARY TEXT REMOVED]</td><td>34</td></tr>
<tr><td>2.</td><td>90% of the nominal vapor volume</td><td>38.1</td></tr>
<tr><td>3.</td><td>80% of the nominal vapor volume</td><td>44.9</td></tr>
</table>

<u>For the HI-TRAC vapor space</u>:

For establishment of the HI-TRAC required minimum air space a direct visual measurement is used prior to placement of the HI-TRAC top lid. This measurement will be verified prior to installation of the lid. Nevertheless, if a deviation is assumed, a significant margin to ensure seal worthiness of the joints remains. The Table 2.5 provides the results of parametric variation on the vapor space. However, if the air volume is increased, thereby decreasing the water volume inside the annulus, the HI-TRAC cavity pressure will reduce. Moreover, the temperature field does not change significantly because the level of water is much higher than the fuel.

As is seen from the Table 2.5, the pressure in the HI-TRAC cavity space increases as the volume of the air space is decreased. Under steady state conditions, the space above the HI-TRAC water mass is filled with compressed air and water vapor, the latter being at a partial pressure in psychometric equilibrium with the water below. The results tabulated in Table 2.5 also shows that even with a 20% inadvertent reduction in the HI-TRAC cavity air space, the cavity pressure remains well below the normal pressure limit of 30 psig.

| Table 2.5: Effect of Reduced Vapor Space on the HI-TRAC Internal Pressure | | | | |
|---|---|---|---|---|
| | Case | Vapor Space Height, in | Vapor Space Volume, $in^3$ | Annulus Cavity Pressure, psig |
| 1. | Nominal | [PROPRIETARY TEXT REMOVED] | | 17.5 |
| 2. | 90% of the nominal volume | | | 18.6 |
| 3. | 80% of the nominal volume | | | 20.2 |

***<u>NRC Review Comment 2.d</u>***

***The application needs to provide design information on the gasket sealing systems, including the gasket dimensions, elastomeric seal material, and justification that the seal will perform as required during normal and accident conditions. This information does not appear to be present in the application, and is necessary to determine the confinement will function during use.***

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

**Response to 2.d**

[PROPRIETARY TEXT REMOVED] To ensure that the seal will perform as required during normal and accident conditions, each STC lid stud will be preloaded in accordance with ASME Code, Section III, Division 1 requirements. This is adequate to (a) seat the gasket and (b) maintain compression between the STC lid and the STC flange under the accident internal pressure (65 psig).

The final gasket dimension for the HI-TRAC top lid will be established prior to fabrication. [PROPRIETARY TEXT REMOVED] To ensure that the seal will perform as required during normal and accident conditions, each lid stud will be preloaded in accordance with ASME Code, Section III, Division 1 requirements. This is adequate to (a) seat the gasket and (b) maintain compression between the lid and the flange under the accident internal pressure (40 psig).

The gasket dimension for the HI-TRAC pool lid [PROPRIETARY TEXT REMOVED]. To ensure that the seal will perform as required during normal and accident conditions, each pool lid stud will be preloaded in accordance with ASME Code, Section III, Division 1 requirements. This is adequate to (a) seat the gasket and (b) maintain compression between the lid and the flange under the accident internal pressure (40 psig).

*NRC Review Comment 2.e*

***The application needs to justify that the fire parameters used in the fire analyses bound those expected for a potential fire affecting the casks. Although the use of Part 71 values is appropriate for general certification of transportation packages, the application needs to verify that the actual HI-TRAC/STC physical characteristics and fuel fire characteristics for the analyzed accident are appropriate. Also, provide a list with the location of any significant permanent fire hazards located close enough to the haul path to affect the transporter and cask.***

***If a fire should occur during transport, the application should specify the expected fire temperatures at the cask surfaces and the duration of the fire.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDCs 3 and 61.***

**Response to 2.e**

Please see the response in Case 9 of Table 1.3 of this document.

*NRC Review Comment 2.f*

***The application in the acceptance test and maintenance program needs to specify leak testing requirements for the entire confinement boundaries of both the STC and HI-TRAC. The leak testing requirements should clearly specify the frequency and allowable leakage rates to ensure the protocols are generally consistent with the requirements for design, fabrication, periodic, and maintenance tests specified in American National Standards Institute (ANSI) standard N14.5.***

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

**Response to 2.f**

With respect to the Pressure Testing, a one time pressure testing (hydrostatic or pneumatic) of the STC shall be performed in accordance with the requirements of the ASME Code Section III, Subsection ND, Article ND-6000 and applicable sub-articles prior to the first fuel loading. This test will be done at the fabrication facility. If hydrostatic testing is used, the STC shall be pressure tested at not less than 125% of Design Pressure. In accordance with ND-3112.1 and NCA-2142.1, the STC Design Pressure is 50 psig. Therefore, to satisfy ASME Code Subsection ND requirements and for added conservatism, hydrostatic testing of the STC will be performed at 65 psig. If pneumatic testing is used, the STC shall be pressure tested to 120% of the Design Pressure (60 psig). The STC vent and drain ports will be used for pressurizing the cavity. Following completion of the required hold period at the test pressure, the surface of the STC welds shall be re-examined by liquid penetrant examination in accordance with ASME Code, Section III, Subsection ND, Article ND-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

A one time pressure testing (hydrostatic or pneumatic) of the HI-TRAC shall be performed in accordance with the requirements of the ASME Code Section III, Subsection ND, Article ND-6000 and applicable sub-articles prior to the first fuel loading. If hydrostatic testing is used, the HI-TRAC shall be pressure tested at not less than 125% of Design Pressure. In accordance with ND-3112.1 and NCA-2142.1, the HI-TRAC Design Pressure is 30 psig. Therefore, to satisfy ASME Code Subsection ND requirements and for added conservatism, hydrostatic testing of the STC will be performed at 40 psig. If pneumatic testing is used, the HI-TRAC shall be pressure tested to 120% of the Design Pressure (36 psig). Following completion of the required hold period at the test pressure, a visual inspection of accessible areas will be made and any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

Maintenance pressure testing (as defined in ANSI N14.5) of the STC and HI-TRAC is not required following the initial acceptance tests to verify continuing performance unless repairs of the equipment were performed. If the STC or HI-TRAC is repaired then re-test will be required as described above.

With respect to the leak testing, the STC and HI-TRAC closures are tested prior to each fuel transfer as described below. Therefore, periodic leak testing (as defined in ANSI N14.5) of the STC and HI-TRAC is not required.

The following operation steps (in Chapter 10 of the licensing report) shall be performed after the STC is placed in the HI-TRAC and the STC lid bolts are tightened.

> The STC will be pressurized to 55 +5/-0 psig with air or nitrogen and held for 10 minutes. Leak tests of the STC lid seal and lid penetrations are

performed per ANSI N14.5, Section A.5.7 (Soap Bubble Test Method). The acceptance criterion is no observed bubbles caused by leakage.

The HI-TRAC will be pressurized to 30 +5/-0 psig with air or nitrogen and held for 10 minutes. Leak tests of the HI-TRAC top lid seal and lid penetrations are performed per ANSI N14.5, Section A.5.7 (Soap Bubble Test Method). The acceptance criterion is no observed bubbles caused by leakage. A leak check of the HI-TRAC pool lid and drain plug is also performed. The acceptance criterion is no observed leakage of water.

*NRC Review Comment 2.g*

***The application should provide a thermal analysis for the loaded STC and consider a misloaded recently irradiated fuel assembly in the event there is a crane malfunction while moving the STC from the spent fuel pool to the HI-TRAC. The duration of the analysis should consider the time needed to repair the crane or use manual crane overrides to return the STC to the spent fuel pool. If the water in the STC reaches the boiling point, the application should show if vents could restrict steam release and result in pressurization of the STC.***

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

**Response to 2.g**

The crane malfunction leading to the scenario of an immobilized STC loaded with fuel has been considered as a failure mode in Case 10 in Table 1.2 and as an accident condition in combination with a fuel misload in Case 2 of Table 1.3. For the accident condition, the STC is conservatively assumed to have a recently irradiated fuel assembly. [PROPRIETARY TEXT REMOVED] The minimum time required for the water to reach boiling is 12 hours. As the water inside the STC begins to boil, the water level starts to decrease. Additionally, the time required for the water level to decrease to the top of the active fuel height is 2.5 hours. In total, it will take 14.5 hours for water level inside the STC at an initial temperature of 100°F to decrease to the top of the active fuel zone. This time period is sufficient to take compensatory measures.

*NRC Review Comment 2.h*

***The application needs to include the calculation package(s) for the thermal-hydraulic analyses of the system during normal and accident conditions. The package should provide design inputs, modeling assumptions, and evaluation of calculation values. The staff needs to verify that the thermal hydraulic system is appropriately analyzed and modeled, in order to verify the acceptability of the temperatures and pressures reported in the application.***

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

**Response to 2.h**

The calculation package HI-2084146R4 is provided in Attachment D.

*3. Criticality Analyses*

*NRC Review Comment 3.a*

***Provide code validation for using the CASMO-4 code, including input data and results, for fuel assembly isotopic concentration for the spent fuel assemblies to be transferred.***

***On page 4-16, the Safety Analysis Report indicates that CASMO-4, version 2.05.14, was used in determining the isotopic concentrations of the spent fuel assemblies to be transferred from IP3 to the IP2 spent fuel pool. However, the staff was unable to find any detailed information regarding the validation of the CASMO-4 code for spent fuels that have been discharged from the core and cooled for various times.***

***In accordance with ANS 8.1, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," Chapter 9 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" and NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," computer codes that are used to calculate effective multiplication factor, keff, and the isotopic concentration of the spent fuel must be validated. The Entergy application, however, does not provide any detailed information concerning the validation of the CASMO-4 code for spent fuel isotopic concentration analysis. The licensee is requested to provide supplementary information for the CASMO-4 computer code that is used in the application for determination of isotopic concentration of the spent fuels to be transferred from IP3 to the IP2 spent fuel pool using the STC.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68.***

**Response to 3.a**

The CASMO code is used for two purposes in the STC analysis for IP3: determination of the spent fuel composition (fuel depletion), and determination of the $k_{inf}$ of fresh and spent fuel for establishing reactivity effects (delta-k) of manufacturing uncertainties or operating conditions (temperature).

For the fuel depletion part, no specific validation is provided. Instead, 5% of the reactivity decrement due to fuel depletion is considered as an additional uncertainty, consistent with the Kopp Memorandum (Ref. [13] in HI-2084176, Attachment J):

For the ability of CASMO to determine $k_{inf}$ values and subsequent reactivity differences, the initial application provided Benchmarking calculations performed by the CASMO code developer (Studsvik), determined a bias and bias uncertainty for those, and included the bias uncertainty with the other uncertainty in the calculation of the maximum $k_{eff}$. These benchmarking calculations were for standard critical experiments with fresh fuel.

To expand this approach to validate CASMO for criticality calculations for spent fuel, standard critical experiments with spent fuel would be required. However, such experiments do not exist, or are not generally available. A direct validation of CASMO for criticality calculations using the usual approach by comparison with spent fuel critical experiments is therefore not possible.

Instead, a different approach for CASMO validation is used here, which is based on code-to-code comparisons between CASMO and MCNP. It is recognized that code-to-code comparisons are a less-preferred approach for code validations. However, note that the Kopp Memorandum specifically states that "A comparison with methods of analysis of similar sophistication (e.g., transport theory) may be used to augment or extend the range of applicable critical experiment data". Nevertheless, the approach taken here does not attempt to generically validate CASMO. Instead, it focuses only on the CASMO evaluations performed for the IP3 STC, in an attempt to validate those. Since CASMO is only used to determine reactivity difference for small changes in certain parameters, the code-to-code comparison compares only reactivity differences, and not absolute $k_{eff}$ or $k_{inf}$ values. The comparisons are performed for selected reactivity differences already determined by CASMO, i.e. the reactivity difference of the exact same parameter variation is determined by MCNP for selected cases, and then compared to the corresponding value determined by CASMO. This approach avoids any applicability issues in the code-to-code comparison.

Before the approach is presented in more detail, an evaluation of the currently used CASMO reactivity differences and its use in the design basis analyses is in order. A review of the calculated reactivity effects of the tolerances (Appendix C in HI-2084176) shows that for all tolerances and for the overall combination, the maximum positive reactivity effect corresponds to fresh fuel, either at the highest or lowest enrichment analyzed. Further, as a conservative simplification of the calculations, only the maximum combined reactivity effect of any burnup and cooling time combination is used in the design basis calculations for both spent and fresh fuel. Since these values are for fresh fuel, they are validated through the benchmarking calculations provided in Appendix B of HI-2084176 (Attachment J). This limits the necessity for validation of CASMO for spent fuel to demonstrating that the reactivity effects for the uncertainties does not exceed the value currently used in the design basis calculation. This value is determined as follows:

| | |
|---|---|
| Uncertainty of Basket Manufacturing Tolerances | 0.0036 delta-k |
| Uncertainty of Fuel Tolerances | 0.0087 delta-k |
| CASMO uncertainty (fresh fuel) | 0.0035 delta-k |
| | ------------------ |

| | |
|---|---|
| Statistically Combined (CASMO) | **0.0100 delta-k** |

For the code-to-code comparison, the reference case and all fuel and basket variations are re-calculated in MCNP, for the highest and lowest enrichment of spent fuel analyzed, i.e. 2.0 and 4.95 wt%. Note that these calculations are performed with larger number of particles than the design basis calculation, to reduce the statistical uncertainty of the calculated differences. The results are then processed in the same fashion as the CASMO results, i.e. differences to the reference case are calculated, and the positive differences are statistically combined. Note that as a simplification, all differences from the MCNP calculations were determined at the 95% level. This is conservative, since it assumes that all individual reactivity differences are concurrently at the upper bound level. The results, in the form of the combined uncertainties, are presented below (all details are included in the updated report HI-2084176).

2.0 wt% initial enrichment, 5.22 GWd/MTU:

| | |
|---|---|
| Uncertainty of Basket Manufacturing Tolerances | 0.0030 delta-k |
| Uncertainty of Fuel Tolerances | 0.0075 delta-k |
| | ------------------ |
| Statistically Combined (MCNP) | **0.0081 delta-k** |

4.95 wt% initial enrichment, 38 GWd/MTU:

| | |
|---|---|
| Uncertainty of Basket Manufacturing Tolerances | 0.0032 delta-k |
| Uncertainty of Fuel Tolerances | 0.0052 delta-k |
| | ------------------ |
| Statistically Combined (MCNP) | **0.0061 delta-k** |

For both enrichments, the combined reactivity effect is bounded by the value used in the design basis calculation stated above. Note that for calculations with 600 ppm soluble, the same conclusion is drawn. This verifies that the use of CASMO for the IP3 STC for determining the reactivity effect of uncertainties and its application in the design basis analyses is appropriate and conservative.

For further details of those analyses see revised Holtec Report HI-2084176 (Proprietary) [Attachment J]

***<u>NRC Review Comment 3.b</u>***

***Provide justification for the applicability of the selected critical experiments to the code benchmark and upper safety limit (USL) calculation of the Indian Point STC criticality calculation. Provide, if necessary, an updated USL with additional applicable benchmark experiments included.***

***Table 4.A.1 provides information on the selected critical experiments that were used in the code benchmark and USL calculation for the Indian Point STC criticality safety***

***evaluation. However, it appears that these critical experiments may not be adequate for this purpose because the fuel compositions from these experiments are very different from that of the IP3 spent fuels. In general, the IP3 spent fuels contain various low quantities of fissile materials such as U-235 and Pu-239 and many actinides and fission products. Therefore, it may be inadequate to use fresh fuel and mixed-oxide (MOX) fuel critical experiments to determine the code performance in terms of bias and uncertainties. The licensee is requested to provide supplemental information that can justify the applicability of these experiments for the spent fuel STC.***

***If the current set of benchmark experiments needs to be augmented by additional benchmark calculations, provide updated critical experiments as necessary.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68 and NUREG-0800.***

**Response to 3.b**

After the clarification that the 5% depletion uncertainty per the Kopp Memo (Ref. [13] in HI-2084176) does not include the validation of the criticality calculations, the validation for MCNP faces the same dilemma as CASMO (see answer to question 3.a), namely the lack of standard critical experiments with spent fuel. However, in the case of MCNP, Holtec has performed comprehensive validation calculations in the context of taking burnup credit for Holtec's transportation cask, HI-STAR 100. These calculations also included benchmarking calculations for fission products and minor actinides, based on evaluations of Commercial Reactor Criticals (CRCs). The methodology is documented in detail in the HI-STAR Safety Analysis Report (SAR) and its supporting calculations, which were submitted under Docket 71-9261. The SAR and the supporting reports are Holtec Proprietary documents. Therefore, only a brief overview is given here and the reader is referred to the applicable sections of the HI-STAR SAR. The burnup credit for the HI-STAR 100 was reviewed by the NRC and approved in 2006.

For a detailed description of the CRC Benchmarking calculation see HI-STAR SAR, Appendix 6.E, Section 6.E.3.3. It contains a detailed description of the calculational sequences of the analyses that were performed, and presents the results. It also contains a detailed comparison between the CRC condition and the condition in the MPC-32 basket within the HI-STAR cask. It is important to note that the 12-assembly basket for the STC was directly derived from the MPC-32, essentially by removing the outer 20 cells. Cell dimensions and construction are essentially identical between the MPC-32 and the STC. Further, the qualified fuel assemblies are also essentially the same, being the Westinghouse 15x15 assemblies for the STC, and both Westinghouse 17x17 and B&W 15x15 assemblies for the MPC-32. The CRCs are therefore applicable to the STC in the same way they are to the MPC-32. The only notable difference between the MPC-32 and STC, however, is the fact that the STC will be filled with borated pool water, which provides a significant additional (although largely uncredited) safety margin, whereas the MPC-32 is presumed to be flooded with fresh water under all normal, off-normal and accident conditions. [PROPRIETARY TEXT REMOVED]. To validate the use of MCNP with spent fuel, the results of the CRC benchmarking are applied to the STC design basis calculations as follows:

- [PROPRIETARY TEXT REMOVED]
- The bias uncertainty from the CRCs are applied as bounding values as a function of burnup and enrichment, [PROPRIETARY TEXT REMOVED].

For further details of those analyses see updated Holtec Report HI-2084176 (Proprietary). Note that no adjustment to the loading curve is required as a result of incorporating this additional uncertainty. This is due to the fact that the loading curve was taken from the IP2 and IP3 spent fuel pool Tech Specs, and still contained margin due to the improved neutron absorption of the STC compared to the spent fuel pool racks.

*<u>NRC Review Comment 3.c</u>*

***Metamic neutron absorber plates are credited in the criticality analysis. However the qualification of Metamic for its safety-related use in the STC is not provided. Therefore, the staff requests the results of the qualification of Metamic for its safety-related use in the STC. The licensee shall provide the service conditions and design requirements identified for the life of the STC. The licensee shall provide the neutron absorber material qualification performed. The qualification must include neutronic as well as mechanical aspects of the Metamic neutron absorber plates necessary for them to perform their safety-related function in the STC. If reliance is placed on precedents, those precedents should be explicitly identified and differences between those precedents and the current application must be identified and justified as to why the precedent remains valid. There may be a need for a periodic surveillance of these plates to check for degradation.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.68.***

**Response to 3.c**

The Metamic™ neutron absorber material, proposed for use in the STC, is manufactured by the Holtec Nanotec Materials Division in Lakeland, Florida. The primary function of the Metamic panels in the STC basket is to ensure criticality safety. These panels are not credited in structural/mechanical evaluations of the STC. [PROPRIETARY TEXT REMOVED]

Metamic™ has been subjected to rigorous tests by various organizations including Holtec International, and has been approved by the USNRC in recent dry storage applications for Holtec (Dockets 71-9261, 71-9336 and 72-1014) as well as recent wet storage applications for Arkansas Nuclear One Units 1 and 2 (Dockets 50-313 and 50-368, [3c.1]), Clinton (Docket 50-461), Diablo Canyon Units 1 and 2 (Dockets 50-275 and 50-323), St. Lucie Unit 2 (Docket 50-389), Turkey Point Unit 3 (Docket 50-250) and Cooper Nuclear Station (Docket 50-298). The USNRC has also previously approved Metamic™ for use in other dry storage [3c.2 and 3c.11] applications.

Metamic™ was developed in the mid-1990s by the Reynolds Metals Company [3.c.7] with the technical support of the Electric Power Research Institute (EPRI) for spent fuel reactivity control in dry and wet storage applications with the explicit objective to eliminate the performance frailties of aluminum cermet type of absorbers reported in the industry. Metallurgically, Metamic™ is a metal matrix composite (MMC) consisting of a matrix of aluminum reinforced with Type 1 ASTM C-750 boron carbide. Metamic™ is characterized by extremely fine aluminum (325 mesh or smaller) and boron carbide ($B_4C$) powder. Typically, the average $B_4C$ particle size is between 10 and 40 microns. The high performance and reliability of Metamic™ derives from the fineness of the $B_4C$ particle size and uniformity of its distribution, which is solidified into a metal matrix composite structure by the powder metallurgy process. This yields excellent homogeneity and a porosity-free material. An array of U.S. patents discloses the unique technologies that underlie the Metamic™ neutron absorber [3.c.3-3c.6].

Metamic™ is a porosity-free material and there is no capillary path through which water can penetrate Metamic™ panels. To determine its physical stability and performance characteristics, Metamic™ was subjected to an extensive array of tests sponsored by EPRI that evaluated the functional performance of the material at elevated temperatures (up to 900°F) and radiation levels (1E+11 rads gamma). The results of the tests documented in an EPRI report [3c.7] indicate that Metamic™ maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state. Accelerated corrosion testing was also performed on multiple Metamic® coupons at 195 °F for 9020 hours in both deionized water to simulate boiling water reactor pool conditions, and deionized water containing 2500 parts per million boron as boric acid to simulate pressurized water reactor pool conditions. The main conclusions provided in the above-referenced EPRI report, which endorsed Metamic™ for dry and wet storage applications on a generic basis, are summarized below:

- The metal matrix configuration produced by the powder metallurgy process with almost a complete absence of open porosity in Metamic™ ensures that its density is essentially equal to the theoretical density.

- The physical and neutronic properties of Metamic™ are essentially unaltered under exposure to elevated temperatures (750° F - 900° F).

- No detectable change in the neutron attenuation characteristics under accelerated corrosion test conditions has been observed.

Additional technical information on Metamic™ in the literature includes independent measurements of boron carbide particle distribution in Metamic™ panels, which showed extremely small particle-to-particle distance [3c.8].

Metamic™ has also been subjected to independent performance assessment tests by Holtec International in the company's laboratories since 2001 [3c.9, 3c.10]. The three-year long experimental study simulated limiting wet and dry environmental conditions and included tests to determine mechanical strength, elongation, uniformity in $^{10}B$ distribution and corrosion. No anomalous material behavior was observed in any of the tests. Corrosion tests were made with Metamic coupons of 25%, 32% and 40% $B_4C$ for a 90-day exposure period at 200°F both in demineralized water and in boric acid solution at 2000 ppm boron. Results of these tests indicate that no corrosion was observed, with only minor increases in weight. Neutron attenuation measurements of the test coupons subjected to these corrosion tests also showed no significant change in $^{10}B$ areal density. These independent Holtec tests essentially confirmed earlier EPRI and other industry reports cited in the foregoing with regard to the suitability of Metamic™ as a neutron absorber in fuel transfer applications.

Holtec International's Quality Assurance Program ensures that Metamic™ will be manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." Consistent with its role in reactivity control, all neutron absorbing material procured for use in the STC will be categorized as Safety Related (SR). Due to the extensive history on the testing and use of Metamic both in dry and wet environment, which shows that there is no degradation of the Metamic over time, a periodic surveillance of these plates is deemed unnecessary.

**References for Response 3.c**


[3.c.1] "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications," Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., Docket No. 50-313 and 50-368, USNRC, June 2003.

[3c.2] USNRC Docket No. 72-1004, NRC's Safety Evaluation Report on NUHOMS 61BT (2002).

[3c.3] U.S. Patent # 6,332,906 entitled "Aluminum-Silicon Alloy formed by Powder", Thomas G. Haynes III and Dr. Kevin Anderson, issued December 25, 2001.

[3c.4] U.S. Patent # 5,965,829 entitled "Radiation Absorbing Refractory Composition and Method of Manufacture", Dr. Kevin Anderson, Thomas G. Haynes III, and Edward Oschmann, issued October 12, 1999.

[3c.5] U.S. Patent # 6,042,779 entitled "Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal and Super Hypereutectic Al/Si Alloys", Thomas G. Haynes III and Edward Oschmann, issued March 28, 2000.

[3c.6] U.S. Patent Application 09/433773 entitled "High Surface Area Metal Matrix Composite Radiation Absorbing Product", Thomas G. Haynes III and Goldie Oliver, filed May 1, 2002.

[3c.7] "Qualification of METAMIC® for Spent Fuel Storage Application," EPRI, 1003137, Final Report, October 2001.

[3c.8] "METAMIC Neutron Shielding", by K. Anderson, T. Haynes, and R. Kazmier, EPRI Boraflex Conference, November 19-20 (1998).

[3c.9] "Use of METAMIC® in Fuel Pool Applications," Holtec Information Report No. HI-2022871, Revision 1 (2002).

[3c.10] "Sourcebook for Metamic™ Performance Assessment" by Dr. Stanley Turner, Holtec Report No. HI-2043215, Revision 2 (2006).

[3c.11] NAC-UMS® Universal Storage System Safety Evaluation Report Amendment No. 3, USNRC Docket 72-1015, 2004.

*4. Transport Roadway Analysis and Cask Tipover Accident*

*NRC Review Comment 4*

***In a letter dated June 11, 2009, ADAMS Accession No. ML091520167, the NRC staff set forth draft fuel cask evaluation criteria. One of these criteria was an analysis for a cask tipover event. In the LAR, the licensee stated that cask tipover was not a credible accident, as the roadway would be qualified for the load, and the vertical cask transporter (VCT) has a redundant drop protection feature. However, there was no roadway analysis provided, just a commitment to perform one before the first cask movement. In order to meet the requirements of GDC 4 (environmental conditions and dynamic effects of missiles (i.e. dropped load) on affected structures, systems, and components (SSCs) that are important to safety) and GDC 61 (adequate safety of the transfer cask with the STC loaded with spent fuel under normal and postulated accident conditions) during the heavy load movement along the haul path, please provide the appropriate analyses in accordance with NUREG-0612.***

***Please provide the following for staff review: (i) the proposed haul path for the heavy load movement; and (ii) the evaluation of all affected important-to-safety (includes safety-related) SSCs along and adjacent to the haul path that demonstrate that these SSCs are capable of withstanding applicable loads and load combinations resulting from the heavy load movement, including those required by GDC 2 and GDC 4, with an acceptable margin of safety. Also, (i) establish that the potential for an accident condition along the haul path is minimized by providing analyses, load test details and/or operating experience that demonstrate that the VCT is of a single-failure-proof design and the entire transport roadway (haul path) has adequate strength capacity under applicable design loads and load combinations during the proposed heavy load movement of spent fuel; or (ii) postulate a worst-case accident condition (e.g. cask tipover) along the haul path and demonstrate that the consequences are acceptable.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDCs 2, 4, and 61.***

**Response to 4.**

The heavy load analysis has been performed to demonstrate that the haul path has adequate load capacity to support the proposed heavy load movement of spent fuel between Unit 3 and Unit 2. The heavy load analysis report provides the haul path and the evaluation underground and applicable adjacent features. Entergy's heavy load analysis report document no. FCX-00570) is provided (Attachment E).

The haul path is being evaluated and modified as necessary to ensure the travel requirements of the VCT are maintained along the entire travel path. Review of plant drawings, surveys and Ground Penetrating Radar (GPR) are being performed. Underground utilities along the haul path will be addressed, as necessary, to ensure they remain unaffected. The haul path is being hardened with the installation of concrete runways and turning pads for the VCT to travel on and to eliminate significant degradation to the haul path surface.

As stated in the LAR, the VCT used to transfer the STC from Unit 3 to Unit 2 is that same VCT which has been qualified to transfer a loaded HI-STORM which can weigh in excess of 400,000 lbs. The VCT incorporates the redundant drop protection feature ensuring that the load can not be dropped. To ensure that the VCT is capable of carrying the load, a load testing of the VCT is performed using 125% of the weight of the HI-STORM 100. During the load test, the VCT is subjected to the load for 10 minutes. All welds on primary structural components are visually inspected following the 125% load testing to determine that welds are not damaged. VCT structural components are visually examined to verify load testing did not cause any deformation or cracking of the base material.

The total combined loaded weight of the HI-TRAC containing the STC and the water inside is less than 200,000 lbs which is significantly less than the test load applied to the VCT. Therefore, it is concluded that the existing VCT is adequate for use in transferring the HI-TRAC containing the STC.

Attachment F contains the VCT specification containing technical data on the VCT.

Attachment O contains the proposed haul path for the fuel transfer.

*5. Crane Design*

*NRC Review Comment 5*

***The licensee proposed a new commitment related to the crane design in Attachment 6 to the LAR. The commitment states:***

> ***The IP3 crane will be upgraded to a single failure proof crane meeting the intent of NUREG-0554 through the use of ASME [American Society of Mechanical Engineers] NOG-1-2004 as the governing design code.***

***The commitment is unclear because NUREG-0554 and ASME Standard NOG-1, 2004, have been accepted by the NRC staff only as design standards for entirely new cranes or entirely new portions of upgraded cranes. Appendix C to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines that supplement or provide alternatives to NUREG-0554 criteria for upgrades of existing cranes, particularly those related to unreplaced portions of upgraded cranes. However, when material properties and weld configurations for unreplaced structures can be reliably determined, the seismic and dynamic structural analysis methods presented in ASME NOG-1, 2004, may be used for the entire upgraded crane.***

***The licensee must clarify if the crane design upgrade itself is part of the LAR or will be implemented without NRC staff review, pursuant to the provisions of 10 CFR 50.59. If***

***the crane upgrade itself is part of the LAR, the licensee should provide a matrix listing the guidelines of NUREG-0554, as modified by the guidelines of Appendix C to NUREG-0612, and a brief description of how the intent of the guideline would be satisfied. Otherwise, the licensee should commit to the guidelines of Appendix C to NUREG-0612 and NUREG-0554, except that the criteria of ASME NOG-1, 2004, may be employed as an acceptable alternative to the NUREG-0554 criteria. Commitment to only the intent of an NRC-approved methodology is unacceptable when implementing the methodology, pursuant to the provisions of 10 CFR 50.59, without NRC staff review.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 1.***

**Response to 5.**

The inter-unit fuel transfer solution involves the complete replacement (bridge and trolley) of the existing IP3 cask handling crane with a single-failure-proof design while maintaining the 40-ton capacity. The replacement of the crane in not part of the LAR and will be implemented pursuant to the provisions of 10 CFR 50.59. Therefore, Entergy commits to the guidelines of Appendix C to NUREG-0612 and NUREG-0554, except that the criteria of ASME NOG-1, 2004, may be employed as an acceptable alternative to the NUREG-0554 criteria.

*6. STC Materials and ASME Boiler and Pressure Vessel Code (ASME Code) Requirements*

*NRC Review Comment 6.a*

***Holtec Licensing Report HI-2094289 (the submittal) identified the 2004 Edition of Section III of the ASME Code, Subsection ND, for stress limits. The submittal is silent on the edition and addenda of Section III of the ASME Code applicable to the fabrication, testing, and inspection of the STC. Provide the ASME Code, Section III, edition and addenda that apply to these activities.***

**Response to 6.a**

The fabrication, testing, and inspection of the STC is governed by the 2004 Edition of Section III of the ASME Code, which is the latest Code edition referenced in 10CFR50.55a, "Codes and standards". The note on the STC (Attachment G) and STC basket (Attachment H) drawings has been revised.

*<u>NRC Review Comment 6.b</u>*

***The STC is capable of being used for multiple cycles over its planned life. Provide a statement that identifies the planned life duration of the STC.***

**Response to 6.b**

The STC is designed for 40 years. The principal design considerations that bear on the adequacy of the STC for the service life are addressed as follows:

<u>Exposure to Environmental Effects</u>

All STC materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The STC is designed for repeated normal condition handling operations with high factor of safety to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the canister's materials, and therefore, will not lead to a fatigue failure in the STC. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the STC utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading, as discussed in Section 8.3 of the Licensing Report.

<u>Material Degradation</u>

As discussed in Chapter 8 of the Licensing Report, all STC materials that are susceptible to corrosion are coated. The controlled environment in which the STC is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the STC materials are subjected do not result in radiation embrittlement or degradation of the STC's shielding materials that could impair the STC's intended safety function. The STC materials are selected for durability and wear resistance for their deployment.

<u>Maintenance and Inspection Provisions</u>

The requirements for periodic inspection and maintenance of the STC throughout the 40-year design life are defined in Chapter 10 of the Licensing Report These requirements include provisions for routine inspection of the STC for damage prior to each use. Precautions are taken during lid handling operations to protect the sealing surfaces of the closure lid. The leak tightness of the STC pressure boundary is verified periodically.

Finally, based on the current inventory of fuel in the IP3 SFP and the projected discharge schedule through the plant re-license period, the maximum number of fuel transfers (uses) will not exceed 250. This number of uses can be accommodated within the 40 year design life of the STC.

*<u>NRC Review Comment 6.c</u>*

***Table 8.2.1 of the submittal provides cycle duration (8 hours to several days) for fuel transfer operations. For postulated accident conditions, paragraph 3.2.3.2(h) of the submittal used 30 days for dose calculations. Provide the limiting duration for an abnormal fuel transfer operation cycle and postulate your action if the duration limit is approached.***

**Response to 6.c**

The limiting duration for an abnormal fuel transfer is 30 days. The postulated event would be a VCT breakdown occurring one time in a campaign, essentially once per year. It is reasonable to assume that the VCT could be repaired or that another VCT could be secured from another site so the HI-TRAC could be moved to a fuel handling building and unloaded within this time. The largest dose contribution from this abnormal event is the postulated effluent dose, since the direct radiation dose from the HI-TRAC is very low (Table 7.4.2 of the licensing report).

The Holtec Users Group members collectively own over a dozen VCTs, less than half of which are in use at any time. In the case of the VCT failure at IPEC, the time to disassemble, ship, and assemble a substitute VCT is less than 10 days. (Holtec recently carried out such a VCT transfer from Salem to Byron Station without any difficulty.) Therefore, the 30-day time limit is adequate.

*<u>NRC Review Comment 6.d</u>*

***Table 8.2.1 of the submittal lists Carboguard 890 coatings for the STC surfaces. Although Section 8.3 states that coatings on carbon steel do not react with borated water, no information is provided to support this statement. The coating applied to the STC's interior surface is in contact with water for several days during the transfer of the spent fuel assemblies, and the coating may be in contact with water for a longer duration as a result of abnormal conditions. Provide technical data on the effects of abnormal time (using the answer to the question above) and temperature (Table 3.1.1 of the submittal) and radiation conditions on the interior coating integrity.***

**Response to 6.d**

Although Carboguard 890 has been extensively used in dry storage applications, scientifically calibrated data on its performance under long-term submergence is not available. Therefore, to eliminate paint as a source of concern, a metallic coating shall be used on the inside of the STC surface [PROPRIETARY TEXT REMOVED]. The metallic coating will be stainless steel. Stainless steel provides excellent corrosion resistance and is compatible with borated water. Stainless steel provides an inert surface to prevent corrosion at elevated temperatures. It should be noted that since the majority of the STC is in contact with water, the emissivity of the coating is not a critical characteristic in this application (i.e., it does not play a significant role in the thermal performance of the system).

[PROPRIETARY TEXT REMOVED]

***NRC Review Comment 6.e***

***The spent fuel cladding must be protected during storage against degradation. The Carboguard 890 product data sheet only identifies the generic ingredients. The product data sheet does not identify the residual and tramp elements that are in the generic ingredients, if any. These residual and tramp elements in the STC interior surface coating could potentially be detrimental to the fuel cladding. Provide technical data on leaching of elements (such as halogens) detrimental to the fuel basket and cladding as a result of abnormal time and temperatures exposure. If the data indicate elevated levels of detrimental elements, provide your mitigating action.***

**Response to 6.e**

Please see response to 6.d above.

*<u>NRC Review Comment 6.f</u>*

***Paragraph 8.5.1 of the submittal states that periodic structural or pressure tests are not required to verify continuing performance. The statement is without supporting technical data. The loading and unloading process may have an accumulative effect on the STC's weld integrity and interior coating integrity. Provide the non-destructive evaluation (NDE) methods, inspection frequency, and acceptance criteria for verifying weld and interior coating integrity is maintained over the planned life duration of the STC.***

**Response to 6.f**

At the end of each campaign or prior to a campaign the accessible parts of the STC and HI-TRAC will be visually examined to verify no deformation, distortion, or cracking occurred. Any evidence of deformation, distortion or cracking will require repair of the equipment. Following any repair, the pressure testing shall be performed as described in the Response 2.f. The interior coating applied will be stainless steel as discussed in Response 6.d. This coating is extremely resistant to the environment in which it will be used. Since all fuel is loaded into the individual cells of the basket they will not come into contact with the interior coating of the STC. As such there is no discernible mechanism for degradation (erosion or corrosion) of the STC internal surfaces.

With regard to the STC welds, they are designed for repeated normal condition handling operations with high factors of safety to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the weld material as demonstrated below.

Under normal operating conditions, the weld stress limit is 0.3Su. [PROPRIETARY TEXT REMOVED] Therefore, incorporating a fatigue strength reduction factor of 4 (conservatively taken from Table NG-3352-1), the effective stress amplitude for calculating usage factor using Figure I-9.1 (ASME Code, Section III Appendices) is 45ksi. [PROPRIETARY TEXT REMOVED] Using Figure I-9.1, the permissible number of cycles corresponding to this stress amplitude is 6,055. Therefore, since the number of STC loading campaigns is conservatively estimated at 250 (see Response 6.b), fatigue failure of the STC welds due to the loading and unloading process is not a concern.

*NRC Review Comment 6.g*

***Paragraph 8.5.2 of the submittal states that the STC seal will be tested prior to each fuel transfer. Although paragraph 8.4.4 and paragraph 10.2.3, procedure step 23 contains leakage test criteria, this information was not referenced in Paragraph 8.5.2. In Paragraph 8.5.2, provide a reference or description of the leakage test process; identify the inspection method, and provide inspection acceptance criteria.***

**Response to 6.g**

The elastomeric seals on the STC and HI-TRAC lid shall be replaced as defined in Table 8.5.1. The STC seal will be tested per ANSI N14.5, Section A.5.7 Soap Bubble test Method, prior to each fuel transfer. The acceptance criterion for the leakage test is no observed bubbles caused by leakage. Any observed leakage shall require the STC to be returned to the fuel pool and the lid removed for inspection of the seal and seal surfaces. Any damage to the seal or seal surface shall be reworked or repaired.

This is consistent with paragraph 8.4.4 and paragraph 10.2.3, procedure step 23

*NRC Review Comment 6.h*

***The STC has a lead shield sandwiched between the inner and outer carbon steel shell. The lead shield is used to reduce dosage in the surrounding area. For the lead shield, provide the NDE method and/or manufacturing process used to verify material soundness and shielding effectiveness. An example of fabricating lead shielding and testing for effectiveness is in Section 9.1.5 of the HI-STORM 100 Final Safety Analysis Report (FSAR).***

**Response to 6.h**

The lead shielding in the HI-TRAC, as described in Section 9.1.5 of the HI-STORM 100 is poured, therefore it is necessary to perform a shielding effectiveness test to ensure no voids are present.

The STC lead shielding will be installed as lead sheets. The design of the sections and the installations instructions shall minimize the gaps between adjacent lead sections and between the lead and the STC walls to the extent practicable.

The lead sheet will be layered for a minimum total thickness [PROPRIETARY TEXT REMOVED]. All sheets regardless of thickness shall be measured for thickness in at least four corner locations, at a minimum of two inches from any edge. The effectiveness of each lead sheet shall be verified by visual examination. The visual examination includes absence of cracks, pores, inclusions, scratches, grooves, or other types of defects that could impair the gamma shielding function of the lead. Defects which exceed a depth of 10% of the plate thickness and an area greater than one square inch shall be rejected. If gap of 1/8" or greater exists around the perimeter, lead shims and/or lead wool is used to fill in the gaps. The lead wool shall be suitably compressed to remove voids. Each plate with a nominal thickness greater than 3/16" (nominal) shall be ultrasonically inspected for the purpose of verifying that the lead plates are free of volumetric or other defects that could diminish the gamma shielding function. The plates shall be ultrasonically (UT) inspected by the pulse-echo straight-beam direct contact method. The UT testing will take place before the installation of the plates. The UT testing ensures that the plates are uniform internally. This is an accepted industry procedure for locating voids within the lead sheets in order to verify the shielding effectiveness of the sheets.

***<u>NRC Review Comment 6.i</u>***

***Paragraph 8.4.5 of the submittal states that certain ferritic steels in the STC are tested in order to assure that these materials are not subject to brittle fracture failures. The STC references ASME specifications. These specifications provide an option for the purchaser to request brittle fracture testing (impact testing). Identify the specific steel items used in STC that were impact tested (such as the reference numbers on the drawings from Section 1.5 of the submittal) and state the temperature acceptance requirement.***

**Response to 6.i**

Table 6.1 below summarizes the fracture toughness test requirements for the STC components. STC components not explicitly listed in the table below are exempt from impact testing per ND-2311(a).

The STC drawing (Attachment G) has been revised to incorporate the fracture toughness test requirements in the following table.

Table 6.1: Fracture Toughness Test Requirements for the STC components

| **Component Name** | **Item No. (see note 1)** | **Test Requirement** | **Test Temperature (see note 2)** | **Acceptance Criteria** |
|---|---|---|---|---|
| STC Base Plate | 3 | Per ND-2331 | 0°F | Required $C_v$ Lateral Expansion Values per Table ND-2331(a)-1<br>Required $C_v$ Energy Values per Table ND-2331(a)-2 |

| Table 6.1: Fracture Toughness Test Requirements for the STC components | | | | |
|---|---|---|---|---|
| **Component Name** | **Item No. (see note 1)** | **Test Requirement** | **Test Temperature (see note 2)** | **Acceptance Criteria** |
| STC Closure Lid | 4 | Per ND-2331 | 0°F | Required $C_v$ Lateral Expansion Values per Table ND-2331(a)-1 Required $C_v$ Energy Values per Table ND-2331(a)-2 |
| STC Inner Shell | 7 | Per ND-2331 | 0°F | Required $C_v$ Lateral Expansion Values per Table ND-2331(a)-1 Required $C_v$ Energy Values per Table ND-2331(a)-2 |
| STC Outer Shell | 8 | Per ND-2331 | 0°F | Required $C_v$ Lateral Expansion Values per Table ND-2331(a)-1 Required $C_v$ Energy Values per Table ND-2331(a)-2 |
| STC Upper Flange | 9 | Per ND-2331 | 0°F | Required $C_v$ Lateral Expansion Values per Table ND-2331(a)-1 Required $C_v$ Energy Values per Table ND-2331(a)-2 |
| STC Closure Lid Stud | 11 | Per ND-2333 | 0°F | Required $C_v$ Values per Table ND-2333-1 |
| STC Lifting Trunnion | 26 | Per ND-2333 | 0°F | Required $C_v$ Values per Table ND-2333-1 |

Notes:

1. Item numbers are in accordance with the STC drawing (Attachment G)
2. Test temperature shall be less than or equal to the Lowest Service Temperature per ND-2331(a). The Lowest Service Temperature for the HI-TRAC Transfer Cask, which carries the loaded STC, is specified as 0°F in the HI-STORM 100 FSAR (Docket No. 72-1014). Therefore, the Lowest Service Temperature for the STC is also set at 0°F.

*NRC Review Comment 6.j*

***Table 5.4.2 of the submittal lists the STC internal pressure as 53.5 psig in the analysis of the loss of the HI-TRAC jacket water. Paragraph 2.2.3 states that the STC is qualified to withstand a normal internal pressure of 50 psig, and Table 3.2.1 has an accident pressure limit of 65 psig. Paragraph 10.2.3, Procedure Step 23 leak test for the STC is at 55 +5/-0 psig per American National Standards Institute (ANSI) standard N14.5, Section A.5.7 ASME Code, Section III, NB-6000 and ANSI N14.5 have a hold test pressure requirement based on system design. Provide the specific internal test pressure that will be used to ensure that the STC will perform its safety function for any credible abnormal condition and satisfy NB-6000 test pressure requirements.***

**Response to 6.j**

The STC pressure boundary is designed to meet the stress limits of the ASME Code, Section III, Subsection ND. Therefore, pressure testing of the STC is performed to satisfy ND-6000 test pressure requirements. The STC and HI-TRAC will be subjected to a one time pressure test as per ASME Section III, Subsection ND at not less than 125% (hydrostatic test) or 120% (pneumatic test) of Design Pressure. Subsequent testing of the seals (STC and HI-TRAC) will be done at the pressures shown in Chapter 10 of the Licensing Report. See Response 2.f for further details, including the specific internal test pressures.

*NRC Review Comment 6.k*

***Paragraph 8.4.2 of the submittal referenced the drawing in Section 1.5 and applicable codes and standards for weld examinations. The drawings in Section 1.5 do not contain STC weld examination criteria. The STC welds perform containment functions and can experience pressure boundary conditions. Provide the examination methods, acceptance criteria, and volume (methods can be ASME Code, Section III, Subsection NB construction and preservice, or Draft NUREG-1536, Revision 1A, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility)." Volume can be expressed in sketches of the weld cross section area, heat affected zone, and base metal.***

**Response to 6.k**

The STC and STC basket drawings (Attachments G and H) have been revised to clearly identify the applicable code (ASME Section III, Subsection ND, ASME Section III, Subsection NF, etc.) for weld procedures, welder qualification, and weld examination for each STC weld. The licensing drawing has also been revised to show applicable weld geometry for previously undefined groove welds on the pressure boundary.

*__NRC Review Comment 6.l__*

***Section 8.2 of the submittal is void of information on weld material, although weld specifications are included in the references. Provide a discussion on the welding material and on the application of the referenced welding specifications (show the weld specification for specific locations on the applicable drawings).***

**Response to 6.l**

The LAR has several references to AWS welding codes and standards that are not referred to in the body of the LAR. The welding of the STC will be in accordance with ASME B&PV Code Section IX. Welding of ancillary equipment for the STC and HI-TRAC will be in accordance with AWS D.1.1 or ASME Section IX. The licensing drawings provided in Attachments G, H and I have been revised to add a note stating such.

*__NRC Review Comment 6.m__*

***Table 3.1.1 of the submittal has a temperature rating for the STC seal of 248 °F. The closed and sealed STC is placed in the HI-TRAC. All of the HI-TRAC temperatures in Table 3.1.1 exceed the 248°F rating for the STC seals. Provide a method for monitoring STC seal temperature, data supporting seal effectiveness at the higher HI-TRAC temperatures, or other ways of mitigating STC seal temperatures exceeding 248 °F.***

**Response to 6.m**

The temperatures presented in Table 3.1.1 are the temperature limits of the materials that will be used in the components of the fuel transfer operation, including the contents (water and fuel) of the system, not the temperatures that the components will experience. The temperatures for normal conditions of transfer are shown in Table 5.3.1. [PROPRIETARY TEXT REMOVED]

Similarly, the temperatures for accident conditions of transfer are shown in Table 5.4.1. The bounding accident for thermal is the loss of water in the HI-TRAC water jacket (outermost shell used for extra neutron shielding) since this water aids in the heat transfer to the environment. [PROPRIETARY TEXT REMOVED]

*NRC Review Comment 6.n*

***In Section 4.7.6 of the submittal it states, "During manufacturing there is a potential for minor damage to the neutron absorber panels from welding the sheathing to the cell walls. Such damage, up to an area equivalent to 1 inch diameter per panel, was considered in Holtec's HI-STAR Transport SAR [K.C], Section 6.4.11, for various baskets similar to the STC basket, and was found to be acceptable. This condition is therefore also acceptable for the STC, without any further calculations." However, the STC will have a much different operating regime/history than the HI-STAR Transport Cask. Provide an evaluation of the welding damage given the intended operating regime/history of the STC.***

***All of the above information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDCs 1, 4, and 61 and the intent of 10 CFR 72.24, 72.122, 72.140, and 72.234.***

**Response to 6.n**

The above HI-STAR criteria will not be applied to the STC basket fabrication. The Metamic panels must pass a rigorous inspection process prior to being certified by the Metamic manufacturer and a rigorous receipt inspection of the Metamic panels occurs at the fabrication facility prior to their installation. Any deviations which may occur during fabrication of the basket will be evaluated to determine acceptability.

*7. Shielding Analyses*

*NRC Review Comment 7.a*

***Modify the shielding evaluation to address the non-fuel hardware that is to be transported with the IP3 assemblies in the STC.***

***Some sections (e.g., Criticality) of the application enclosure evaluating the fuel transfer indicate that non-fuel hardware (e.g., BPRA, CEA, APSR, WABA, etc.) will be moved with the fuel assemblies. However, this hardware was not accounted for in the shielding evaluation. The evaluation should include the parameters important for shielding, such as the types of non-fuel hardware to be transported, the material specifications and assumed impurity levels (including the basis for the assumption), the maximum equivalent burnup and minimum cooling time for these items covered by the shielding evaluation, and the impacts on dose rates.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).***

**Response to 7.a**

BPRAs were utilized in the calculations to address the impact on the dose rates due to the presence of non-fuel hardware. In general, BPRAs are the bounding non-fuel hardware, except in comparisons to CRAs which tends to provide higher dose rates than the BPRAs in the bottom region of the system. In the HI-STORM Certificate of Compliance 1014, Amendments 2 through 6, the CRAs may only be loaded in the four center positions in the MPC-32. Therefore, the surrounding fuel assemblies will provide shielding of the CRAs in the radial direction. The same requirement will apply to the STC and will therefore ensure that the calculated radial dose rates are bounding for BPRAs. As far as the dose rates in the bottom region is concerned, operators are not present below the STC, so potentially higher dose rates due to CRAs will not pose a concern.

Table 2.1.25 of the HI-STORM 100 FSAR lists the allowable burnups and cooling times for non-fuel hardware that corresponds to the BPRA. These values correspond to a Co-60 source and decay heat from the BPRA of 895 curies Co-60 and 14.4 watts for each BPRA. The cobalt-59 impurity level in the analysis was assumed to be 1 g/kg for all non-fuel hardware pieces.

Accounting for BPRAs increases the dose rate in the radial and top axial directions of the STC [PROPRIETARY TEXT REMOVED] while the dose rate in the bottom axial direction is virtually unchanged.

Accounting for BPRAs in the HI-TRAC (containing the STC) proved to only increase the dose rates slightly in the top axial direction [PROPRIETARY TEXT REMOVED] while the dose rates in the radial and bottom axial directions remained unchanged.

***NRC Review Comment 7.b***

***Provide the following fuel specifications: maximum uranium mass loading of the assembly types and the size/length of axial blankets.***

***The shielding evaluation should address these fuel specifications. The maximum mass loading is a main driver in determining the assembly type that is used as the design basis assembly for shielding evaluations. Axial blankets of sufficient length can have a noticeable effect on source term and thus dose rate. The shielding evaluation should appropriately address these fuel characteristics.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).***

**Response to 7.b**

The mass loading of the design basis fuel used in the shielding evaluations, B&W 15x15, is 495.5 kg of U (562.0 kg of $UO_2$). This is a maximum weight that bounds all of Indian Point fuel types.

While Indian Point utilizes axial blankets, they are not included in the shielding model. The reason for not including them is that they are not expected to impact the shielding evaluations because of the low burnup achieved in the top and bottom regions of the fuel assembly. The analysis utilizes a generic burnup profile (taken from the HI-STORM FSAR), which allows for a higher burnup at the ends than axial blankets would provide. In addition, the axial height of the axial blankets is a small portion (about 6-8 in.) in comparisons to the total active fuel length and will therefore not impact the dose rate values significantly. Furthermore, the burnup from the center region of the fuel is the 'driver' due to its peak burnup value.

*NRC Review Comment 7.c*

***Provide acceptance tests for the STC shielding to ensure the as-fabricated shielding features will perform as designed.***

***The application (see enclosures 1 and 2) does not describe any acceptance testing of the design features relied upon for shielding in the STC. One of those features is the lead between the steel inner and outer shells of the STC body. Also, it is not clear how this lead is placed in the STC (i.e., poured or installing of pre-cast sections). The fabrication of the lead shielding should be described, including how development of voids or gaps in the shielding will be precluded. The staff notes that descriptions regarding fabrication of the shielding and testing of its effectiveness are described for the HI-TRAC transfer casks in the HI-STORM 100 FSAR and these fabrication descriptions and effectiveness tests have been found to be acceptable in the licensing activities for the HI-STORM 100 system. Also, there are no acceptance tests described for the Metamic neutron absorber plates. While a significant criticality safety design feature, the licensee's shielding analysis also relies upon the Metamic plates for the shielding design. Thus, the application should describe an acceptance testing program for ensuring the plates perform as designed. Or, the licensee could quantify the effect of the Metamic plates on dose rates and show the effect of the plates is negligible and, therefore, for the purposes of the shielding design only, an acceptance testing program is not needed.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104.***

**Response to 7.c**

See Response 6.h for the shielding effectiveness of the lead sheets.

See Response 3.c for the qualification of Metamic for the intended function. Installation of neutron absorber panels into the fuel basket shall be performed in accordance with written and approved procedures. Travelers and quality control procedures shall be in place to assure each required cell wall of the STC basket contains a neutron absorber panel in accordance with the STC basket drawing in Attachment H. These quality control processes, in conjunction with in-process manufacturing testing, provide the necessary assurances that the neutron absorber will perform its intended function. The Metamic neutron absorber does not have a significant effect on shielding. No additional testing or in-service monitoring of the neutron absorber material will be required.

*NRC Review Comment 7.d*

***Provide a conservative evaluation of off-normal conditions and the resulting doses.***

***The application does not provide an evaluation of off-normal conditions and the doses that would result from such scenarios. Instead, the application leaves this evaluation for later. Since the application is for use of the STC at a single site, Indian Point, the possible scenarios for off-normal conditions should be relatively easy to postulate and a reasonably conservative evaluation can be performed. Further, the licensee is applying the regulatory criteria from 10 CFR 72.104 to the shielding and radiation protection evaluation. Those criteria apply to normal and anticipated occurrences (i.e., off-normal conditions). Thus, to verify that the design meets these criteria, the evaluation needs to address off-normal conditions. The evaluation should include appropriate descriptions of the conditions assumed, including time duration, with appropriate justification, as well as dose estimates (from both direct radiation and effluent release) to show that the 72.104 criteria are met when off-normal conditions are considered.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).***

**Response to 7.d**

An off-normal event (in compliance with 10 CFR 72.104) of 30 days duration (conservatively representing the time it could take to correct the off-normal condition) has been evaluated and added to the calculation package. The dose contribution from an off-normal event when transporting the HI-TRAC between Unit 3 and Unit 2 was found to be 1.4 mrem from direct radiation and 2.2 mrem from the combined dose (direct radiation and effluent dose release). Note that an effluent dose release during normal conditions (as is assumed in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, Washington D.C., January 1977), in which a fraction of fuel rods breach in the STC, is not expected to occur. In addition, both the STC and the HI-TRAC lids will be leak tested.

*__NRC Review Comment 7.e__*

***Include in the shielding evaluation the contributions from the other "uranium fuel cycle operations within the region" as well as the number of fuel transfers that can be performed to demonstrate compliance with the criteria of 10 CFR 72.104(a).***

***The criteria of 10 CFR 72.104(a) includes the contributions from other fuel cycle operations in the region in demonstrating compliance with the 25 mrem annual dose limit. For Indian Point, these operations include the reactor units and associated facilities as well as the operating ISFSI. The evaluation does not include the contributions from these facilities. In addition, the criteria in 10 CFR 72.104(a) are annual dose limits. The evaluation only considers a single transfer under normal conditions. It is anticipated that multiple transfers will occur each year. Thus, an evaluation against the 72.104(a) criteria needs to consider the number of fuel transfers per year as well as anticipated occurrences (see the previous question). The evaluation should clearly show that the contributions of direct radiation and effluents are included in the analysis. Depending upon this evaluation's results, fuel transfers may need to be limited to a set number per year.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).***

**Response to 7.e**

Demonstration of compliance to the dose limit of 25 mrem/yr to a member of the public can be demonstrated for the entire fuel cycle under normal operations (see Table 7.1 below).

| Table 7.1: Total Site Boundary Dose | | |
|---|---|---|
| **Dose Contribution** | **Dose Rate** | **Reference** |
| Site (e.g., operating plant facilities and other site sources such as the Temporary Low Level Storage Building) | 0.43 mrem/500 hrs* (7.5 mrem/8760 hrs) | 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System, Rev. 7, Entergy Nuclear |
| ISFSI (approximately 137 m from the edge of the ISFSI in the west direction) | 17.1 mrem/500 hrs* | HI-2094405, Rev. 0. Dose Versus Distance from a HI-STORM 100S Version B Containing the MPC-32. |
| STC Transfers (24 transfers/yr) | 0.38 mrem/yr | HI-2084109, Rev. 4. Shielding Design Calculation of Transfer Bell for Indian Point 3 |
| **Total** | **17.9 mrem/yr** | |

* The closest site 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) which could result in continuous occupancy, 500 hours per year was used as the occupancy time.

*__NRC Review Comment 7.f__*

***The application needs to include the calculation package(s) for the radiological release analyses during normal and accident conditions. The staff needs to verify that the system is appropriately analyzed and modeled, in order to verify the acceptability of the radiation doses reported in the application.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.104(a).***

**Response to 7.f**

The calculation package (HI-2084109, Rev 4) is provided in Attachment K.

*__8. Miscellaneous__*

*__NRC Review Comment 8.a__*

***The NRC staff requests that the following documents be submitted, as they are needed for the NRC review:***

***i. [L.I.] Holtec International Report HI-2084118, "Shielded Transfer Canister Structural Calculation Package" Latest Revision***

***ii. [U.C.] Holtec International Report HI-2084118, Revision 1, "STC Structural Calculation Package"***

iii. *[U.D.] Holtec International Report HI-2094345, Revision 0, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation"*
iv. *[U.B] IPEC HI-STORM 100 Cask System 72.212 Evaluation Report.*

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in GDC 61 and the intent of 10 CFR 72.212.***

**Response to 8.a**

i. See Attachment N
ii. See Attachment C
iii. See Attachment L
iv. See Attachment M

*NRC Review Comment 8.b*

***The application needs to update each reference to the HI-STORM FSAR. Each reference needs to include a specific citation to the FSAR section, revision number, and date. The safety bases of the application appear to rely in part on the safety bases of design and analytical information in the FSAR. The staff needs to verify that the specific safety bases are applicable to this application and whether the application relies on information changed in the HI-STORM FSAR under the 72.48 process. To facilitate the review, the licensee should provide a table identifying each FSAR reference, applicable regulatory requirements, applicable industry/code requirements, and a description of how compliance with each is attained.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34.***

**Response to 8.b**

The applicable industry/code requirements for the spent fuel transfer equipment are as follows:

Shielded Transfer Canister

The material procurement, design, fabrication, and inspection of the STC are per ASME Section III, Subsection ND (2004 Edition). The material procurement, design, fabrication, and inspection of the STC basket are per ASME Section III, Subsection NG (2004 Edition).

HI-TRAC Transfer Cask

The spent fuel transfer project will utilize the HI-TRAC 100D as licensed by the NRC under Docket 72-1014 and as described in the HI-STORM 100 FSAR. As such, the HI-

TRAC has been designed, fabricated, inspected, and material procured per ASME Section III, Subsection NF (2005 Edition with 1996 and 1997 Addenda) with the trunnions being designed to ANSI N14.6. For the purposes of the spent fuel transfer project the HI-TRAC has also been re-evaluated against the stress limits imposed by ASME Section III, Subsection ND (2007 Edition) including the modified HI-TRAC lid specific to this project.

Table 8.1 is provided to indicate the HI-STORM 100 FSAR Revision and applicable sections for each citation of the HI-STORM 100 FSAR in the licensing report. Table 8.2 is provided to show the compliance with the regulatory requirements.

| Table 8.1: Applicable sections of HI-STORM 100 FSAR | | |
|---|---|---|
| Licensing Report Location of FSAR reference | Subject of the reference | Location in HI-STORM 100 FSAR (Revision 7 except where noted) |
| Page 1-8, Subsection 1.3.2 | Description of HI-TRAC 100D | Subsection 1.2.1.2.3 |
| Page 2-3, Subsection 2.2.6 | Shielding capacity of the HI-TRAC 100D | Section 5.3 |
| Page 3-3, Subsection 3.1.3 | Temperature limits of HI-TRAC materials | Table 2.2.3 |
| Page 3-4, Subsection 3.1.4 | Description of HI-TRAC 100D | Subsection 1.2.1.2.3 |
| Page 3-4, Subsection 3.1.4 | Weight of loaded multi-purpose canister (MPC) | Table 3.2.1 |
| Page 3-14, Table 3.2.3 | Postulated fire event | Subsection 4.6.2.1 |
| Page 6-3, Subsection 6.1.1.2 | Description of HI-TRAC 100D | Subsection 1.2.1.2.3 |
| Page 6-4, Subsection 6.1.2.2 | HI-TRAC material structural properties | Section 3.3 |
| Page 6-13, Subsection 6.2.3 | HI-TRAC 100D trunnions | Subsection 3.4.3.4 |
| Page 6-16, Subsection 6.2.3.4 | Licensing Drawings | Section 1.5, drawings 2145 and 4128 |
| Page 6-19, Subsection 6.2.5 | HI-TRAC g-load limit of 45 g's during handling | Subsection 2.2.3.1 |
| Page 6-20, Subsection 6.2.5 | HI-TRAC g-load limit of 45 g's during handling | Subsection 2.2.3.1 |
| Page 6-22, Subsection 6.2.6 | Stability of free-standing HI-STORM | Subsection 3.4.7.1 |
| Page 6-22, Subsection 6.2.6 | Center of gravity of loaded HI-TRAC | Table 3.2.3 |
| Page 7-2, Section 7.0 | Use of SCALE | Section 5.1 |
| Page 7-2, Section 7.0 | Determination of Design Basis Fuel | Section 5.2 |
| Page 7-5, Subsection 7.2.2 | Use of SCALE and source term determination | Section 5.2 |
| Page 7-5, Subsection 7.2.2 | Cobalt-59 impurity levels and cobalt 60 scaling factors | Subsection 5.2.1 and Table 5.2.10 |
| Page 7-11, Subsection 7.3.1 | Axial distribution of source term based on Axial burnup distribution | Figure 2.1.3 and Table 2.1.11 |
| Page 7-11, Subsection 7.3.2 | Composition and densities of shielding materials | Table 5.3.2 |

| Table 8.1: Applicable sections of HI-STORM 100 FSAR | | |
|---|---|---|
| Licensing Report Location of FSAR reference | Subject of the reference | Location in HI-STORM 100 FSAR (Revision 7 except where noted) |
| Page 7-16, Subsection 7.4.1 | Axial distribution of source term based on Axial burnup distribution | Figure 2.1.3 and Table 2.1.11 |
| Page 7-19, Subsection 7.4.5 | Dose Contribution to Site Boundary | Subsection 5.1.2 |
| Page 7-19, Subsection 7.4.6 | Effluent dose methodology | FSAR Revision 1, Chapter 7 |
| Page 8-1, Section 8.2 | Structural properties of SA 516 Gr 70, SA 515 Gr 70, and SA 36 | Tables 3.3.2 and 3.3.6 |
| Page 8-2, Section 8.2 | Properties and description of Alloy X | Appendix 1.A and Table 3.3.1 |
| Page 8-8, Section 8.3 | Brittle fracture of HI-TRAC materials | Section 3.1.2 |
| Page 8-10, Section 8.4 | Inspection and Acceptance tests of the HI-TRAC | Table 9.1.3 |
| Page 10-3, Subsection 10.1.2 | Maintenance of HI-TRAC 100D | Table 9.2.1 |

***NRC Review Comment 8.c***

***Please include a compliance matrix which identifies the regulatory requirements that apply to this LAR and the specific acceptance criteria for each SAR chapter that is used to demonstrate compliance with the regulatory requirement. This can be a reference to the pages in the LAR.***

***This information is needed for the NRC staff's review to ensure compliance with the criteria contained in 10 CFR 50.34 and the intent of other criteria in 10 CFR Part 72.***

**Response to 8.c**

| Table 8.2: COMPLIANCE MATRIX FOR THE PROPOSED INTER UNIT TRANSFER OPERATION OF THE SHIELDED TRANSFER CANISTER | | |
|---|---|---|
| Applicable 10CFR72 or 10CFR50 Requirement | Applicable Guidance Documents | License Report Section where the requisition is addressed |
| 10 CFR 72.104(a) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ. | | Section 3.1.2, Section 7.1.2, Section 7.4.6 |
| 10 CFR 72.106(b) Accident conditions: Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel, high-level radioactive waste, handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters. | | Section 3.1.2, Section 7.1.2, Section 7.4.6 |

| Table 8.2: COMPLIANCE MATRIX FOR THE PROPOSED INTER UNIT TRANSFER OPERATION OF THE SHIELDED TRANSFER CANISTER | | |
|---|---|---|
| Applicable 10CFR72<br>or 10CFR50 Requirement | Applicable Guidance Documents | License Report Section where the requisition is addressed |
| 10 CFR50. 68 b (2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. | NUREG-0800, Section 9.1.1, Criticality Safety of Fresh and Spent Fuel Storage and Handling. USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis. | Section 3.1.1, Section 4.1, Section 4.3 |
| | SFST-ISG-11 Rev.3: In order to assure integrity of the cladding material, the spent fuel cladding temperatures must be below 400 degree C in accordance with SFST-ISG-11 for short-term operations since no similar regulations/limits exist in 10 CFR Part 50. | Section 3.1.3, Section 5.3.2, Section 5.4 |

| Table 8.2: COMPLIANCE MATRIX FOR THE PROPOSED INTER UNIT TRANSFER OPERATION OF THE SHIELDED TRANSFER CANISTER | | |
|---|---|---|
| Applicable 10CFR72<br>or 10CFR50 Requirement | Applicable Guidance Documents | License Report Section where the requisition is addressed |
| | NUREG-0612, Section 5.1.6: The purpose of the upgrading is to improve the reliability of the handling- system through increased factors of safety and through redundancy or duality in certain active components. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," provides guidance for design, fabrication, installation, and testing of new cranes that are of a high reliability design. For operating plants, Appendix C to this report, "Modification of Existing Cranes," provides guidelines on implementation of NUREG-0554 for operating plants and plants under construction. | Section 1.3,<br>Section 2.2.8,<br>Section 3.1.4,<br>Section 6.1.2,<br>Section 6.2.3,<br>Table10.0.1,<br>Section 10.1.3, |
| | NUREG-0612, ANSI N 14.6:<br>NURG-0612, Section 5.1.6.(1)(a): Special lifting devices that are used for heavy loads in the area where the crane is to be upgraded should meet ANSI N14.6 1978, "Standard For Special lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials," as specified in Section 5.1.1(4) of this report except that the handling device should also comply with Section 6 of ANSI N14.6-1978. | Section 1.3,<br>Section 1.4,<br>Section 2.2.8,<br>Section 3.1.4,<br>Section 6.1.2,<br>Section 6.2.3,<br>Section 10.1.3 |

| Table 8.2: COMPLIANCE MATRIX FOR THE PROPOSED INTER UNIT TRANSFER OPERATION OF THE SHIELDED TRANSFER CANISTER | | |
|---|---|---|
| Applicable 10CFR72<br>or 10CFR50 Requirement | Applicable Guidance Documents | License Report Section where the requisition is addressed |
| 10 CFR50- Appendix A: Criterion 61: Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. | | Section 1.4,<br>Section 2.1,<br>Section 5.0,<br>Section 5.1,<br>Section 7.1.2,<br>Section 8.4,<br>Section 9.1.2. |
| 10 CFR50- Appendix A: Criterion 62: Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations | | Section 3.1.1,<br>Section 4.3,<br>Section 4.5.4,<br>Section 4.7. |

| Table 8.2: COMPLIANCE MATRIX FOR THE PROPOSED INTER UNIT TRANSFER OPERATION OF THE SHIELDED TRANSFER CANISTER | | |
|---|---|---|
| Applicable 10CFR72 or 10CFR50 Requirement | Applicable Guidance Documents | License Report Section where the requisition is addressed |
| | NUREG 0800- Section 15.7.4: Radiological Consequences of fuel handling accidents: The Accident Evaluation Branch acceptance criteria for this Standard Review Plan section are based on requirements of<br>10 CFR Part 100 with respect to the calculated radiological consequences of a fuel handling accident and General Design Criterion 61 with respect to appropriate containment, confinement, and filtering systems. | The acceptance criteria from 10<br>CFR 72 were used rather than 10 CFR 100, since the 10 CFR 72 regulations are more restrictive.<br>The requirements of 10 CFR 72 and GDC-61 are discussed above. |

***B. Questions That May Be Addressed Now Or In Future Staff Requests***

***1. Technical Specification (TS) Changes***

***NRC Review Comment 1.a***

***Please explain how the proposed TS changes meet the requirements of 10 CFR 50.36. Also, it appears that IP3 should have a TS for 2000 ppm boron in the spent fuel pool rather than relying on a statement in the updated FSAR to provide the 2000 ppm requirement.***

**Response to 1.a**

The requested information shall be provided at a later date under a separate cover letter.

***NRC Review Comment 1.b***

***b. There are typographical errors on IP3 TS Bases pages B3.7.18-1 and B3.7.18-2.***

**Response to 1.b**

Revised IP3 TS Bases pages shall be provided at a later date under a separate cover letter.

***2. Administrative Items***

***NRC Review Comment 2.a***

***The information contained in Section 4.5.3 and Table 4.5.3 should not be marked as proprietary information. It is copied from NUREG/CR-6801, which is a publicly available document.***

**Response to 2.a**

A revised non-proprietary copy of the submittal with the above requested Section and Table shall be provided under a separate cover letter.

***3. ALARA***

***NRC Review Comment 3.a***

***Describe the actions performed in the event that measured dose rates exceed the calculated (or "expected") dose rates.***

***Operations descriptions in the Enclosures include operations No. 18 on page 10-9 and No. 33 on page 10-10. These operations are dose rate measurements to check dose rates against calculated (or "expected") dose rates. The operations should also***

***describe actions to be taken in the event that measured dose rates exceed the calculated values.***

**Response to 3.a**

In the event the measured surface dose rates exceed the calculated (or predetermined) dose rate limits, the following guidance will be provided in the administrative procedures:

a. Perform radiological survey on top of the STC lid as it breaks the surface of the water and compare to the expected, or predetermined, dose rates (per step 18 [p. 10-9] of HI-2094289).
b. If dose rates exceed expectations, lower STC back into the pool.
c. Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.
d. 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. If the higher dose rate values are not acceptable, a reload of the STC will be performed.

***<u>NRC Review Comment 3.b</u>***

***Clarify that the estimated occupational exposures estimated in Section 7.4.7 of the enclosures include the dose from effluents.***

***Occupational exposures result from direct radiation from the STC as well as effluents from the STC with the leak rate defined in the application. It is not clear that the occupational exposure estimates include potential contributions from both of these sources.***

**Response to 3.b**

The occupational exposures presented in the calculation package have been updated to also include the effluent dose. The total person-rem exposure increased by approximately 35%, estimating that the combined dose (from direct radiation and effluent dose release) is about 1 person-rem for primary personnel and 0.53 person-rem for secondary personnel. It should be mentioned, however, that an effluent dose release during normal conditions of intact fuel, in which a fraction of fuel rods breach, is not expected to occur. In addition, both the STC and the HI-TRAC lids will be leak tested

***<u>NRC Review Comment 3.c</u>***

***Provide a sample shielding calculation input file.***

***A sample input file can help with the review of the shielding analysis. A sample input file allows the reviewer to quickly understand the model and alleviate the need for***

***questions regarding parts of the analysis that are not clear in the application's description of the analysis model, thus speeding up the review.***

**Response to 3.c**

A sample shielding input file is provided in Attachment P.

*NRC Review Comment 3.d*

***Provide information to demonstrate that compliance will be achieved with the radiation dose limits for individual members of the public as required in 10 CFR 20.1301(a)(1), (a)(2), (b), and (e).***

1. ***Such information may include a map of the proposed transfer route from IP3 to IP2 with approximate nearest distances to Indian Point's controlled area and unrestricted area boundaries.***

2. ***If members of the public are allowed in the controlled area and/or the restricted area, provide information to demonstrate that compliance will be achieved with the 100 mrem dose limit in 20.1301(a)(1) as required in 20.1301(b).***

3. ***Provide a dose assessment to demonstrate that the dose limit of 2 mrem in an hour in the unrestricted area will be met as required in 20.1301(a)(2).***

4. ***Provide a dose assessment to demonstrate that compliance will be achieved with the EPA generally applicable environmental radiation standard for real individuals in the unrestricted area as required in 10 CFR 20.1301(e).***

***All the information above is needed to ensure compliance with the criteria in GDC 61, 10 CFR Part 20, and the intent of 10 CFR 72.104.***

**Response to 3.d.1**

The requested information shall be provided at a later date under a separate cover letter.

**Response to 3.d.2**

The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, as required by 10 CFR 20.1301(a)(1). There are administrative controls in place that limits the members of the public to have access to controlled areas during certain times and events (e.g., STC transfers). Conservatively assuming an occupancy time of 500 hours per year, the estimated dose rate to the public is 45 mrem/year (based on the 10 CFR 20 unrestricted area dose rate of 0.09 mrem/hr).

**Response to 3.d.3**

The dose contribution to the on-site members of the public in the 10 CFR 20 unrestricted area was calculated to be 0.09 mrem/hr. This dose value is based on a conservatively estimated distance of 10 m from the HI-TRAC surface to the 10 CFR 20 unrestricted area. It can be concluded that the dose contribution to the on-site members of the public is well below the regulatory limit of 2 mrem/hr. Furthermore, no credit is taken for shielding from support buildings.

**Response to 3.d.4**

The requested information shall be provided at a later date under a separate cover letter.

*4. Transport Analyses*

*NRC Review Comment 4*

***Please provide information on the structural capacities of the air pads, low profile transporter and VCT and how they compare with the maximum loads placed on them during the proposed spent fuel transfer. Also, provide the maximum height above the ground at which the HI-TRAC 100D transfer cask may be carried on the VCT during transport along the haul path.***

**Response to 4**

VCT

Minimum Lift capacity of the VCT is typically 205 tons (410 kips). The maximum lifted load is the amplified load of the HI-TRAC under earthquake event which is 220 kips.

Air Pads

The total load on the Air-pads is equal to the amplified weight of the loaded HI-TRAC under the influence of the earthquake event which is 220 kips.

LPT (low profile transporter)

Typically the vertical and lateral capacity of the LPT rollers is 150 and 40 metric tons (330.69 and 88.18 kips) respectively and LPT is governed by the capacity of the rollers. [PROPRIETARY TEXT REMOVED]

Attachments:

A: Holtec Report HI-2094429, Rev 0, "Evaluation of Hazards along the Haul Path for Fuel Transfer between IP-3 and IP-2".

B: Drawing 7176, Rev 1, "Bottom Missile Shield; Licensing Drawing".

C: Holtec Report HI-2084118 Rev 1, "Shielded Transfer Canister Structural Calculation Package".

D: Holtec Report HI-2084146 Rev 4, "Thermal Hydraulic Analysis of IP3 Shielded Transfer Canister".

E: Entergy Document No. FCX-00570-00 EC 16693, "Haul Path Evaluation".

F: Holtec Report HI-2032977, "Generic Crawler Specification".

G: Drawing 6013 Rev 7, "Indian Point Unit 3 Shielded Transfer Canister Assembly; Licensing Drawing – General Arrangement".

H: Drawing 6015 Rev 4, "Indian Point Unit 3 Shielded Transfer Canister Basket Assembly; Licensing Drawing – General Arrangement".

I: Drawing 6571 Rev 2, "Indian Point Unit 3 HI-TRAC Transfer Cask Top Lid Assembly; Licensing Drawing – General Assembly".

J: Holtec Report HI-2084176 Rev 2, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Canister".

K: Holtec Report HI-2084109 Rev 4, "Shielded Transfer Canister Shielding Report".

L: Holtec Report HI-2094345, Rev 0, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation".

M: "IPEC HI-STORM 100 Cask System 72.212 Evaluation Report".

N: Holtec Report HI-2084118 Rev 2, "Shielded Transfer Canister Structural Calculation Package".

O: Proposed haul path design for fuel transfer operation.

P: MCNP Sample Input file for STC regionalized loading.

Q: Holtec Report HI-2022847 Rev 5, "Spent Nuclear Fuel Source Terms".

ATTACHMENT A

to

ATTACHMENT 1

**Holtec Report HI-2094429, Rev 0, "Evaluation of Hazards along the Haul Path for Fuel Transfer between IP-3 and IP-2".**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT B

to

ATTACHMENT 1

**Drawing 7176, Rev 1, "Bottom Missile Shield; Licensing Drawing".**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT C

to

ATTACHMENT 1

**Holtec Report HI-2084118 Rev 1, "Shielded Transfer Canister Structural Calculation Package"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT D

to

ATTACHMENT 1

**Holtec Report HI-2084146 Rev 4, "Thermal Hydraulic Analysis of IP3 Shielded Transfer Canister"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT E

to

ATTACHMENT 1

**Entergy Document No. FCX-00570-00 EC 16693, "Haul Path Evaluation"**

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

Entergy

# CALCULATION/ANALYSIS SHEET

| | | |
|---|---|---|
| STATION/UNIT IPEC - UNIT 2 | CALCULATION NO. FCX-00570-00 | PAGE 1 OF 94 |
| PREPARER/DATE: LilianaKandic/04/20/04 | REVIEWER/DATE: Dave Rollins | CLASS |
| SUBJECT OF COMPUTATION: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths | | MOD NO./ PROJ. NO. ER-04-2-053/IP2-03-21444 |

## CALCULATION COVER PAGE

☒ IP-2 ☐ IP-3 ☐ JAF ☐ PNPS ☐ VY

| | | |
|---|---|---|
| Calculation No. FCX-00570-00 | This revision incorporates the following MERLIN DRNs or Minor Calc Changes: | Sheet 1 of 94 |
| Title: Evaluation of HoltecTransporter IP2 and IP3 Haul Paths | | ☐ QR ☒ NQR |
| Discipline: CIVIL ENGINEERING | | Design Basis Calculation? ☐ Yes ☒ No |

This calculation supercedes/voids calculation: N/A

Modification No./Task No/ER No: ER-04-2-053/IP2-03-21444

☒ No software used
☐ Software used and filed separately (Include Computer Run Summary Sheet). If "YES", Code:
☐ Software used and filed with this calculation. If "YES", Code:

System No./Name:

Component No./Name:

**Print / Sign**

| REV # | STATUS (Prel. Pend, A, V, S) | PREPARER | REVIEWER/DESIGN VERIFIER | OTHER REVIEWER/ DESIGN VERIFIER | APPROVER | DATE |
|---|---|---|---|---|---|---|
| 00 | Pend. | Liliana Kandic Liliana Kandic 04/21/04 | Dave Rollins Dave J Rollins | 5/7/05 Per Telecon | R. Drake | 6/8/06 |
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| Entergy | CALCULATION/ANALYSIS SHEET | |
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| SUBJECT OF COMPUTATION: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths | | MOD NO./ PROJ. NO. ER-04-2-053/IP2-03-21444 |

## RECORD OF REVISIONS

**Calculation Number:**

| Revision No. | Description of Change | Reason For Change |
|---|---|---|
| 0 | Initial Issue | N/A |
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## CALCULATION SUMMARY PAGE

Calculation No. FCX-00570-00 Revision No. 00

CALCULATION OBJECTIVE: Evaluate the capacity of existing underground utilities (pipes and underground structures) as well as the road to sustain loads from the fully loaded Transporter when traveling along the travel path toward the Independent Spent Fuel Storage Installation Facility (ISFSI).

CONCLUSIONS: The following underground structures need protection: electrical manhole located along the route from the IP2 FSB toward the present IP2 Security Gate, underground duct buried along the road leading from IP3 to IP2 Security gate, manholes located in front of the IP2 Security Gate, drain inlets and manhole along the IP2 portion of the road. All are depicted on Figure 14 within the body of the calculation

ASSUMPTIONS: Only simplifying conservative assumptions that do not require verification are used in the calculation. All such assumptions are clearly identified within the calculation and all are made to increase conservatism in the design

DESIGN INPUT DOCUMENTS:

1.1 Indian Point Calculation SGRP-C-003, Rev. 4 (Reference 6.2.1)
1.2 IP2 drawings listed in Reference Section 6.2
1.3 IP3 drawings listed in Attachment A to this calculation
1.4 Indian Point Drawing 9321-F-1004: Plan of Entrance Roads, Units No. 1, 2 and 3 (Reference 6.2.5)

AFFECTED DOCUMENTS:
N/A

METHODOLOGY: Methodology of the evaluation underground utilities, structures as well as the road conforms to the specific structural requirements, all of which are detailed in Section 8 of the body of the calculation.

| Entergy | CALCULATION/ANALYSIS SHEET | |
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| STATION/UNIT IPEC - UNIT 2 | CALCULATION NO. FCX-00570-00 | PAGE 4 OF 94 |
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## DESIGN VERIFICATION COVER PAGE

☒ IP-2 ☐ IP-3 ☐ JAF ☐ PNPS ☐ VY

| Document No. FCX-00570-00 | Revision 0 | Page 1 of 7 |
|---|---|---|
| Title: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths | | |

☐ Quality Related ☒ Non Quality Related

**DV Method:** ☒ ***Design Review*** ☐ ***Alternate Calculation*** ☐ ***Qualification Testing***

| VERIFICATION REQUIRED | DISCIPLINE | VERIFICATION COMPLETE AND COMMENTS RESOLVED (DV print, sign, and date) |
|---|---|---|
| ☐ | Electrical | |
| ☐ | Mechanical | |
| ☐ | Instrument and Control | |
| ☒ | Civil/Structural | DAVID J ROLLINS 4-20-04<br>David J Rollins 5/17/05 per Telecon C. Evers |
| ☐ | | |
| ☐ | | |

Print/Sign After Comments Have Been Resolved

| Originator: Liliana Kandic *Liliana Kandic* | Date: 04/20/04 |
|---|---|

Entergy

# CALCULATION/ANALYSIS SHEET

| | | |
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| SUBJECT OF COMPUTATION: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths | | MOD NO./ PROJ. NO. ER-04-2-053/IP2-03-21444 |

## DESIGN VERIFICATION CHECKLIST

IDENTIFICATION:

Document Title: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths

Doc. No.: FCX-00570-00 Rev. 00 QA Cat.

Verifier: Dave Rollins (Print) Dave J Rollins (Sign) 4-20-04 (Date)

Manager authorization for supervisor performing verification.

☒ N/A

5/17/05 Per Telecon C F Evers

Print Sign Date

**DISCIPLINE:**

☒ Civil/Structural
☐ Electrical
☐ I & C
☐ Mechanical
☐ Other

METHOD OF VERIFICATION:

Design Review ☒ Alternate Calculations ☐ Qualification Test ☐

**1. Design Inputs** – Were the inputs correctly selected and incorporated into the design?

Design inputs include design bases, plant operational conditions, performance requirements, regulatory requirements and commitments, codes, standards, field data, etc. All information used as design inputs should have been reviewed and approved by the responsible design organization, as applicable.

*All inputs need to be retrievable or excerpts of documents used should be attached.*

See site specific design input procedures for guidance in identifying inputs.

Yes ☒ No ☐ N/A ☐

**Reference**

Page No. ____

**OR**

Paragraph No. 7

*Completion of the Reference Boxes is optional for all questions.*

**Verifier Comments:**

**Resolution:**

**2. Assumptions –** Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are assumptions identified for subsequent re-verification when the detailed activities are completed?

Yes ☒ No ☐ N/A ☐

**Reference**
Page No. ______
**OR**
Paragraph No. 5

**Verifier Comments:** Assumptions are minimal and conservative.

**Resolution: N/A**

**3. Quality Assurance –** Are the appropriate quality and quality assurance requirements specified?

Yes ☐ No ☐ N/A ☒

**Reference**
Page No. ______
**OR**
Paragraph No. ______

**Verifier Comments:**

**Resolution:** N/A

Entergy

# CALCULATION/ANALYSIS SHEET

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| | |
|---|---|
| **4. Codes, Standards and Regulatory Requirements** – Are the applicable codes, standards and regulatory requirements, including issue and addenda properly identified and are their requirements for design met?<br><br>Yes ☒ No ☐ N/A ☐<br><br>**Verifier Comments:**<br><br>**Resolution:** | **Reference**<br>Page No. ______<br>**OR**<br>Paragraph No. ______ |
| **5. Construction and Operating Experience** – Have applicable construction and operating experience been considered?<br><br>Yes ☐ No ☐ N/A ☒<br><br>**Verifier Comments:**<br><br>**Resolution:** N/A | **Reference**<br>Page No. ______<br>**OR**<br>Paragraph No. ______ |
| **6. Interfaces** – Have the design interface requirements been satisfied and documented?<br><br>Yes ☐ No ☐ N/A ☒<br><br>**Verifier Comments:**<br><br>**Resolution:** | **Reference**<br>Page No. N/A<br>**OR**<br>Paragraph No. N/A |

Entergy

# CALCULATION/ANALYSIS SHEET

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**7. Methods** – Was an appropriate analytical method used?

Yes ☒ No ☐ N/A ☐

**Reference**
Page No. ____________
**OR**
Paragraph No. ____________

**Verifier Comments:**

**Resolution:** N/A

**8. Design Outputs** – Is the output reasonable compared to the inputs?

Yes ☒ No ☐ N/A ☐

**Reference**
Page No. throughout calc.
**OR**
Paragraph No. ____________

**Verifier Comments:**

**Resolution:**

**9. Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that design requirements have been satisfactorily accomplished?

Yes ☒ No ☐ N/A ☐

**Reference**
Page No. ____________
**OR**
Paragraph No. ____________

**Verifier Comments:**

**Resolution:** N/A

**10.** **Records and Documentation –** Are requirements for record preparation, review, approval, retention, etc., adequately specified?

Are all documents prepared in a clear legible manner suitable for microfilming and/or other documentation storage method? Have all impacted documents been identified for update?

Yes ☒ No ☐ N/A ☐

**Reference**
Page No. __________
**OR**
Paragraph No. __________

**Verifier Comments:**

**Resolution:**

**11.** **Software Quality Assurance-** For a calculation that utilized software applications (e.g., GOTHIC, SYMCORD), was it properly verified and validated in accordance with ENN IT-104 or previous site SQA Program?

Yes ☐ No ☐ N/A ☒

**Reference**
Page No. __________
**OR**
Paragraph No. __________

**Verifier Comments:**

**Resolution:** N/A

Entergy

# CALCULATION/ANALYSIS SHEET

| | | |
|---|---|---|
| STATION/UNIT **IPEC - UNIT 2** | CALCULATION NO. FCX-00570-00 |  |
| PREPARER/DATE: LilianaKandic/04/20/04 | REVIEWER/DATE: Dave Rollins | CLASS |
| SUBJECT OF COMPUTATION: Evaluation of Holtec Transporter IP2 and IP3 Haul Paths | | MOD NO./ PROJ. NO. ER-04-2-053/IP2-03-21444 |

OTHER COMMENTS

RESOLUTIONS

**All comments for "NO" answers have been resolved satisfactorily.**

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## LIST OF FIGURES

## LIST OF EFFECTIVE PAGES

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| 1-94 | 0 |

## 1.0 BACKGROUND

Spent fuel will be off-loaded from the Fuel Storage Building's spent fuel pool of IP2 and IP3, placed in storage containers and transported to the Independent Spent Fuel Storage Installation (ISFSI) facility. The fuel will be transported to the ISFSI on a Self-Propelled Cask Transporter (further in the text referred as Transporter) (Reference 6.1.16). The Transporter with fuel canisters (HI-STORM 100S Cask, Ref. 6.1.16) *will travel along the defined haul route as shown on sketch,* **Figure 14** *(Section 9.0) of this* calculation. This haul route will cross over various existing underground utilities and structures.

Numerous utilities and structures have been analyzed for the loads imposed due to process of replacing the Steam Generator (Calc. # SGRP-C-003, Ref. 6.2.1). In addition, condition of the affected utilities due to the load from Prime Mover and Transporter specified in Westinghouse calculation IP2-602-0302-001 (found as Attachment A to Ref. 6.2.1) has been previously analyzed. However, the haul path for the fully loaded Transporter is not the same as the haul path for the Steam Generator Replacement Project. With these facts in mind, condition of the undergrounds due to the crossing of the Transporter loaded with HI-STORM 100S Cask will be assessed.

For utilities and structures that were previously analyzed and were found to be affected by the fully loaded Prime Mover and Transporter, the load from the Transporter will be assessed and compared with the condition previously analyzed. Buried utilities in the IP2 Fuel Storage Building Alley (FSB) are not in the scope of this analysis.

Since the irradiated load will be off-loaded from the spent fuel pool of IP3 and transported to the ISFSI, buried utilities in the IP3 Fuel Storage Building Alley (FSB) will be exposed to the loads from the fully loaded Transporter. IP3 undergrounds that will be affected by the fully loaded Transporter will be identified and assessed for the applied external load. Identification of the utilities is based on IP3 drawings listed in **Table I** (Section 5.0) of this calculation. Identified utilities are evaluated for the applied external load of the fully loaded Transporter.

## 2.0 PURPOSE

The purpose of this calculation is to:

1) Evaluate the capacity of existing underground utilities (pipes and underground structures) to sustain loads from the fully loaded Transporter when traveling along the travel path. This implies the following:

- The affected utilities and structures analyzed in Ref. 6.2.1 will be assessed for the updated load condition
- The utilities in the IP3 FSB alley and along the IP3 haul road that will be affected by this load will be identified and analyzed.

2) Evaluate the haul road capacity to sustain loads from the fully loaded Transporter.

## 3.0 METHOD OF ANALYSIS

As stated in Section 1.0 of this calculation, utilities and structures within the Steam Generator Replacement Project haul path have previously been identified and assessed for the applicable load due to the Replacement of the Steam Generator.
The utilities and structures that are located along the Transporter haul path will be evaluated in the following way:
First, the applicable load due to the fully loaded Transporter (bearing pressure is 50 psi, as per Ref. 6.1.16) will be compared to the load that was utilized in the course of the previous analysis.
Reference 6.2.1 contains evaluation of the underground utilities for the pressure imposed by the concentrated force of P =13130 lb and P = 14443 lb. The pressure intensity at various depths due to this load is implicitly contained in the applied formula (Ref. 6.2.1, pg. 9-11). This is because the actual intensity of a vehicular live load along with the effect due to the pipe deformation is taken into account. Hence, in order to compare two loads, the pressure intensity at various depths for both loads: vehicular load (13.13 kip) and the 50 psi (fully loaded Transporter) will be found by the same methodology, compared and thus obtained conclusion about applicability of the previous analysis.
If the load due to the fully loaded Transporter is greater than the one analyzed in the past, the utilities will be evaluated for this larger load.

Utilities and structures that have not been analyzed before, the ones located in the IP3 FSB alley and along the haul road, will be analyzed on a case by case basis if their condition can not be enveloped by the previously analyzed condition.

For the purpose of this analysis, the underground utilities are grouped into the following categories:

Group 1: Underground pipes
Group 2: Underground electrical ducts
Group 3: Other

The methodology used to evaluate the respective item is described in the following sections for each group.

### 3.1 Underground pipes

Based on Specification No. 9321-01-44-1, Rev. 1: Specification for Yard Storm Drainage and Yard and Building Standpipe Fire Protection System (Ref. 6.2.11), the following pipe material are utilized: Carbon Steel Pipes, Cast Iron Pipes, Corrugated Metal Pipes. Further, Section defining Cast Iron pipes are specified as either AWWA Class 150 or extra-heavy cast iron soil pipe. Pipes AWWA Class 150 are ductile iron pipes (Ref. 6.1.1, Table 5.23).

Due to different requirements for determining pipe capacity, the underground pipes have been further categorized into the classification listed below. This classification is based on the pipe material and stiffness.

- Cast Iron pipes
- Ductile Iron pipes
- Corrugated Metal Pipes (CMP)
- Rigid galvanized steel pipes
- Flexible steel pipes
- PVC pipes

Evaluation of the underground pipes within the haul path is conducted in a following manner: consistent with the previous calculations (Calc. # SGRP-C-003, Ref. 6.2.1 and its Attachment A: Westinghouse Calc. IP2-002-0302-001) performed for assessment of underground utilities at Indian Point, all underground pipes are categorized as pressurized and not pressurized.

Methodology for small (diameter $D \leq 6"$) nonpressurized pipes (conduits) is basically to assess the maximum external pressure that the conduit can sustain and to compare it with the existing external load.

The specific methodology for the evaluation of pipes in each category is discussed further in greater detail in Section 8.

### 3.2 Underground electrical ducts

The methodology for evaluating underground electrical ducts is based on the theory of beams on elastic foundations as described in Beams on Elastic Foundations (Ref. 6.1.4). The cross-sectional forces have been computed for the line load due to the maximum force acting along the duct. For the concrete ducts without reinforcement, the concrete tension stress has been determined and compared to the limit expressed by the concrete Modulus of rupture $f_r$. A lower limiting value for $f_r$ based on information contained in ACI 318 (Reference 6.3.1) has been conservatively applied. In addition, the guidance provided in Technical Manual AFM 88-6 (Reference 6.1.15, Table 1) has been implemented. For ducts that are reinforced, capacity of the duct is based on the design requirements contained in ACI 318 (Reference 6.3.1).

### 3.3 Other

The evaluation is based on the assessment of the actual loads placed over the items by the fully loaded Transporter. Stresses are computed by elastic analysis method and compared against the limits specified by the respective specifications.

## 4.0 ASSUMPTIONS

### 4.1 Loads

The configuration of the Transporter used to transport spent fuel containers to the ISFSI area is provided in 4 Point Lift System Drawing (Reference 6.1.16). As indicated in this drawing, the fully loaded Transporter weighs a maximum of 577,000 pounds and exerts an average ground bearing pressure of 50 psi. The dimensions of one contact surface are: width w = 2'-5 1/2" and the length l = 16'-5 3/16" (Ref. 6.1.9). The actual surface over which the load is transmitted to the road is a series of grouser plates. The number and dimensions of these plates are not known, so the effective surface that transfers the load is not exactly equal to the width x length area. However, the distance between the grouser plates is small (based on Holtec Transporter used in the Pacific Gas and Electric Diablo Canyon and Humbolt Bay plants it is about 4.5 inches). Since the load spreads at an angle less than 45°, at any depth more than about 2.5 inches below the surface, full width and the full length of the Transporter contact surface will be utilized as the effective width and the effective length of the bearing surface.

Load distribution beneath the Transporter is calculated per the guidance of AASHTO for HS20 truckload distribution (Ref. 6.1.2, page. 4-38 to 4-41 and Ref. 6.1.10). It is based on the maximum value for *average* pressure intensity occurring at various depths beneath the road surface. The AASHTO method calculates the load at different depths by taking into account the fact that the load spreads out in both directions as it travels through the soil. The load at various depths is directly dependent on the area over which the load is spread. The ASSHTO approach described in Reference 6.1.10 assumes that the footprint of the load increases at a rate equal to the value of 1.75/2 times the depth, an angle of 41.18 degrees (tan $\alpha$ =1.75/2 = 0.875).
In addition to the AASHTO approach described above, the load distribution beneath the Transporter is calculated per Boussinesq solution integrated by Hall and Newmark, presented in Ref. 6.1.1, eq. 2-14.

The more conservative load found by the application of these two approaches will be applied in the process of evaluation of capacity of the utilities and structures exposed to the loads from the fully loaded Transporter.

### 4.2 Impact Factor

In assessing underground utilities beneath a road surface, the applied load must be increased by an impact factor to account for the dynamic nature of the load application. That is, the speed at which vehicles travel causes the load on the road surface to increase. However, the Transporter used to transport the spent fuel containers travels at a very slow speed. In addition, only one Transporter traverses the haul path at any time. Due to these conditions, the impact factor will be very small and is assumed to be negligible for the purpose of assessing the underground utilities along the haul path. Note that the Concrete Pipe Handbook (Reference 6.1.2, pg. 4-40) states that even for regular highway traffic, the impact factor is zero for any utilities more than 3'-0" below the surface of the road.

### 4.3 Additional Assumptions

In addition to the assumptions described in the previous sections, certain other conservative assumptions are made throughout the analysis. These assumptions are made for the purpose of simplifying the analysis and are clearly identified within the body of the calculation. Since all of these assumptions are conservative, they do not require confirmation and there are no open assumptions made within the calculation.

## 5.0 INPUT AND DESIGN CRITERIA

The following documents were the source of information used as a basis of the evaluation:

1.5 Indian Point Calculation SGRP-C-003, Rev. 4 (Reference 6.2.1)
1.6 IP2 drawings listed in Reference Section 6.2
1.7 IP3 drawings listed in Attachment A to this calculation
1.8 Indian Point Drawing 9321-F-1004: Plan of Entrance Roads, Units No. 1, 2 and 3 (Reference 6.2.5)

**Table I** defines all of the utilities that have been identified at the IP3 section of the Haul Path. All of the underground pipes are enveloped by the pipes analyzed for IP2 section of the Haul Path.

| Reference Document | Item description | Depth below grade | Note |
|---|---|---|---|
| | | | |
| Dwg 9321-F-41823 | 8"FP | 5' | Enveloped by IP2 utilities evaluation |
| | 2" DW-15 | | " |
| | 1" HY | | " |
| | 2" PWG | | " |
| | 1 1/2" AA | | " |
| Dwg 9321-F-41853 | 8" storm drain | 5' | Enveloped by IP2 utilities evaluation |
| | 4" FP | 4'-7 9/16" | " |
| | MH#12 | | " |
| Dwg 9321-F-41813 | 10" FP | 4'-6" | Enveloped by IP2 utilities evaluation |
| | 24"CMP | | " |
| Dwg 9321-F-13603 | MH #12 | | Enveloped by IP2 utilities evaluation |
| Dwg 9321-F-40053 | MH #12 | | |
| | MH #13 | | |
| | MH #14 | | |
| | 12" CMP | | Enveloped by IP2 utilities evaluation |
| | 18"CMP | | " |
| | 8"CMP | | " |

**Table I**

## 6.0 REFERENCES

### 6.1 Industry Literature

6.1.1 Buried Pipe Design, A.P. Moser, McGraw -Hill, 1990
6.1.2 Concrete Pipe Handbook, American Concrete Pipe Association, 1981
6.1.3 Handbook of Steel Drainage and Highway Construction Products, AISI, 1967
6.1.4 Beams on Elastic Foundation, M. Hetenyi, the University of Michigan, 1946
6.1.5 Design of Concrete Structures, A. Nilson and G. Winter, McGraw-Hill, eleventh edition
6.1.6 Design and Construction of Sanitary and Storm Sewers, ASCE Manual on Engineering Practice, No. 37, WPCF Manual of Practice No. 9
6.1.7 Cast Iron Pipe Institute Publication "Recommendations for Deep Burial of Cast Iron Soil Pipe", 1983
6.1.8 Roark's Formulas for Stress and Strain, McGraw-Hill, sixth edition
6.1.9 Advanced Soil Mechanics, Braja M.Das, Hemisphere Publishing Corporation, 1983
6.1.10 ASTM C 857-95: Standard Practice for Minimum Structural Design Loading for Underground Precast Concrete Utility Structures
6.1.11 STM A120-65: Standard Specification for Black and Hot-Dipped Zinc-Coated (Galvanized) Welded and Seamless Steel Pipe for Ordinary uses
6.1.12 ASTM A53-65: Standard Specification for Welded and Seamless Steel Pipe
6.1.13 ASTM A106-65: Standard Specification for Seamless Carbon Steel Pipe for High Temperature Service
6.1.14 ASTM D1785-646-65: Standard Specification for Poly (Vinil Chloride)(PVC) Plastic Pipe, Schedule 40,80 and 120
6.1.15 Standard Practice for Concrete Pavements, Department of the Army and the Air Force Technical Manual AFM 88-6, Chap.8
6.1.16 4 Point Lift Systems drawing: Lift System Crawler Mounted Self-Propelled Cask Transporter 210 Payload Capacity.
6.1.17 Department of the Army Corps of Engineers: Engineer Manual EM 1110-3-138, April 1984: Pavement Criteria for Seasonal Frost Conditions, Mobilization Construction
6.1.18 Department of the Army Corps of Engineers: Engineer Manual EM 1110-3-141, April
6.1.19 Department of the Army and the Air Force: AFM88-3, Chap. 7, October 1983: Soils and Geology Procedures for Foundation Design of Buildings and Other Structures (Except Hydraulic Structures) 1984: Airfield Flexible Pavement, Mobilization Construction
6.1.20 Structural Engineering Handbook, Gaylord and Gaylord, McGraw Hill, Second Edition

### 6.2 Indian Point Documentation

6.2.1 Indian Point Calculation SGRP-C-003
6.2.2 Drawing Number: A239208-5: Installation of Conduits for 13.8/.480 KV Substation Outside V.C. Hatch
6.2.3 Drawing Number: 9321-H-4016-2: Yard Piping, Potable Water piping for Temporary construction Bldg.
6.2.4 Drawing Number: A217469-2: Maintenance and Outage Bldg. Septic System Plans, Sections and Details
6.2.5 Drawing No. 9321-F-1004-1: Plan of Entrance Roads, Units No. 1, 2 &3"
6.2.6 Drawing Number: A207621-8: Location of Underground Runs Bet. Indian Pt. & Buchanan, Plan
6.2.7 Drawing Number: A207622-1: Location of Underground Runs Bet. Indian Pt. & Buchanan, Sections & Details
6.2.8 Drawing Number: A218 486-07
6.2.9 Drawing No. 9321-F-40143: Yard Storm Drains Sections and Details
6.2.10 Drawing No. WP-2090-001, Rev. 4: Steam Generator Haul Route and Upgrades
6.2.11 Specification No. 9321-01-44-1, Rev. 1: Specification for Yard Storm Drainage and Yard and Building Standpipe Fire Protection System

### 6.3 Design Codes

6.3.1 ACI 318-83
6.3.2 Steel Construction Manual, AISC, eighth edition

### 6.4 Correspondence

6.4.1 Eric G. Lewis (Holtec) e-mail to Geoffrey Schwartz( Indian Point) dated May 1, 2003: Subject: Stack-up Heights and Weights

## 7.0 AFFECTED DRAWINGS

N.A.

# 8.0 CALCULATION

## 8.1 TRANSPORTER LOAD DISTRIBUTION

Distribution of loads from the fully loaded Transporter is determined in the following ways:

Based on the guidance of AASHTO for HS20 truck load distribution as described in the Concrete Pipe Handbook (Reference 6.1.2, pages 4-39 to 4-41) and ASTM C 857-95 (Reference 6.1.10).

As shown in **Figure 1**, at some point below the surface, depending on the distance between loads, the distributed loads overlap. However, consistent with the procedure described in the Concrete Pipe Handbook (Reference 6.1.2, pages 4-39 to 4-41), the *average* pressure intensity at the elevation of the outside top of the pipe is calculated.

Area over which load is spread:

As stated in Section 4.1, the wheel load is spread over an area that is l = 197" by w = 29.5". The bearing areas are 155" apart (**Figure 1**). The effective area over which the load is spread is determined using the following equations. Note that the equation varies with depth, due to overlap of the distributed load areas from the two track surfaces. See **Figure 1** for further explanation and description.

From the surface to a depth $h_1$:

$$h_1 = \frac{155.0''/2}{\tan\alpha} = \frac{77.5}{0.875} = 88.57 \text{ in} = 7.38 \text{ f}$$

below the surface, where x is the depth of interest, the area over which the load is spread is:

$A_1 = (197+1.75H)(29.5+ 1.75H)$, where H is any depth that is less or equal to $h_1$.

For any depth H that is H > 88.57 in, we have for the area over which the load is spread:

$A_2 = (214+1.75H)(197+1.75H)$.

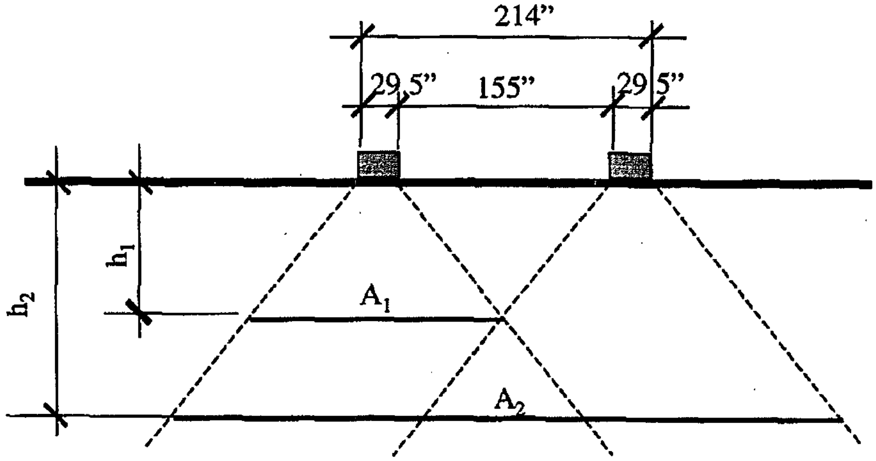


**Figure 1-** Transverse Load Distribution

For various depths the pressure imposed by the fully loaded transporter is found as:

$H \leq h_1$: $p_1 = \dfrac{L/2}{A_1} = \dfrac{288500}{A_1}$ psi

$H > h_1$: $p_2 = \dfrac{L/2}{A_2} = \dfrac{288500}{A_2}$ psi

For 1' increments of depth H, the pressure intensity due to the fully loaded Transporter is tabulated in **Table 1**:

| H (ft) | h (in) | l (in) | w (in) | A ($in^2$) | L (lb) | p = L/A ($lb/in^2$) | p = L/A ($kip/ft^2$) |
|---|---|---|---|---|---|---|---|
| 1 | 12 | 218 | 50.5 | 11009 | 288500 | 26.20 | 3.77 |
| 2 | 24 | 239 | 71.5 | 17088.5 | 288500 | 16.88 | 2.43 |
| 3 | 36 | 260 | 92.5 | 24050 | 288500 | 11.99 | 1.72 |
| 4 | 48 | 281 | 113.5 | 31893.5 | 288500 | 9.04 | 1.30 |
| 5 | 60 | 302 | 134.5 | 40619 | 288500 | 7.10 | 1.02 |
| 6 | 72 | 323 | 155.5 | 50226.5 | 288500 | 5.74 | 0.82 |
| 7 | 84 | 344 | 176.5 | 60716 | 288500 | 4.75 | 0.68 |
| **7.38** | **88.56** | **351.98** | **368.98** | 129873.58 | 577000 | 4.44 | 0.63 |
| 8 | 96 | 365 | 382 | 139430 | 577000 | 4.13 | 0.59 |
| 9 | 108 | 386 | 403 | 155558 | 577000 | 3.70 | 0.53 |
| 10 | 120 | 407 | 424 | 172568 | 577000 | 3.34 | 0.48 |
| 11 | 132 | 428 | 445 | 190460 | 577000 | 3.02 | 0.43 |
| 12 | 144 | 449 | 466 | 209234 | 577000 | 2.75 | 0.39 |
| 13 | 156 | 470 | 487 | 228890 | 577000 | 2.52 | 0.36 |
| 14 | 168 | 491 | 508 | 249428 | 577000 | 2.31 | 0.33 |
| 15 | 180 | 512 | 529 | 270848 | 577000 | 2.13 | 0.30 |
| 16 | 192 | 533 | 550 | 293150 | 577000 | 1.96 | 0.28 |
| 17 | 204 | 554 | 571 | 316334 | 577000 | 1.82 | 0.26 |
| 18 | 216 | 575 | 592 | 340400 | 577000 | 1.69 | 0.24 |
| 19 | 228 | 596 | 613 | 365348 | 577000 | 1.57 | 0.22 |
| 20 | 240 | 617 | 634 | 391178 | 577000 | 1.47 | 0.21 |
| 21 | 252 | 638 | 655 | 417890 | 577000 | 1.38 | 0.19 |

**Table 1** - Live load due to the fully loaded Transporter, at depth H

B) Pressure calculation based on Hall and Newmark integrated Bousinesq method: (Ref. 6.1.1, eq. 2.14):

$W_{sd} = C_s P f' B_c,$

where: $W_{sd}$ = load on pipe, lb/unit length (say ft )

$C_s$ = f (M/H, D/H), where M and D are the length and the width, respectively, of the area over which the distributed load acts.

P = intensity of the distributed load ($lb/ft^2$)

p = 50 psi = $50 x 144\ lb/f^2 = 7200\ lb/ft^2$

$B_c$ = diameter of pipe, (ft)

The load distribution is depicted on **Figure 2**:

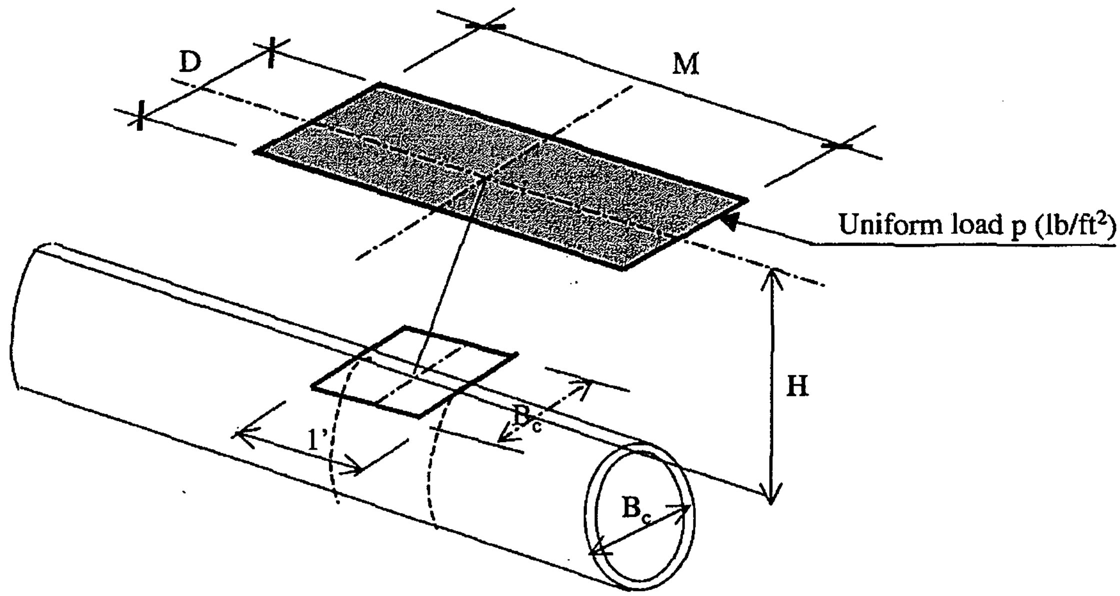


**Figure 2** - Distributed superimposed load vertically centered over conduit

D = 29.5 in, M = 197 in

Table 2.6 of Ref. 6.1.1 provides values of coefficient $C_s$ as a function of depth H:
$C_s = f(D/2H, M/2H)$

F' is the Impact factor. F' = 1.0 as described in Section 4.2.

In order to obtain the load per square foot, we have: $W_{sd} = C_s P f'$.
Consequently, the pressure at various depths will be determined independently of the $B_c$ value.
Tabulation of pressures at various depths is presented in **Table 2**:

| | H | M/2H | D/2H | Cs | p | F' | $W_{sd}$ | $W_{sd}$ |
|---|---|---|---|---|---|---|---|---|
| (ft) | (in) | | | | (lb/ft$^2$) | | (lb/ft$^2$) | (kip/ft$^2$) |
| 1 | 12 | 8.21625 | 1.229167 | 0.868 | 7200 | 1 | 6249.60 | 6.25 |
| 1.5 | 18 | 5.4775 | 0.819444 | 0.74 | 7200 | 1 | 5328.00 | 5.33 |
| 2 | 24 | 4.108125 | 0.614583 | 0.62 | 7200 | 1 | 4464.00 | 4.46 |
| **2.5** | **30** | 3.2865 | 0.491667 | 0.54 | 7200 | 1 | 3888.00 | 3.89 |
| 3 | 36 | 2.73875 | 0.409722 | 0.46 | 7200 | 1 | 3312.00 | 3.31 |
| 4 | 48 | 2.054063 | 0.307292 | 0.355 | 7200 | 1 | 2556.00 | 2.56 |
| 5 | 60 | 1.64325 | 0.245833 | 0.3 | 7200 | 1 | 2160.00 | 2.16 |
| 6 | 72 | 1.369375 | 0.204861 | 0.238 | 7200 | 1 | 1713.60 | 1.71 |
| 7 | 84 | 1.17375 | 0.175595 | 0.22 | 14400 | 1 | 3168.00 | 3.17 |
| 7.38 | 88.56 | 1.113313 | 0.166554 | 0.23 | 14400 | 1 | 3312.00 | 3.31 |
| 8 | 96 | 1.027031 | 0.153646 | 0.15 | 14400 | 1 | 2160.00 | 2.16 |
| 9 | 108 | 0.912917 | 0.136574 | 0.108 | 14400 | 1 | 1555.20 | 1.56 |
| 10 | 120 | 0.821625 | 0.122917 | 0.103 | 14400 | 1 | 1483.20 | 1.48 |
| 11 | 132 | 0.746932 | 0.111742 | 0.09 | 14400 | 1 | 1296.00 | 1.30 |
| 12 | 144 | 0.684688 | 0.102431 | 0.08 | 14400 | 1 | 1152.00 | 1.15 |

**Table 2 -** Distributed load by integrated Bousinesq method

Obviously, the loads found by this method are of higher intensity than the loads found by load distribution per angle with $\tan\alpha = 0.875$ (1.75/2). Therefore, these loads (pressure over the conduit) will be utilized in the course of the evaluation of the capacity of the underground utilities to sustain the applied loads.

In order to ensure conservatism this load will be checked with the load obtained from the standard method of the pressure distribution which distributes the load at an angle of 30°. The pressure will be checked at depth h = 2 ft = 24" under the surface:

$l_{24"} = l + 2xh \tan 30° = 197 + 2x24x \tan 30° = 224.7$ in
$w_{24"} = w + 2xh \tan 30° = 29.5 + 2 x24x \tan 30° = 57.2$ in

$A_{24"} = l_{24"} \times w_{24"} = 224.7 \times 57.2 = 12853$ in$^2$

$$p_{24"} = \frac{L/2}{A_{24"}} = \frac{288500}{12853} = 22.44 \text{ psi} = 3.23 \text{ kip/ft}^2 < 4.46 \text{ kip/ft}^2$$

Therefore, loads from **Table 2** will be compared with the load utilized in the evaluation of the utilities exposed to the "Single concentrated load per tire" of 13.13 kip, i. e. load from SPMT for which the utilities have been assessed in Ref. 6.2.1.

The pressure at the same depths under the grade will be found applying the same method and compared with the pressure from the fully loaded Transporter.

The pressure from concentrated force is found as the Hall and Newmark integrated Boussinesq solution. It is presented in the following form: eqn. 2.13 from Ref. 6.1.1:

$$W_{sc} = C_s \left(\frac{PF'}{L}\right)$$

Where $W_{sc}$ = load on pipe, lb/unit length
$P$ = concentrated load, lb
$F'$ = impact factor
$L$ = effective length of conduit (3 feet or less), ft
$C_s$ = load coefficient which is a function of $B_c/(2H)$ and $L/(2H)$, where H = height of fill from top of pipe to ground surface, ft; and $B_c$ = diameter of pipe, ft

**Table 3** contains tabulated values for pressure due to the concentrated load, in a form that is comparable with the pressure from the fully loaded Transporter. The depths at which the pressure was found correspond to the depths of pipes utilized in the course of analysis of Ref. 6.2.1.

| 2" Pipe with OD = 2.38 in, P =13130 lb | | | | | |
|---|---|---|---|---|---|
| H (ft) | $B_c/(2H)$ | $L/(2H)$ | $C_s$ | $W_{sc} = C_s(PF'/L)$ (lb/ft) | pressure $w = 12W_{sc}/OD$ ($lb/ft^2$) |
| 2.00 | 0.050 | 0.750 | 0.10 | 437.67 | 2206.72 |
| 6" Pipe with OD = 6.625 in, P =13130 lb | | | | | |
| H (ft) | $B_c/(2H)$ | $L/(2H)$ | $C_s$ | $W_{sc} = C_s(PF'/L)$ (lb/ft) | pressure $w = 12W_{sc}/OD$ ($lb/ft^2$) |
| 2.00 | 0.135 | 0.750 | 0.11 | 481.43 | 871 |
| 3.00 | 0.090 | 0.500 | 0.08 | 350.13 | 634 |
| 8" Pipe with OD = 8.62 in, P =13130 lb | | | | | |
| H (ft) | $B_c/(2H)$ | $L/(2H)$ | $C_s$ | $W_{sc} = C_s(PF'/L)$ (lb/ft) | pressure $w = 12W_{sc}/OD$ ($lb/ft^2$) |
| 5.00 | 0.072 | 0.300 | 0.05 | 218.83 | 304.64 |
| 10" Pipe with OD = 10.75 in, P =13130 lb | | | | | |
| H (ft) | $B_c/(2H)$ | $L/(2H)$ | $C_s$ | $W_{sc} = C_s(PF'/L)$ (lb/ft) | pressure $w = 12W_{sc}/OD$ ($lb/ft^2$) |
| 7.00 | 0.064 | 0.214 | 0.04 | 175 | 195.4 |

**Table 3** - Pressure from concentrated Force
P = 13130lb (Ref. 6.2.1)

The loads utilized in analysis of Ref. 6.2.1 imply a pressure that is of lower intensity than the loads imposed by the fully loaded Transporter.

Therefore, the utilities will be evaluated on a case by case basis for the average ground bearing pressure of 50 psi as per Ref. 6.1.16.

## 8.2 EVALUATION OF THE UNDERGROUND PIPES

### 8.2.1 Cast Iron Pipes

Based on Ref. 6.2.5, Section III, Cast iron pipes are extra-heavy cast iron soil pipe. The guidance of Cast Iron Soil Pipe Institute publication: "Recommendations for Deep Burial of Cast Iron Soil Pipe" (Ref. 6.1.7) has been followed in the process of the pipe evaluation.

Evaluation of this group of pipes is performed by comparison of the anticipated soil pressure to an equivalent pipe test load. The test load (strength) is a laboratory three-edge bearing test, which actually measures the strength of a pipe cross section: a thin-walled symmetrically loaded ring. To simplify the analysis, the design approach specified in Reference 6.1.7 utilizes the following conservative and reasonable assumptions.

- Ring deformations are small; moreover, the ring is so stiff that ring deflection is generally less than 1 or 2% in typical installations.

- All loads and reactions are vertical. Radial pressures, such as internal pressure (or external hydrostatic pressure, including internal vacuum) are disregarded. This assumption is made because cast iron soil pipes with D/t = 60 can withstand over 100 psi of external hydrostatic pressure and vacuum. 100 psi is equivalent to a depth of cover of 230 feet of water. Clearly, internal hydrostatic pressure is of no concern in typical design.

In addition to checking the ring capacity, the longitudinal bending capacity will also be checked.

All of the cast iron pipes are assessed using the following process:

1. Determine the earth load
2. Determine the live load
3. Select the installation condition
4. Compare the pipe crushing strength with the field imposed conditions, i.e. check the required strength as follows:

$P_w = \dfrac{12W}{D_m}$, (Ref. 6.1.7, Equation 1, pg. 105) where:

$P_w$ = maximum allowable vertical soil pressure (lb/f), listed in Table 2, Chapter VII of Reference 6.1.7.

W = three-edge bearing load at failure (strength per foot of length of pipe) (lb/f), (Table 2, page 103, Reference 6.1.7)

$D_m$ = mean diameter of pipe ( OD-t)

In addition, the pipe crushing strength will be compared to the applied load. That is, the required strength will be checked as follows:

$$\text{Required three-edge bearing strength} = \frac{\text{design load x factor of safety}}{\text{bedding factor}}$$

5. Longitudinal bending condition is checked as per Ref. 6.1.7, Table 4, pg. 112. This table defines the pipes that will fail by ring crushing before the maximum bending load is attained.

The determinations listed above have been performed in a following way:

1. The earth load on a rigid pipe is computed based on Marston's load theory. The basic approach is that for rigid pipe the side fills may be very compressible in relation to the pipe and the pipe carries practically the entire load. The load on a rigid pipe in a trench (ditch) is found as:

   $W_d = C_d \gamma B_d^2$ Ref. 6.1.1, eqn. 2.4

   $B_d$ is the trench width. The minimum required trench width is

   $B_d = 1.25x\ (OD) + 12"$ Ref. 6.1.7, Ch. VI, pg. 93

   OD is found in Table 2, pg. 103, Ref. 6.1.7

   $C_d$ is the coefficient found from Figure 2.2 of Ref. 6.1.1.

Figure 2.2 is a plot of $H/B_d$ versus $C_d$, for various soil types as defined by their ($K\mu'$) values where ($K\mu'$) is a function of the coefficient of internal friction of the fill material. In the body of the calculation, the ($K\mu'$) value is always taken as 0.165, what corresponds to the fill made of sand and gravel.

   $\gamma$ is taken as $\gamma = 120\ lb/ft^3$, as unit weight for wet sand.

2. The live load determination is based on the approach presented in Section 8.1 of this calculation.

3. The Bedding requirement and the Load Factor are based on Handbook of Steel Drainage and Highway Construction Products (Reference 6.1.3). The most conservative installation case is considered. That is, the pipe is supported throughout its length by a hard flat surface. Bedding of Class D (Flat bottom case, Figure 3.4, pg. 45 of Ref. 6.1.1) results in a Bedding Factor of 1.1. (Table 3.2, pg. 44, Ref. 6.1.1).

For the conservative installation case assumed in the analysis, a Safety Factor of 1.25 is included in each equation for P (Ref. 6.1.7, pg. 116 and Table A-10, pg. 117)

As described previously, the Impact Factor is taken as zero for evaluating these pipes. In addition to the characteristics of the load as described before, all pipes in this category are minimum 3'-0" below grade.

To verify that the simplifying assumption that internal pressure is not a concern, the pipe capacity to withstand the internal pressure will be numerically checked. The equation of the load-pressure parabola is:

$w = W\sqrt{\frac{P-p}{P}}$, in which

W = ring test crushing load with no internal pressure (lb/f)
P = bursting pressure with no external load (psi)
P and W are any combination of internal pressure and external load which will just cause fracture of the pipe.
w = external load ( lb/ft), p = internal pressure ($lb/in^2$)

The values of W and P are as follows:

W = the three-edge bearing ring test crushing load

$P = \frac{2St}{D}$ in which:

S = bursting tensile strength (psi), where S denotes the strength of the iron in the pipe an are based on periodic full-length bursting tests and ring tests as specified in ASA Standard A21.6 (AWWA C106). As per ASTM 120 (Ref. 6.1.10) S = 18,000 psi

t = net thickness

D = nominal pipe size (in).

The design value for w is found in the body of the calculation.

The internal pressure that the pipe can withstand is found as: $p = P - \frac{w^2 P}{W^2} = P\left[1-\left(\frac{w}{W}\right)^2\right]$

This approach is supported by ANSI A21.1, Section 1-3.1, which is also included in Chapter 4 of Ref. 6.1.1 (pg. 117).

The cast iron pipes are evaluated in the following section of the calculation using the procedure described above.

8.2.1.1 Nonpressurized Cast Iron Pipes

1. Determine Earth Load:

Conservatively, the Earth Load will be determined for a 6" diameter pipe, for the depth of burial H = 7.0 feet. 7.0 feet is the deepest depth below the grade that has been identified in existing reference material, applicable for the Transporter haul path. It is the maximum depth applied in the course of analysis conducted in Ref. 6.2.1. Therefore, the weight of the earth above the pipe is conservatively assessed. If the pipe is adequate for the load of this magnitude, it definitely can sustain the pressure of soil with less depth.

$\gamma = 120$ lb/ft$^3$, the soil unit weight

OD = 6.5 in (Ref. 6.1.7, Chapter VII, Table 2)

$B_d = 1.25 \times OD + 12" = 1.25 \times 6.5 + 12 = 20.125$ in

$\frac{H}{B_d} = \frac{7.0' \times 12}{20.125} = 4.17$. From Figure 2.2 of Reference 6.1.1, using soil condition B, $C_d = 2.3$

$W_d = C_d \gamma B_d^2 = 2.3 \times 120 \times (\frac{20.125}{12})^2 =$ 776.28 lb/ft

Earth surface load on the pipe, $p_s = 776.28$ lb/ft /(6.5/12) = 1433.13 lb/ft$^2$, say p = 1.43 kip/ft$^2$

2. Determine Live Load:

For various depths the Live Load intensity is found in **Table 2.**
Conservatively assessed value for total pressure $p = p_s + W_{sd}$ is tabulated in **Table 4**:

| H | H | Wsd | Wsd | p = 1.43+ Wsd |
|---|---|---|---|---|
| (ft) | (in) | ($lb/ft^2$) | ($kip/ft^2$) | ($kip/ft^2$) |
| 1 | 12 | 6249.60 | 6.25 | 7.68 |
| 1.5 | 18 | 5328.00 | 5.33 | 6.76 |
| 2 | 24 | 4464.00 | 4.46 | 5.89 |
| **2.5** | **30** | 3888.00 | 3.89 | 5.32 |
| 3 | 36 | 3312.00 | 3.31 | 4.74 |
| 4 | 48 | 2556.00 | 2.56 | 3.99 |
| 5 | 60 | 2160.00 | 2.16 | 3.59 |
| 6 | 72 | 1713.60 | 1.71 | 3.14 |
| 7.38 | 88.56 | 3312.00 | 3.31 | 4.74 |
| 7 | 84 | 3168.00 | 3.17 | 4.60 |

**Table 4** - Conservatively assessed pressure intensity above the cast iron pipe at various depths

The intensity of pressure p from **Table 4** is compared with the values for the pressure $p_{crushing}$ at the ring-crushing load W:

| D | OD | t | $D_m$ = OD-t | W* | p = 12W/$D_m$ | $p_{crushing}$ |
|---|---|---|---|---|---|---|
| (in) | (in) | (in) | (in) | (lb/ft) | ($lb/ft^2$) | ($kip/ft^2$) |
| 2 | 2.38 | 0.19 | 2.19 | 9331 | 51128.77 | 51.13 |
| 3 | 3.5 | 0.25 | 3.25 | 10885 | 40190.77 | 40.19 |
| 4 | 4.5 | 0.25 | 4.25 | 8324 | 23503.06 | 23.50 |
| 5 | 5.5 | 0.25 | 5.25 | 6739 | 15403.43 | 15.40 |
| 6 | 6.5 | 0.25 | 6.25 | 5660 | 10867.20 | 10.87 |
| 8 | 8.62 | 0.31 | 8.31 | 6546 | 9452.71 | 9.45 |
| 10 | 10.75 | 0.37 | 10.38 | 7465 | 8630.06 | 8.63 |
| 12 | 12.75 | 0.37 | 12.38 | 6259 | 6066.88 | 6.07 |
| 15 | 15.88 | 0.44 | 15.44 | 7097 | 5515.80 | 5.52 |

**Table 5**
Maximum allowable vertical soil pressure for extra-heavy cast iron pipe

*Note: Minimum ring crushing load W for cast iron pipes is tabulated in Table 2 (pg. 103) of Ref. 6.1.7.

The pipe crushing strength will be compared to the applied load. That is, the required strength will be checked as follows:

$$\text{Required three-edge bearing strength} = \frac{\text{design load x factor of safety}}{\text{bedding factor}} = \frac{\text{design load x 1.25}}{1.1}$$

Required three-edge bearing strength = 1.136 x design load

Comparison of **Table 4** and **Table 5** provides the following conclusion: Minimum crushing strength of cast iron pipe of 12" diameter is 6.07 kip/ft$^2$. At 2 feet below the grade, we have conservatively assessed the load of 5.89 kip/ ft$^2$. The exact application of the soil pressure would result in a smaller load and, hence, the pipe would conform to the required check. However, it is not a reasonable expectation to find a 12" pipe buried at 2 feet under the ground.

Pipes with smaller diameter have crushing load greater than 12" pipe; hence they are adequate even for the depth of 2' below the ground.

3.0 Longitudinal bending check:

As per Table 4 of Ref. 6.1.7, pipes do not need evaluation of beam stresses. Per Table 1, the maximum allowable depth for theses pipes, assuming the most conservative trench condition is more than 30 feet.

Conclusion:

- All cast iron pipes with diameter D ≤ 6" are safe for the load due to the Transporter when they are buried even one foot under the grade. Pipes with diameter less than 6" do not need protection even when closer to the grade than 12 in.

8.2.1.2 Pressurized Cast Iron Pipes

Internal pressure that pipe can withstand is determined for the limiting depth of 2 feet. This is adequate because smaller diameter conduits could be placed at depths as shallow as 2'.
Thus, this is a conservative approach in the course of the assessing the internal pressure, since there is no pipe that is exposed to the larger external load than the one at 2 feet depth. If the pipe can be internally pressurized and still resist the pressure from Transporter, when only 2 feet under the grade, than it is the

bounding load case. The allowable operating pressure will be found for the pipes diameter 2" to 6" buried at 2 feet depth:

- Burst strength of pipe with no external load is found as $P = \frac{2St}{D_M}$ (lb/in$^2$)
- External load w (lb/ft) is found as: $w = \frac{W_{sd}}{OD} x12$, where $W_{sd}$ is the external pressure tabulated in **Table 4**.
- Crushing load W (lb/ft) is found in **Table 5**.
- Internal pressure $p = P - \frac{w^2P}{W^2}$ (lb/in$^2$), $P = \frac{2x18000xt}{D_M}$

Usual conservative method that refers to the design of pressure piping is to assume that the pressure p: p = 2(working pressure + 100 lb/in$^2$ surge allowance) (Ref. 6.1.1 ).

<u>Small diameter cast iron pipes</u>: D ≤ 6.0 in

For pipes buried 2 feet deep under the grade we have:

| | Pipe Diameter | | | | |
|---|---|---|---|---|---|
| **H = 2 feet** | 2" | 3" | 4" | 5" | 6" |
| **$D_M$(in)** | 2.19 | 3.25 | 4.25 | 5.25 | 6.25 |
| **t (in)** | 0.19 | 0.25 | 0.25 | 0.25 | 0.25 |
| **S(lb/in$^2$)** | 18000.00 | 18000.00 | 18000.00 | 18000.00 | 18000.00 |
| **P(lb/in$^2$)** | 3123.29 | 2769.23 | 2117.65 | 1714.29 | 1440.00 |
| **W(lb/ft)(crushing strength)** | 9331.00 | 10885.00 | 8324.00 | 6739.00 | 5660.00 |
| **$W_{sd}$(lb/ft$^2$)** | 5890.00 | 5890.00 | 5890.00 | 5890.00 | 5890.00 |
| w(lb/ft) =$W_{sd}$ x OD/12 | 1168.18 | 1717.92 | 2208.75 | 2699.58 | 3190.42 |
| **p =P-(w$^2$P/W$^2$)** | 3074.33 | 2700.25 | 1968.55 | 1439.19 | 982.46 |

**Table 6 -** Allowable extra heavy cast iron pipe operating pressure p at depth of 2'deep under grade

<u>Conclusion:</u>

Based on **Table 6** we can conclude that the pipes with diameter D<6" have adequate capacity to sustain the combination of external and internal load. The internal pressure has been conservatively taken as 500 psi in the earlier pipe evaluations (Ref. 6.2.1). Based on the results tabulated in **Table 6,** small conduits can withstand this pressure.

Calculation SGRP-C-003 (Ref. 6.2.1) performed assessment for D = 6" pipe buried at 3', hence there is no evidence that D = 6" cast iron pipe is pressurized and placed 2 feet under the grade. The case of D = 6" pressurized pipe at 3' depth below the grade is addressed in the next section.

<u>Large diameter cast iron pipes</u>: D ≥ 6.0 in

Based on Ref. 6.2.1 (including Attachment A, i. e. Westinghouse calculation IP2-002-0302-01), we have a 6" pipe at the depth of 3 feet, 8" pipe at the depth of 5 feet, 10" pipe at the depth of 7 feet and 16" pipe at the depth of 4 feet. In addition, review of the IP3 drawings (**Table I**, Section 5.0) has resulted in finding a 10" pipe buried at 4'-6". Since the check of Ref. 6.1.7 confirmed that the largest pipe diameter is 15", the calculation will be done with this D =15" and H = 4.0 ft.

The earth pressure will be found for respective pipes, using the methodology outlined in Section 8.2.1:

| **D** (in) | **OD**(in) | **H** (ft) | $B_d$ (in) | $H/B_d$ | $C_d$ | $W_d$ (lb/ft) |
|---|---|---|---|---|---|---|
| 6.00 | 6.50 | 3.00 | 20.125 | 1.79 | 1.40 | 472.52 |
| 8.00 | 8.62 | 5.00 | 22.775 | 2.63 | 1.70 | 734.83 |
| 10.00 | 10.75 | 4.50 | 25.438 | 2.12 | 1.50 | 808.83 |
| 10.00 | 10.75 | 7.00 | 25.438 | 3.30 | 2.00 | 1078.44 |
| 15.00 | 15.88 | 4.00 | 31.850 | 1.51 | 0.12 | 101.44 |

**Table 7** - Earth pressure above cast iron pipes

The Live load is found from **Table 4**. The values for internal pressure (found per methodology of Section 8.2.1) are tabulated in **Table 8**:

| | | | | | | |
|---|---|---|---|---|---|---|
| D | (in) | 6.00 | 8.00 | 10.00 | 10.00 | 15.00 |
| OD | (in) | 6.50 | 8.62 | 10.75 | 10.75 | 15.88 |
| H | (ft) | 3.000 | 5.000 | 4.500 | 7.00 | 4.00 |
| $B_d$ | (l) | 20.125 | 22.775 | 25.44 | 25.44 | 31.85 |
| $H/B_d$ | | 1.79 | 2.63 | 2.12 | 3.30 | 1.51 |
| $C_d$ | | 1.40 | 1.70 | 1.50 | 2.00 | 0.12 |
| $W_d$ | (lb/ft) | 472.52 | 734.83 | 808.83 | 1078.44 | 101.44 |
| $W_{sd}$ **(Table 2)** | (lb/ft$^2$) | 3312.00 | 2160.00 | 2360.00 | 3168.00 | 2556.00 |
| $W_{sd}$xOD/12 | (lb/ft) | 1794.00 | 1551.60 | 2114.17 | 2838.00 | 3382.44 |
| $w = W_d+W_{sd}$ | (lb/ft) | 2266.52 | 2286.43 | 2923.00 | 3916.44 | 3483.88 |
| t | (in) | 0.25 | 0.31 | 0.37 | 0.37 | 0.36 |
| $P = 2St/D$ | (lb/in$^2$) | 1440.00 | 1342.96 | 1283.24 | 1283.24 | 835.05 |
| W | (lb/ft) | 5660.00 | 6546.00 | 7465.00 | 7465.00 | 7097.00 |
| $p = P-(w^2/W^2)P$ | (lb/in$^2$) | 1209.09 | 1179.12 | 1086.49 | 930.03 | 633.82 |

**Table 8** - Allowable operating pressure for cast iron pipes under the Transporter load

The pipes are adequate for the loads. Having in mind that load due to the Transporter is not a permanent load, the internal pressure intensity that the pipe can withstand is acceptable.

### 8.2.2 Ductile Iron Pipes

(Pipes conforming to AWWA1 Class 150, per Ref. 6.2.4)

The Ductile Iron Pipes within the haul path are identified in Ref. 6.2.4, Section II, 2. as AWWA Class 150 pipe.

Per Ref. 6.1.1, pg. 207, all Ductile Iron Pipes conform to AWWA C-150 "Thickness design of ductile iron pipe", class 52, and AWWA C-104 "Cement mortar lining for ductile iron". ($F_y$ = 42.0 ksi, Ref. 6.1.1, pg. 209).

Methodology for the evaluation follows AWWA C150 approach, which is also given in <u>Buried Pipe Design</u> (Reference 6.1.1, pg.137). The thickness of the ductile iron pipe is checked by considering stresses due to earth loads, ring deflection and internal pressure separately and independently.

1.0 Trench Load assessment:

- Earth load, $P_e$ (lb/in$^2$), Table 5.24 of Reference 6.1.1 or as a soil prism load: $P_e = \gamma H$.
- Pressure $P_t$ (lb/in$^2$), determined from Section 8.1 of this calculation
- Trench load is $P_v = P_e + P_t$ (lb/in$^2$)

2.0 *Conduit wall compression:*

Conduit wall compression is: $C = P\,\frac{\text{span}}{2} = P \times \frac{OD}{2}$ (Ref. 6.1.3, *Section II, eqn.* 1)

P is the total load on the pipe

Buckling stress $f_c = \frac{C}{12xA}$, where A = t x 1" (t = thickness of the pipe) is compared to $f_{b,\,all}$.

Internal pressure check is performed for the safe working stress (50% of yield), and working pressure plus surge)

Buckling of the wall: Allowable buckling stress: $f_b = 0.5x\ F_y = 0.5x\ 42{,}000$ psi = 21,000 psi (Ref. 6.1.1, pg. 138)

3.0 Bending stress check:

Based on Ref. 6.1.8 (Table 17.1), the maximum bending moment is = M = 0.3183WR, where R is the Radius to the centroid of the cross-section. Bending stress is checked:

$$f_b = \frac{M}{S}\ \text{, where}\quad S = \frac{bd^2}{6} = \frac{1.0 \times t^2}{6}$$

Total resulting stress is a sum of the bending stress and the wall thrust stress.

The stress produced by the total external loading is limited to 48,000 lb/in$^2$ (Ref. 6.1.1, pg. 137). This stress is the sum of the bending stress and the wall thrust stress.

4.0 Deflection:

The ring deflection is limited to 3 percent (Ref. 6.1.1, pg. 138). This is a design condition independent of wall stress, limited to protect inner lining from cracking or spalling.

Iowa formula (Ref. 6.1.1 en. 3.5), with 3 percent limit:

$$\Delta x = \frac{D_L K W_c r^3}{EI + 0.061E' r^3},$$

Where: $D_L$ = deflection lag factor, taken as $D_L$ = 1.0 when soil prism load is calculated (Reference 6.1.1, page 52)

K = bedding constant, conservatively taken as K = 0.11
(Table 3.3, pg. 52 of Ref. 6.1.1)
$W_c$ = load on pipe,
r = mean radius = OD- t
EI = pipe wall stiffness:
E = $30\text{x}10^6$ psi, for steel pipes,
E = 24 $\text{x}10^6$ psi, for ductile iron pipes (Table 4.4 of Ref. 6.1.1),

$$I = \frac{t^3}{12}\ (\text{in}^4/\text{in})$$

E' = Modulus of Soil Reaction, as per Table 3.4 of Ref. 6.1.2, or taken as E' = 2000 psi for moderate proctor (85%-95% density) backfill

5.0 Internal pressure:

The wall stress due to internal pressure is checked separately and must be equal to or less than 21,000 lb/in$^2$ (Ref. 6.1.1, Chapter IV, pg. 138).

The stress is: $S = \frac{PD}{2t}$ (limit 21,000 lb/in$^2$), and the allowable pressure:

$P = \frac{2tS}{D}$, where P = working pressure + surge allowance

<u>Determination of conduit stresses:</u>

<u>Ring compression</u>:

Ring compression stresses due to external load for small conduits ($D \leq$ 6.0 in) are found for the depth of burial of 2'. The stresses for large conduits $D \geq 6.0$ are found for the depth of burial of 3'. **Table 9** presents a tabulation of these stresses. It is obvious that the stresses are below the allowable stress of 21,000 psi.

*) Note that dimensions for size 2" are not listed in AWWA C150. The dimensions are taken from ASTM A53.

H = 2'  $W_{sd}$ = 4464 lb/ft$^2$

| Pipe size | OD | Minimum wall thickness $t_{min}$ | $A = t_{min}x12$ | Wc = 120xHxOD/12 | $W_{sd}$xOD/12 | $P_t = W_c+W_{sd}$ | $p_b = P_t/2A$ |
|---|---|---|---|---|---|---|---|
| | (in) | (in) | (in$^2$/ft) | (lb/ft) | (lb/ft) | (lb/ft) | (lb/in$^2$) |
| 2" | 2.38 | 0.15 | 1.85 | 47.50 | 883.50 | 931.00 | 251.89 |
| 3" | 3.96 | 0.25 | 3.00 | 79.20 | 1473.12 | 1552.32 | 258.72 |
| 4" | 4.80 | 0.26 | 3.12 | 96.00 | 1785.60 | 1881.60 | 301.54 |
| 6" | 6.90 | 0.25 | 3.00 | 138.00 | 2566.80 | 2704.80 | 450.80 |

H = 3'  $W_{sd}$ = 3312.0 lb/ft$^2$

| Pipe size | OD | Minimum wall thickness $t_{min}$ | $A = t_{min}$ x 12 | Wc = 120xHxOD/12 | $W_{sd}$xOD/12 | $P_t = W_c+W_{sd}$ | $p_b = P_t/2A$ |
|---|---|---|---|---|---|---|---|
| | (in) | (in) | (in$^2$/ft) | (lb/ft) | (lb/ft) | (lb/ft) | (lb/in$^2$) |
| 6" | 6.90 | 0.25 | 3.00 | 207.00 | 1904.40 | 2111.40 | 351.90 |
| 8" | 9.05 | 0.27 | 3.24 | 271.50 | 2497.80 | 2769.30 | 427.36 |
| 10" | 11.10 | 0.29 | 3.48 | 333.00 | 3063.60 | 3396.60 | 488.02 |
| 12" | 13.20 | 0.31 | 3.72 | 396.00 | 3643.20 | 4039.20 | 542.90 |
| 16" | 17.40 | 0.34 | 4.08 | 522.00 | 4802.40 | 5324.40 | 652.50 |

**Table 9**

Ring compression stresses for small conduits buried at 2 feet below grade

Determination of bending stresses:

Bending stresses are tabulated in **Table 10.**

H = 2' $W_{sd}$ = 4464 lb/ft$^2$

| **Pipe size** | **OD** | **Minimum wall thickness $t_{min}$** | **S = $t_{min}^2/6$** | **Wc = 120xHxOD/ 12** | **$W_{sd}$xOD/12** | **$P_t$ = $W_c$+$W_{sd}$** | **R = (OD-t)/2** | **$M_b$ = 0.3183 $P_t$xR** | **$f_b$ = $M_b$/S** |
|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | (in$^3$) | (lb/ft) | (lb/ft) | (lb/ft) | (in) | (lbin/in) | (lb/in$^2$) |
| 3" | 3.96 | 0.25 | 0.01 | 79.20 | 1473.12 | 1552.32 | 1.86 | 76.38 | 7332.50 |
| 4" | 4.80 | 0.26 | 0.01 | 96.00 | 1785.60 | 1881.60 | 2.27 | 113.29 | 10055.72 |
| 6" | 6.90 | 0.25 | 0.01 | 138.00 | 2566.80 | 2704.80 | 3.33 | 238.55 | 22900.95 |

H = 3' $W_{sd}$ = 3312.0 lb/ft$^2$

| **Pipe size** | **OD** | **Minimum wall thickness $t_{min}$** | **S = $t_{min}^2/6$** | **Wc = 120xHx OD/12** | **$W_{sd}$xOD/ 12** | **$P_t$ = $W_c$+$W_{sd}$** | **R = (OD-t)/2** | **$M_b^{*)}$ = 0.3183 $P_t$xR** | **fb = $M_b$/S** |
|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | (in$^2$/ft) | (lb/ft) | (lb/ft) | (lb/ft) | (in) | (lbin/in) | (lb/in$^2$) |
| 6" | 6.90 | 0.25 | 0.01 | 207.00 | 1904.40 | 2111.40 | 3.33 | 186.22 | 17876.76 |
| 8" | 9.05 | 0.27 | 0.01 | 271.50 | 2497.80 | 2769.30 | 4.39 | 322.47 | 26540.78 |
| 10" | 11.10 | 0.29 | 0.01 | 333.00 | 3063.60 | 3396.60 | 5.41 | 486.96 | 34741.67 |
| 12" | 13.20 | 0.31 | 0.02 | 396.00 | 3643.20 | 4039.20 | 6.45 | 690.52 | 43112.33 |
| 16" | 17.40 | 0.34 | 0.02 | 522.00 | 4802.40 | 5324.40 | 8.53 | 1204.69 | 62527.13 |

**Table 10 -** Ductile pipe stresses due to bending

*)**Note:** Numeric discrepancies off the tabulated results occur due to the significant numbers involved in the calculation.

Obviously, the pipe total stresses are within the limit unless the pipe is 16" in diameter. Pipe of 12" diameter is within the allowable limit, even though there is no evidence that the pipe of this diameter is placed at 3' below the grade. Pipes that are placed deeper are exposed to lower stresses due to the live load, and, hence total stresses are lower.

Based on Ref. 6.2.1, the 16" pipe is buried at the depth of 4 feet. For this depth we have from **Table 2**: $W_{sd} = 2556$ lb/ft$^2$.

Therefore:
$P_t = 120.0 \times 4.0 \times 17.4/12 + 2556 \times 17.4/12 = 696.0$ lb/ft + 3706.2 lb/ft = 4402.2 lb/ft

$M_b = 0.3183 \times P_t \times R = 0.3183 \times 4402.2 \times 8.53/12 = 996.03$ lbin/in

$f_b = M_b/S = 996/0.02 = 49801$ lb/in$^2$.

The pressure determined by the AASHTO method is, at a depth of 4 feet is: $p = 9.04 \times 144$ lb/ft$^2$ =1301.76 lb/ft$^2$ (Table 1 of Section 9.1). This is a r =1301.76/2556 = 0.509 times lower load, and hence, results in 0.509 time lower stress:

$f_b = 0.509 \times 49801 = 25379.42$ lb/in$^2$

Since the stress is below the yield, for one time load application, such as the Transporter traveling over the pipe, the pipe is found to be satisfactory.

<u>Deflection check:</u>

Iowa formula (Ref. 6.1.1 en. 3.5), with 3 percent limit: $\Delta x = \dfrac{D_L K W_c r^3}{EI + 0.061E'r^3}$. Results are tabulated in **Table 11**:

| H = 2' | $P_t$ from Table 10 | | | $D_L = 1.0$, $K = 0.11$, $E = 29x10^6$ lb/in$^2$, $E' = 2000$ lb/in$^2$ | | | | |
|---|---|---|---|---|---|---|---|---|
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **$I = t_{min}^3/12$** | **$W_c = P_t/12$** | **r** | **EI** | **dx** | **(dx/D)x100** |
| | (in) | (in) | | (lb/in) | (in) | lb/(in) | (in) | % |
| 3" | 3.96 | 0.25 | 0.00130 | 129.36 | 1.86 | 37760.42 | 0.002357 | 0.06 |
| 4" | 4.80 | 0.26 | 0.00146 | 156.80 | 2.27 | 42475.33 | 0.004595 | 0.10 |
| 6" | 6.90 | 0.25 | 0.00130 | 225.40 | 3.33 | 37760.42 | 0.021575 | 0.32 |
| **H = 3'** | $P_t$ from **Table 10** | | | $D_L = 1.0$, $K = 0.11$, $E = 29x10^6$ lb/in$^2$, $E' = 2000$ lb/in$^2$ | | | | |
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **$I = t_{min}^3/12$** | **$W_c = P_t/12$** | **r** | **EI*)** | **dx** | **(dx/D)x 100** |
| | (in) | (in) | | (lb/in) | (in) | lb/(in) | (in) | % |
| 6" | 6.90 | 0.25 | 0.00130 | 175.95 | 3.33 | 37760.42 | 0.016841 | 0.25 |
| 8" | 9.05 | 0.27 | 0.00164 | 230.78 | 4.39 | 47567.25 | 0.0371 | 0.42 |
| 10" | 11.10 | 0.29 | 0.00203 | 283.05 | 5.41 | 58940.08 | 0.062866 | 0.58 |
| 12" | 13.20 | 0.31 | 0.00248 | 336.60 | 6.45 | 71994.92 | 0.094714 | 0.73 |
| 16" | 17.40 | 0.34 | 0.0033 | 443.70 | 8.53 | 94984.67 | 0.177454 | 1.04 |

**Table 11** - Ductile iron pipes deflection

*)**Note:** Numeric discrepancies off the tabulated results occur due to the significant numbers involved in the calculation.

The pipe deflections are within the allowable range of 3%.

Allowable internal pressure:

| Pipe size | OD | t | $D_M$ | P |
|---|---|---|---|---|
| | (in) | (in) | (in) | (lb/in$^2$) |
| 3" | 3.96 | 0.25 | 3.71 | 2830.189 |
| 4" | 4.80 | 0.26 | 4.54 | 2405.286 |
| 6" | 6.90 | 0.25 | 6.65 | 1578.947 |
| 8" | 9.05 | 0.27 | 8.78 | 1291.572 |
| 10" | 11.10 | 0.29 | 10.81 | 1126.735 |
| 12" | 13.20 | 0.31 | 12.89 | 1010.085 |
| 16" | 17.40 | 0.34 | 17.06 | 837.0457 |

**Table 12** - Allowable internal stresses for ductile iron pipes

Obviously, the allowable internal stresses are not governing the stress condition.

### 8.2.3 Steel Pipes

As per Specification No. 9321-01-44-1, Rev. 1: Specification for Yard Storm Drainage and Yard and Building Standpipe Fire Protection System (Ref. 6.2.4) (and per Specification No. 9321-01-248-18, Rev. 3, Ref. 6.2.5: Specification for Fabrication of Piping Systems), the steel pipes conform to The ASTM A53 and ASTM 106, grade A ($F_y$ = 30.0 ksi for either one, as per Ref. 6.1.11 and 6.1.12 respectively).

For most flexible pipes, such as steel, ductile iron and thermal plastic, a combined loading analysis is not necessary. The pipe is designed and will be checked as if external loading and internal pressure were acting independently (Ref. 6.1.1, pg. 119). Usually, internal pressure governs the design.

1. Determination of the Earth Load

Determination of the Earth Load will be performed by guidance of Ref. 6.1.1, equation 2.11. The prism load is found as

$W_c = \gamma HB_c$, $\gamma$ = unit weight of soil

H = depth at which soil pressure is required
$B_c$ = Outside diameter of pipe

2. Determination of the Live Load

Live Load is determined as per Section 8.1 of this calculation and tabulated in **Table 2.**

3. Conduit wall compression

Conduit wall compression is: $C = P\frac{\text{span}}{2} = P \times \frac{OD}{2}$ (Ref. 6.1.3, Section II, eqn. 1)

P is the total load on the pipe

Buckling of the wall: Allowable buckling stress: $f_b = 0.5x\ F_y = 0.5x\ 30000$ psi = 15000 psi (Ref. 6.1.1, pg 139)

Buckling stress $f_c = \frac{C}{12xA}$, where A = t x 1" (t = thickness of the pipe) is compared to $f_{b,\ all}$.

4. Deflection check:

Modified Iowa formula: (Ref. 6.1.1, eqn. 3.5)

$$\Delta x = \frac{D_L K W_c r^3}{EI + 0.061E' r^3},$$

Where: $D_L$ = deflection lag factor, taken as $D_L$ = 1.0 when soil prism load is calculated (Reference 6.11, page 52)

K = bedding constant, conservatively taken as K = 0.11
(Table 3.3, pg. 52 of Ref. 6.1.1)

$W_c$ = load on pipe,
r = mean radius = OD- t
EI = pipe wall stiffness:
E = $30 \times 10^6$ psi, for steel pipes,
E = $24 \times 10^6$ psi, for ductile iron pipes (Table 4.4 of Ref. 6.1.1),

$$I = \frac{t^3}{12} \text{ (in}^4\text{/in)}$$

E' = Modulus of Soil Reaction, as per Table 3.4 of Ref. 6.1.2, or taken as E' = 2000 psi for excellent backfill at 95% density

For rigid pipes in typical installations, deflection is limited to between 1 and 2%.

For flexible pipes in typical installations, deflection is limited to less than 5%.

5. Bending stress check:

Based on Ref. 6.1.8 (Table 17.1), the maximum bending moment is = M = 0.3183WR, where R is the Radius of the centroid of the cross-section. Bending stress is checked:

$$f_b = \frac{M}{S} \text{, where } S = \frac{bd^2}{6} = \frac{1.0 \times t^2}{6}$$

For flexible pipes, the bending stress is calculated per eqn. 4.11 of Ref. 6.1.1 (pg. 116):

$$\sigma = D_f E \left(\frac{\Delta y}{D}\right)\left(\frac{t}{D}\right)$$, where: D = pipe outside diameter

t = pipe wall thickness

E = Young's modulus

$\Delta y$ = vertical deflection (as calculated in step 4)

$D_f$ = shape factor = $\dfrac{F}{\Delta y}$ = 6.7 EI/r$^3$ (Ref. 6.1.1, page 48)

Determination of conduit wall compression:

For all conduits buckling stress is found and tabulated in **Table 13**. Note that small conduits have been analyzed for depth of cover of 2 feet and large conduits for 3'. Values obtained for conduit wall compression are well below buckling allowable stress of 0.5 x30 ksi = 15.0 ksi.

| H = 2', $W_{sd}$ = 4464 lb/ft$^2$ | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **A = $t_{min}$x12** | **Wc = 120xHxOD/12** | **$W_{sd}$xOD/12** | **$P_t = W_c + W_{sd}$** | **$f_b = P_t/2A$** |
| | (in) | (in) | (in$^2$/ft) | (lb/ft) | (lb/ft) | (lb/ft) | (lb/in$^2$) |
| 2" | 2.375 | 0.154 | 1.85 | 47.50 | 883.50 | 931.00 | 251.89 |
| 3" | 3.500 | 0.216 | 2.59 | 70.00 | 1302.00 | 1372.00 | 264.66 |
| 4" | 4.500 | 0.237 | 2.84 | 90.00 | 1674.00 | 1764.00 | 310.13 |
| 5" | 5.563 | 0.258 | 3.10 | 111.26 | 2069.44 | 2180.70 | 352.18 |
| 6" | 6.625 | 0.188 | 2.26 | 132.50 | 2464.50 | 2597.00 | 575.58 |

| H = 3' | | $W_{sd}$ = 3312.0 lb/ft$^2$ | | | | | |
|---|---|---|---|---|---|---|---|
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **A = $t_{min}$ x 12** | **Wc = 120xHxOD/12** | **$W_{sd}$xOD/12** | **$P_t = W_c + W_{sd}$** | **$f_b = P_t/2A$** |
| | (in) | (in) | (in$^2$/ft) | (lb/ft) | (lb/ft) | (lb/ft) | (lb/in$^2$) |
| 6" | 6.625 | 0.280 | 3.36 | 198.75 | 1828.50 | 2027.25 | 301.67 |
| 8" | 8.625 | 0.250 | 3.00 | 258.75 | 2380.50 | 2639.25 | 439.88 |
| 10" | 10.750 | 0.250 | 3.00 | 322.50 | 2967.00 | 3289.50 | 548.25 |
| 12" | 12.750 | 0.250 | 3.00 | 382.50 | 3519.00 | 3901.50 | 650.25 |
| 16" | 16.000 | 0.375 | 4.50 | 480.00 | 4416.00 | 4896.00 | 544.00 |

**Table 13**
Buckling stress
for small conduits ( D≤ 6.0") at depth of 2 feet below grade
and for arge diameter conduits (D≥6.0") at depth of 3 feet below grade

Determination of Pipe deflection:

Pipe deflection per Modified Iowa formula with the actual percent value are tabulated in **Table 14**:

| H = 2' | $P_t$ from Table 13 | | | $D_L$ = 1.0, K = 0.11, E = 29x10$^6$ lb/in$^2$, E' = 2000 lb/in$^2$ | | | | |
|---|---|---|---|---|---|---|---|---|
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **$I = t_{min}^3/12$** | **$W_c = P_t/12$** | **r** | **EI** | **dx** | **dx/D** |
| | (in) | (in) | | (lb/in) | (in) | lb/(in) | (in) | % |
| 2" | 2.375 | 0.154 | 3.044E-04 | 77.58 | 1.11 | 8826.305 | 0.0013 | 0.058512 |
| 3" | 3.500 | 0.216 | 8.398E-04 | 114.36 | 1.64 | 24354.43 | 0.002237 | 0.068121 |
| 4" | 4.500 | 0.237 | 1.109E-03 | 147.00 | 2.13 | 32170.79 | 0.004695 | 0.110135 |
| 5" | 5.563 | 0.258 | 1.431E-03 | 181.73 | 2.65 | 41502.65 | 0.008521 | 0.160627 |
| 6" | 6.625 | 0.280 | 1.829E-03 | 216.42 | 3.17 | 53050.67 | 0.013348 | 0.210374 |

| H = 3' | $P_t$ from Table 10 | | | $D_L$ = 1.0, K = 0.11, E = 29x10$^6$ lb/in$^2$, E' = 2000 lb/in$^2$ | | | | |
|---|---|---|---|---|---|---|---|---|
| **Pipe size** | **OD** | **Minimum wall thickness, $t_{min}$** | **$I = t_{min}^3/12$** | **$W_c = P_t/12$** | **r** | **EI*)** | **dx** | **dx/D** |
| | (in) | (in) | | (lb/in) | (in) | lb/(in) | (in) | % |
| 6" | 6.625 | 0.280 | 1.829E-03 | 175.95 | 3.17 | 53050.67 | 0.010852 | 0.171037 |
| 8" | 8.625 | 0.277 | 1.771E-03 | 230.78 | 4.17 | 51363.67 | 0.030647 | 0.367116 |
| 10" | 10.750 | 0.279 | 1.810E-03 | 283.05 | 5.24 | 52484.29 | 0.063838 | 0.609667 |
| 12" | 12.750 | 0.330 | 2.995E-03 | 336.60 | 6.21 | 86847.75 | 0.076398 | 0.61512 |
| 16" | 16.000 | 0.375 | 4.395E-03 | 443.70 | 7.81 | 127441.4 | 0.125383 | 0.80245 |

**Table 14** - Steel Pipe Deflection

*)**Note:** *Numeric discrepancies off the tabulated results occur due to the significant numbers involved in the calculation.*

Deflections of all pipes are less than 1%, hence stresses due to bending from flexible installations are not applicable.

Bending stresses will be found per Roark methodology (Ref. 6.1.8): M = 0.3183WR, where R is the Radius of the centroid of the cross-section. Bending stress is checked:

$$f_b = \frac{M}{S}\text{, where}\quad S = \frac{bd^2}{6} = \frac{1.0 \times t^2}{6}$$

H =2.0' $W_{sd}$ = 4464 lb/ft$^2$

| Pipe size | OD | Minimum wall thickness $t_{min}$ | S = $t_{min}^2$/6 | Wc = 120xHxOD/12 | $W_{sd}$xOD/12 | $P_t$ = $W_c$+$W_{sd}$ | R = (OD-t)/2 | $M_b$ *) = 0.3183 $P_t$xR | $f_b$ = $M_b$/S |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | (in) | (in) | (in$^3$) | (lb/ft) | (lb/ft) | (lb/ft) | (in) | (lbin/in) | (lb/in$^2$) |
| 2" | 2.375 | 0.154 | 0.004 | 47.50 | 883.50 | 931.00 | 1.11 | 27.42 | 6937.99 |
| 3" | 3.500 | 0.216 | 0.008 | 70.00 | 1302.00 | 1372.00 | 1.64 | 59.76 | 7684.69 |
| 4" | 4.500 | 0.237 | 0.009 | 90.00 | 1674.00 | 1764.00 | 2.13 | 99.73 | 10653.54 |
| 5" | 5.563 | 0.258 | 0.011 | 111.26 | 2069.44 | 2180.70 | 2.65 | 153.43 | 13829.86 |
| 6" | 6.625 | 0.280 | 0.013 | 132.50 | 2464.50 | 2597.00 | 3.17 | 218.54 | 16724.92 |

| Pipe size | OD | Minimum wall thickness $t_{min}$ | S = $t_{min}^2$/6 | H | $W_{sd}$ | Wc = 120xHxOD/12 | $W_{sd}$xOD/12 | $P_t$ = $W_c$+$W_{sd}$ | R = (OD-t)/2 | $M_b$ *) = 0.3183 $P_t$xR | $f_b$ = $M_b$/S |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | (in$^2$/ft) | ft | (lb/ft$^2$) | (lb/ft) | (lb/ft) | (lb/ft) | (in) | (lbin/in) | (lb/in$^2$) |
| 6" | 6.625 | 0.280 | 0.013 | 3.00 | 3312.00 | 198.75 | 1828.50 | 2027.25 | 3.17 | 170.59 | 13055.68 |
| 8" | 8.625 | 0.277 | 0.013 | 5.00 | 2160.00 | 431.25 | 1552.50 | 1983.75 | 4.17 | 219.63 | 17174.59 |
| 10" | 10.750 | 0.279 | 0.013 | 5.00 | 2160.00 | 537.50 | 1935.00 | 2472.50 | 5.24 | 343.36 | 26466.27 |
| 10" | 10.750 | 0.279 | 0.013 | 7.00 | 3168.00 | 752.50 | 2838.00 | 3590.50 | 5.24 | 498.62 | 38433.62 |
| 12" | 12.750 | 0.330 | 0.018 | 5.00 | 2160.00 | 637.50 | 2295.00 | 2932.50 | 6.21 | 483.04 | 26613.89 |
| 16" | 16.000 | 0.375 | 0.023 | 4.00 | 2556.00 | 640.00 | 3408.00 | 4048.00 | 7.81 | 838.85 | 35791.07 |

**Table 15** - Pipe stresses due to bending

*) **Note:** Numeric discrepancies off the tabulated results occur due to the significant numbers involved in the calculation.

Conduits larger than 6" are shown to be stressed at more than half the yield stress.

Pipes bending stresses due to the pressure determined by the AASHTO methodology are tabulated in **Table 16**:

| **OD** | **Minimum wall thickness $t_{min}$** | **$S = t_{min}^2/6$** | **H** | **$W_{sd}$** (Table 1) | **Wc = 120xHx OD/12** | **$W_{sd}$xOD/ 12** | **$P_t = W_c+W_{sd}$** | **R = (OD-t)/2** | **$M_b$[*)] = 0.3183 PxR** | **fb = $M_b$/S** |
|---|---|---|---|---|---|---|---|---|---|---|
| (in) | (in) | ($in^2$/ft) | ft | ($lb/ft^2$) | (lb/ft) | (lb/ft) | (lb/ft) | (in) | (lbin/in) | ($lb/in^2$) |
| 6.625 | 0.280 | 0.01 | 3.00 | 1727.40 | 198.75 | 953.67 | 1152.42 | 3.17 | 96.98 | 7421.69 |
| 8.625 | 0.277 | 0.01 | 5.00 | 1022.77 | 431.25 | 735.12 | 1166.37 | 4.17 | 129.13 | 10097.98 |
| 10.750 | 0.279 | 0.01 | 5.00 | 1022.77 | 537.50 | 916.23 | 1453.73 | 5.24 | 201.88 | 15561.11 |
| 10.750 | 0.279 | 0.01 | 7.00 | 684.20 | 752.50 | 612.93 | 1365.43 | 5.24 | 189.62 | 14615.90 |
| 12.750 | 0.330 | 0.02 | 5.00 | 1022.77 | 637.50 | 1086.69 | 1724.19 | 6.21 | 284.01 | 15647.91 |
| 16.000 | 0.375 | 0.02 | 4.00 | 1302.58 | 640.00 | 1736.77 | 2376.77 | 7.81 | 492.53 | 21014.64 |

**Table 16** - *Bending on the pipes due to the load determined* by the AASHTO methodology

*)**Note:** Numeric discrepancies off the tabulated results occur due to the significant numbers involved in the calculation.

Obviously, the pipes are stressed below the yield. For this type of application, when the Transporter load is a one at a time load, and when the stress in the pipe is not beyond the yield, we can conclude that the pipes are adequate and do not require protection.

### 8.2.4 Corrugated Metal Pipes (CMP)

Based on Ref. 6.2.1, Attachment A, pg. 10, corrugated metal pipes are placed at 4 feet below the grade.

Conservatively, 24" CMP is evaluated for this condition, based on guidance of Ref. 6.1.3.

Earth load: $W_e = \gamma H = 100 \times 4.0 = 400$ psf ( $\gamma = 100$ lb/ $ft^3$ per Ref. 6.1.3, pg. 42)

Live load: $W_{sd} = 2556\ \text{lb/ft}^2$
Total load: $P_t = 400 + 2556 = 2956\ \text{lb/ft}^2$

<u>Ring Compression</u>: $C = P_t \dfrac{\text{span}}{2} = 2956 \times \dfrac{24}{12\text{x}2} = 2956\ \text{lb/ft} = 246.33\ \text{lb/in}$

14 Ga. steel (Ref. 6.1.3, Table 2-4) with a minimum of 5/16" single riveting provides a seam strength of 18200 lb/ft which is sufficient.

<u>Buckling of wall</u>: C = 2956 lb/ft

Buckling of the wall: $f_b = \dfrac{C}{12\text{x}A}$

A = area of the pipe wall, for 14 Ga = 0.0792 $\text{in}^2/\text{in}$ (Reference 6.1.3, Table 2-3)

$$f_b = \frac{C}{12\text{x}A} = \frac{2956}{12 \times 0.0792} = 3110.27\ \text{psi} < 16500\ \text{psi} = f_{b,\text{all}}$$ ( Ref. 6.1.3, pg. 53)

Deflection check: (Reference 3.1.4, eqn. 4, pg. 51)

$$\Delta x = \frac{D_L K W_c r^3}{EI + 0.061E'r^3} = \frac{1.5 \times 0.1 \times (246.33 \times 2) \times 12^3}{30\text{x}10^6 \times 0.00057 + 0.061 \times 1400 \times 12^3} = 0.853\ \text{in}$$

$I = 0.00057\ \text{in}^4$ / in of width (Reference 6.1.3, Table 2-3)

$\dfrac{\Delta x}{D} = \dfrac{0.853}{24} = 0.0355 \sim 3.6\ \%$ < allowable deflection 5% ( 0.05 x 24 = 1.2 in) , therefore the pipe is satisfactory.

The pipe is adequate even assuming the minimum thickness available.

### 8.2.5 PVC Pipes

PVC sewer pipes are flexible pipes and the evaluation method applied is described in Chapter 3 of Reference 6.1.1.

3"PVC H = 3.0' OD = 3.5 in (Ref. 6.1.14)

Earth load: $W_e$ = 120 pcf x 3' x 3.5"/12" = 105 lb/ft

Live load: $W_{sd}$ x OD = 3312 lb/ft$^2$ x 3.5/12 = 966 lb/ft
Total load on the pipe: W = 105 lb/ft + 966 lb/ft = 1071 lb/ft = 89.25 lb/in

Deflection check: (Reference 6.1.1, eq. 3.5 pg. 51)

$$\Delta x = \frac{D_L K W_c r^3}{EI + 0.061E'r^3}$$

${}_{min}t = 0.216$ in Ref. 7.1.4, $I = {}_{min}t^3/12 = 8.4 \times 10^{-4}$

E = 420000 psi (Table 5.8 of Ref. 6.1.1)

EI = 352.7 lbin$^2$

$$\Delta x = \frac{1.0 \times 0.11 \times 89.25 \times 1.75^3}{353 + 0.061 \times 1400 \times 1.75^3} = 0.06 \text{ in}$$

Conservatively take: $\Delta x = 0.1$

$\frac{\Delta x}{D} = \frac{0.1}{3.0} = 0.035 \sim 3.5$ % < allowable deflection 5% (0.05 x 24 = 1.2 in), therefore the pipe is satisfactory.

Since the pipes are rated for 160 psi, (Ref. 6.2.3) we have: for minimum hydrostatic-design stress of 2000 lb/in$^2$: DR = OD/t = $\frac{2p}{P} + 1 = \frac{2 \times 2000}{160} + 1 = 26$ (Ref. 6.1.1, pg. 181),
The minimum selected pipe must have stiffness higher than 46. The min. pipe stiffness is 115 lb/in$^2$.

DR = OD/t = 3.5/t = 26 , hence: min t = 3.5/26 = 0.1346 in
min wall thickness is t = 0.216 in is found in Ref. 6.1.14, Table 2.

Conservatively, the stress will be checked as for rigid pipe:

$S = t^2/6 = 0.216^2/6 = 0.0078\ in^3$

M = 0.3183 WR = 0.3183 x 89.25x (3.5/2) = 49.71 lb-in/in

Stress: $f_b$ = M/S = 49.71/0.0078 = 6373.65 $lb/in^2$

Typical PVC-Pipe Properties Table 5.8, Ref. 6.1.1 specifies a tensile strength of 7000 psi.

Therefore, PVC pipes of 3" diameter are adequate.

Pipe diameter 1¼", sch. 40 has 0.14 in min. wall thickness (Ref. 6.1.13, Table 2).

Earth load: $W_e$ = 120 pcf x 3' x 1.66"/12" = 49.8 lb/ft

Live load: $W_{sd}$ x OD = 3312 $lb/ft^2$ x 1.66/12 = 458.16 lb/ft

Total load on the pipe: W = 49.8 lb/ft + 458.16 lb/ft = 507.96 lb/ft = 42.33 lb/in

Bending stress is:

$S = 0.14^2/6 = 0.00327\ in^3$

M = 0.3183 x 42.33x (1.66/2) = 11.18 lb-in/i

Stress: $f_b$ = M/S = 11.18/ 0.00327 = 3420 $lb/in^2$

Therefore, PVC pipes of 1¼" diameter, sch. 40 are adequate.

## 8.3 UNDERGROUND DUCTS

### 8.3.1 Underground Duct Cross-Section Shown On Drawing A239208-5

(Ref. 6.2.2)

The underground duct cross-section is depicted on **Figure 3**:
Depth of cover: min H = 3 ft, Ref. 6.2.2

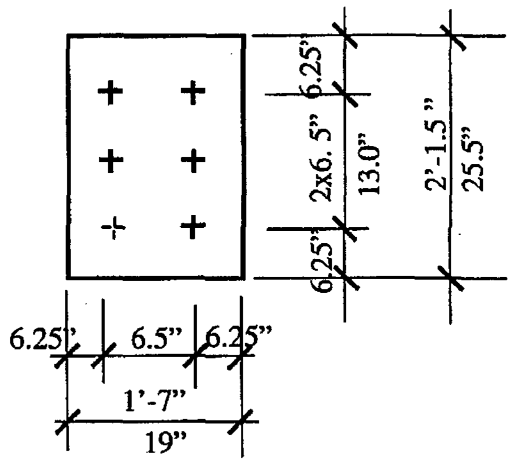


**Figure 3** - Electrical duct cross-section

Conservative account of the section with six holes will be considered:

$$A_I = 19 \times 25.5 - 6 \times 4.5^2 \frac{\pi}{4} = 389.1 \text{ in}^2$$

$$y = 25.5/2 = 12.75 \text{ in}$$

$$I_{x,net} = 19 \times \frac{25.5^3}{12} - (0.785 \times 2.25^4) \times 6 - 2 \times \frac{4.5^2 \pi}{4} \times 6.5^2 \times 2 = 26253.84 - 120.71 - 2687.83 =$$

$$I_{x,net} = 23445.3 \text{ in}^4$$

$$S_x = \frac{I_x}{25.5} x2 = \frac{23445.3x2}{25.5} = 1838.85 \text{ in}^3$$

Load on the duct: H = 3.0' (min), Ref. 6.2.2

Earth load: $p_{e'} = 3.0' x 120 \text{ lb/ft}^3 = 360 \text{ lb/ft}^2$
$p_e = 360 \text{ x } 19/12 = 570 \text{ lb/ft} = 47.5 \text{ lb/in}$

Transporter load: $p_{t'} = 3312 \text{ lb/ft}^2$ **(Table 2)**

$p_t = 3312 \text{ x } 19/12 = 5244 \text{ lb/ft} = 437 \text{ lb/in}$

Total load on the duct: $q = p_e + p_t = 47.5 + 437 = 484.5 \text{ lb/in}$

This load will be placed along the whole affected length of duct, which is taken as
L = 3'+ 19/12 +3' = 7.58 ft = 91 in

Duct will be evaluated as an infinite beam, loaded along 10.5 feet of length with the load
q = 484.5 lb/in

As per Ref. 6.1.4, eqn. 7c: $M_c = \frac{q}{4\lambda^2}(B_{\lambda a} + B_{\lambda b})$, where:

q = 484.5 lb/in

$\lambda = \sqrt[4]{\frac{k_s}{4E_c I_{net}}}$, where $k_s$ = b x k = 19" x 400 pci = 7600 psi  Ref. 6.1.4, pg.2, eqn (a)

$$\lambda = \sqrt[4]{\frac{7600}{4 \times 3.15 \times 10^6 \times 23445.3}} = 0.013 \frac{1}{in}$$

$\lambda l = 0.013 \text{ x } 91" = 1.18$

$\pi/4 < 1.18 < \pi$ ,

Therefore, the beam behaves as a beam of medium length. The characteristic of this group of beams is that force acting at one end of the beam has a finite, not negligible effect on the forces at the other end. As described in Reference 6.1.4, the forces at one point counter the forces at other points, reducing the total load. Consequently, when the formula for an infinite beam is used to design a beam of medium length, the forces will be very conservative.

| Point | a | b | La | Lb | B(La) | B(Lb) | C(La) | C(Lb) | M | Q |
|---|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | | | | | | | (lb/in$^2$) | (lb/in) |
| 1 | 0 | 29.5 | 0.0000 | 0.3835 | 0.0000 | 0.2550 | 1.0000 | 0.3770 | 182752.11 | 5805.06 |
| 2 | 7.35 | 22.15 | 0.0956 | 0.2880 | 0.0867 | 0.2129 | 0.8180 | 0.5060 | 214759.70 | 2907.32 |
| 3 | 14.75 | 14.75 | 0.1918 | 0.1918 | 0.1573 | 0.1573 | 0.6531 | 0.6531 | **225513.14** | 0.00 |
| 4 | 22.13 | 7.375 | 0.2876 | 0.0959 | 0.2128 | 0.0870 | 0.5065 | 0.8174 | 214832.26 | -2897.49 |
| 5 | 29.5 | 0 | 0.3835 | 0.0000 | 0.2550 | 0.0000 | 0.3770 | 1.0000 | 182752.11 | -5805.06 |

**Table 17** - Duct cross-sectional forces

Concrete tensile stress: $\sigma = \dfrac{1.7M}{S} = \dfrac{1.7 \times 225513.14}{1838.85} = 208.48$ lb/in$^2$ < Allowable rupture stress: ${}_{\min}f_r =$ $8\sqrt{3000}$ = ${}_{\min}f_r = 438$ lb/in$^2$ (Reference 6.1.5, Table 2.2, page 49). Based on Ref. 6.1.14, Table 1, Approximate range of Modulus of Rupture for 3000 psi concrete is 450-525 psi. In addition, the duct is reinforced, hence the duct bending capacity is adequate.

Shear check:

Max shear load: $Q = \dfrac{q}{4\lambda} 1.0 = 1.7 \times \dfrac{484.5}{4 \times 0.013} = 1.7 \times 9317.3$ lb $= 15839.42$ lb $= 15.84$ kip

Concrete section area: A = 389.1 in$^2$

Shear capacity for the net section:

$V_c = 2\,\Phi\sqrt{3000}$, Ref. 6.3.1, Section 11.3.1.1

$\Phi = 0.85$, Ref. 6.3.1, Section 9.3.2.3

$V_c = 2 \times 0.85 \times \sqrt{3000} \times 389.1\ in^2 = 36228\ lb = 36.23\ kip$ (Ref. 6.3.1, Section 11.3.1.1) >2x 15.84 = 31.68 kip

Therefore, this duct has sufficient capacity to resist the loads from the Transporter.

### 8.3.2 Underground Duct Located Along The Travel Route, Shown On Dwg. A207621-8 and A207622-1

(Ref. 6.2.6 and 6.2.7, Section J-J)

The duct cross-section is shown on drawing A207622, Section J-J (Ref. 6.2.7). Based on guidance for spacings provided on Dwg. 218486-07 ( Ref. 6.2 8), the duct cross-section is depicted on **Figure 4**:

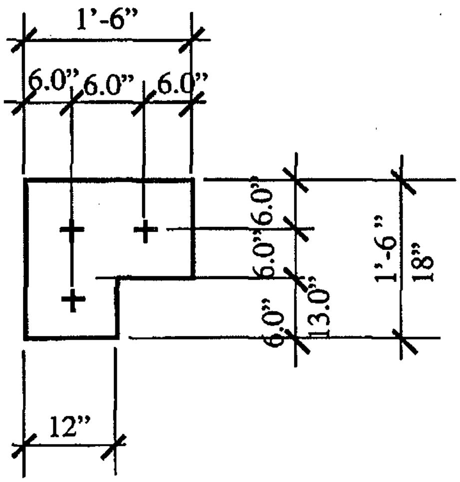


**Figure 4** - Duct cross-section geometry

Duct cross-section properties (included are 3 - 5" conduits):

$$A_1 = 12.200 \times 6.0 - \frac{5.563^2 \pi}{4} = 72.0 - 24.3 = 47.69\ in^2$$

$$A_2 = 18.0 \text{ x } 12.0 - 2\text{x}\frac{5.563^2\pi}{4} = 216.0 - 48.61 = 167.39 \text{ in}^2$$

$$A = A_1 + A_2 = 47.69 + 167.39 = 215.08 \text{ in}^2$$

$$y_{CG} = \frac{47.69 \text{ x } 3.0 + 167.39\text{x}12.0}{215.08} = 10.0 \text{ in}^2$$

$$\Delta y_1 = 10.0 - 3.0 = 6.86 \text{ in}$$

$$\Delta y_2 = 12.0 - 10.0 = 2.14 \text{ in}$$

$$I_x = 12.0\text{x}\frac{6.0^3}{12} + 72.0\text{x}7.0^2 + 18.0\text{x}\frac{12^3}{12} + 216.0\text{x}2.0^2 - 3\text{x } 0.785\text{x}2.7815^4 - \frac{4.0^2\pi}{4}(7.0^2 + 2\text{x}2.0^2) =$$

$$I_x = 216.0 + 3528.0 + 2592 + 864.0 - 140.96 - 1385.42 = 5673.62 \text{ in}^4$$

$$S_x^{TOP} = \frac{I_x}{(18 - 10.0)} = 5673.62/8.0 = 709.2 \text{ in}^3$$

$$S_x^{BOT} = \frac{I_x}{10.0} = 5673.62/10 = 567.36 \text{ in}^3$$

Load on the duct: H = 2.0' (min), Ref. 6.2.7

Duct will be evaluated as an infinite beam, loaded along the whole length with the load:

Earth load: $p_{e'} = 2.0'\text{x}120 \text{ lb/ft}^3 = 240 \text{ lb/ft}^2$
$p_e = 240 \text{ x } 18.0/12 = 360 \text{ lb/ft} = 30.0 \text{ lb/in}$

Transporter load: $q = 3.23 \text{ kip/ft}^2 = 3230.0/144 = 22.43 \text{ lb/in}^2$, which is found by the conventional 30° load spread method, defined in Section 9.1 of this calculation.

$p_t = 22.43 \text{ x } 18 = 403.75 \text{ lb/in}$

Total load on the duct: $q = p_e + p_t = 30.0 + 403.75 = 433.75 \text{ lb/in}$

Cross-sectional forces:

As per Ref. 6.1.4, eqn. 7c: $M_c = \frac{q}{4\lambda^2}(B_{\lambda a} + B_{\lambda b})$, where:

$\lambda = \sqrt[4]{\frac{k_s}{4E_c I_{net}}}$, where $k_s = b \times k =$ 12"x 400 pci = 4800 psi Ref. 6.1.4, pg.2, eqn (a)

$$\lambda = \sqrt[4]{\frac{4800}{4 \times 3.15 \times 10^6 \times 5673.62}} = 0.016\ \frac{1}{in}$$

$\lambda l = 0.016 \times 197" = 3.17 > \pi$, consequently, the beam behaves as a Long Beam.

The cross-sectional forces are determined and tabulated in **Table 18**:

| POINT | a | b | L | La | Lb | B(La) | B(Lb) | C(La) | C(Lb) | M | Q |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | (1/in) | | | | | | | (lb/in$^2$) | (lb) |
| 1 | 0.00 | 197.00 | | 0.000 | 3.152 | 0.000 | 0.000 | 1.000 | -0.042 | -188.53 | **7064.15** |
| 2 | 20.00 | 177.00 | | 0.320 | 2.832 | 0.228 | 0.018 | 0.461 | -0.074 | 104356.59 | 3625.22 |
| | 29.50 | 167.50 | | 0.472 | 2.680 | 0.284 | 0.031 | 0.272 | -0.092 | 133063.56 | 2466.11 |
| 3 | 30.00 | 167.00 | | 0.480 | 2.672 | 0.286 | 0.031 | 0.263 | -0.093 | 134283.28 | 2412.90 |
| 4 | 40.00 | 157.00 | | 0.640 | 2.512 | 0.315 | 0.048 | 0.108 | -0.113 | 153614.01 | 1500.19 |
| 5 | 50.00 | 147.00 | 0.016 | 0.800 | 2.352 | 0.322 | 0.068 | -0.009 | -0.135 | 165160.49 | 849.36 |
| 6 | 60.00 | 137.00 | | 0.960 | 2.192 | 0.314 | 0.091 | -0.094 | -0.156 | 171335.10 | 418.62 |
| 7 | 70.00 | 127.00 | | 1.120 | 2.032 | 0.294 | 0.117 | -0.152 | -0.176 | 174119.64 | 163.87 |
| 8 | 80.00 | 117.00 | | 1.280 | 1.872 | 0.266 | 0.147 | -0.187 | -0.193 | 175048.24 | 39.83 |
| 9 | 90.00 | 107.00 | | 1.440 | 1.712 | 0.235 | 0.179 | -0.204 | -0.204 | **175199.39** | 0.73 |
| 10 | 98.5 | 98.50 | | 1.576 | 1.576 | 0.207 | 0.207 | -0.208 | -0.208 | 175192.51 | 0.00 |

**Table 18** - Duct cross-sectional forces

Concrete tensile stress: $\sigma = \frac{1.7M}{S} = \frac{1.7 \times 175199.39}{567.36} = 524.96\ \text{lb/in}^2$

Allowable rupture stress: ${}_{min}f_r = 8\sqrt{3000} = {}_{min}f_r = 438\ \text{lb/in}^2$ (Reference 6.1.5, Table 2.2, page 49). Based on Ref. 6.1.15, Table 1, Approximate range of Modulus of Rupture for 3000 psi concrete is 450-525 psi.

The duct concrete does not have sufficient capacity to resist the load. The concrete will crack and the tension will be resisted by the duct reinforcement. The reinforcement capacity is found next:

$A_s = 2x\,0.2\ in^2 = 0.4\ in^2$

$$a = \frac{A_s f_y}{0.85 f_c' b} = \frac{0.4x40.0x1000}{0.85x3000x12.0} = 0.52\ in$$

$d = 18.0\text{-}2.0 = 16.0\ \ in$

Reinforcement bending capacity:

$$\Phi M = \Phi A_s\, f_y\, (d - \frac{a}{2}) = 0.9\ x\ 0.4x40000\ (16.0 - \frac{0.52}{2}) = 226635.29\ lb\text{-}in\ = 226.64\ kip\text{-}in$$

$\Phi = 0.9$, Ref. 6.3.1 Section 9.3.

$\Phi M = 226.64\ kip\text{-}in\ < 1.7\ x175.199\ kip\text{-}in = 297.84\ kip\text{-}in$

The duct reinforcement does not have sufficient capacity to withstand the imposed load.

Shear check:
Max shear load: maxQ = 1.7 x 7064.`5 lb = 12.0 kip

Concrete section area: $A = 215.08\ in^2$

Shear capacity for the net section:

$V_c = 2\ \Phi \sqrt{3000}$, Ref. 6.3.1, Section 11.3.1.1

$\Phi = 0.85$, Ref. 6.3.1, Section 9.3.2.3

$V_c = 2\ x\ 0.85\ x\ \sqrt{3000}\ x\ 215.08\ in^2 = 20026.71\ lb = 20.03\ kip < 2x12.0 = 24.0\ kip$
(Ref. 6.3.1, Section 11.3.1.1)

The duct needs protection since it fails in bending and shear.

8.3.2.1. Protection Plate Stress Analysis

Plate cross-section properties are depicted on **Figure 5**:

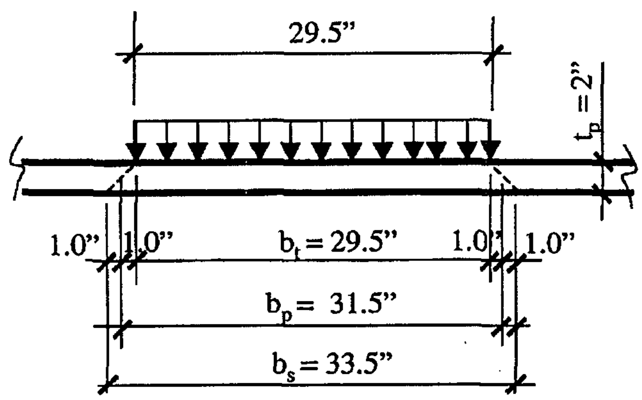


**Figure 5**
Beam within the plate cross-section geometry

Two-way action, i.e. the capacity of the plate to resist the load both ways will be utilized.

Participating load (fraction of the total load p) will be assigned to the beams in both directions, longitudinal and transverse. The amount of the participating load will be determined on the basis of the equal deflection under the load at the point which is characteristic for beams in both directions.

Longitudinally, beam under the Transporter track will be assigned the portion of the load $\alpha$ p.

Transverse beam is loaded under the track width with the $(1-\alpha)$p load. In addition, the rigidity of this beam will be increased to account for the plate action, i.e. the fact that the remainder of the plate prevents the saddle-shaped or anticlastic curvature. The EI term will be increased by a factor $\frac{1}{(1-\upsilon^2)}$ to match the flexural rigidity of the plate. With this in mind, the $\lambda$ coefficient for the Transverse Beam will be calculated.

Deflection under each beam is calculated as:

$$y_c = \frac{q}{2k}\left(2 - D_{\lambda a} - D_{\lambda b}\right) \quad \text{(Ref. 6.1.4 eqn. 7a),}$$

in which the maximum value of $y_c$ occurs under the load.

$A_p = 31.5x2.0 = 63.0\ in^2$

$I_p = 31.5x\dfrac{2.0^3}{12} = 21.0\ in^4,$ $\qquad S_p = 21.0\ in^3$

$b_s = 29.5 + 2x2.0 = 33.5\ in$

k is reasonably conservatively taken as $k = 350\ lb/in^3$

$k_s = b_s\ x\ k = 33.5\ x\ 350 = 11725\ \ lb/in^2$ $\qquad$ Ref. 6.1.4, pg.2, eqn (a)

$$\lambda = \sqrt[4]{\frac{k_s}{4E_sI_p}}$$

$$\lambda = \sqrt[4]{\frac{11725}{4\,x\,29.0\,x\,10^6\,x\,21.0}} = 0.0468\ \frac{1}{in}$$

$\lambda l = 0.0468\ x\ 197'' = 9.23\ > \pi,$ $\therefore$ the beam behaves as a Long Beam (Ref. 6.1.4 , Chapter III, pg. 46 )

As per Ref. 6.1.4, eqn. 7c and 7d respectively:

$M_c = \dfrac{q}{4\lambda^2}(B_{\lambda a} + B_{\lambda b})$, and $Q_c = \dfrac{q}{4\lambda}(C_{\lambda a} + C_{\lambda b})$ where:

$q = 50\ lb/in^2$

$p = 50\ x\ 29.5 = 1475\ lb/in$

Coefficients $D_{\lambda a}$ and $D_{\lambda b}$ are calculated for the longitudinal beam and for the transverse beam/slab.

For Longitudinal beam we have:

$\lambda = 0.0468 \frac{1}{\text{in}}$, $\quad y_c = \frac{\alpha q}{2k}(2 - D_{\lambda a} - D_{\lambda b})$

$\lambda a = 0.0468 \times 197/2 = 4.61 \qquad \lambda b = 0.0468 \times 197/2 = 4.61$

$D_{\lambda a} = -0.0012 = D_{\lambda b}$ (Table 1 of Ref. 6.1.4)

$$y = \frac{\alpha p}{2 \times 11725}[2 - 2x(-0.0012)] = 85.39\text{x } 10^{-6}\, \alpha\, p$$

For Transverse beam $\lambda$ will be assessed with the increased rigidity:

$$\lambda = \sqrt[4]{\frac{k}{4\frac{1}{1-\upsilon^2}E_c I_c}} = \sqrt[4]{\frac{11725}{4\text{x}\frac{1}{1-0.3^2}\text{x}29.0\text{x}10^6\text{x } 21.0}} = 0.0457\, \frac{1}{\text{in}}$$

$$\lambda = 0.0457 \frac{1}{\text{in}}$$

$a = 29.5/2 = 14.75$ in, $\quad b = a = 29.5/2 = 14.75$ in

$\lambda\ a = \lambda\ b = 0.0457 \times 14.75 = 0.674 \quad D_{\lambda a} = D_{\lambda b} = 0.398$ (Ref. 6.1.4, Table I)

$$y = \frac{(1-\alpha)p}{2 \times 11725}(2 - 2 \times 0.398) = 51.343 \times 10^{-6}(1-\alpha)p$$

Therefore: $\qquad 85.39 \times 10^{-6}\, \alpha\, p = 51.343 \times 10^{-6}(1-\alpha)p$

$$(85.39 + 51.343)\alpha = 51.343$$

$$\alpha = 0.375$$

The Longitudinal beam resists the load transferred over the track length. The load intensity is:
$0.375p = 0.375 \times 50.0\ \text{lb/in}^2 \times 29.5\ \text{in} = 553.125\ \text{lb/in}$

The deflection under the beam is $\quad y = 85.39\text{x } 10^{-6}\, \alpha\, p = 85.39\text{x } 10^{-6} \times 553.125 = 0.0472$ in

The maximum pressure onto the road is: $p = yk = 0.0472$ in x 11725 lb/in$^2$ = 553.79 lb/in

This pressure over the width of truck is: $p = 553.79/33.5 = 16.53$ psi

The pressure at 2 feet below the grade, distributed at 30°:

Area: $A_{24"} = l_{24"} \times w_{24"} = 12853$ in$^2$, as determined in Section 8.1

$$P_{T,24"} = 16.53 \times \frac{197x29.5}{A_{24"}} = 16.53x\frac{5811.5}{12853} = 16.53 \times 0.45 = 7.47 \text{ psi} = 1076.26 \text{ lb/ft}^2$$

$p_{T,24"} = 7.47 \times 18.0 = 134.46$ lb/in

Total load on the cover: $q = 35.2 + 134.46 = 169.66$ lb/in

This load is a fraction of the load without the plate. That is: 169.66/438.95 = 0.386 of the load acting on the duct without protection.

Resulting concrete tensile stress: $\sigma = 0.386x\frac{1.7M}{S} = 0.386 \, x\frac{1.7 \times 201534.83}{567.36} = 0.386x \; 603.86 \text{ lb/in}^2 =$ 233.1 lb/in$^2$ < Allowable rupture stress: $_{min}f_r = 8\sqrt{3000} = {}_{min}f_r = 438$ lb/in$^2$ (Reference 6.1.5, Table 2.2, page 49). Based on Ref. 6.1.14, Table 1, Approximate range of Modulus of Rupture for 3000 psi concrete is 450-525 psi.

Shear check:
Max shear load: maxQ = 1.7 x 8126.02 x 0.386 lb = 5.33 kip
$V_c = 2 \times 0.85 \times \sqrt{3000} \times 215.08 \text{ in}^2 = 20026.7 \text{ lb} = 20.0 \text{ kip} > 2x5.33 \text{ kip} = 10.66 \text{ kip}$

Therefore, the duct is adequately protected.

### 8.3.3 Underground Duct Located Along The Travel Route, Shown On Dwg. A207621-8 And A207622-1

(Ref. 6.2.6 and 6.2.7, Section H-H)

The duct cross-section is shown on drawing A207622, Section H-H (Ref. 6.2.7). It is depicted on **Figure 5**:

For $\lambda = 0.013$ 1/in and the load along the total length of the Transporter truck surface with the intensity found at 2' depth: $q = 22.43 \times 21.12 = 473.72$ lb/in
(Section 8.1)

The cross-section forces are computed in the following Table using the same formulae as for the previous duct sections:

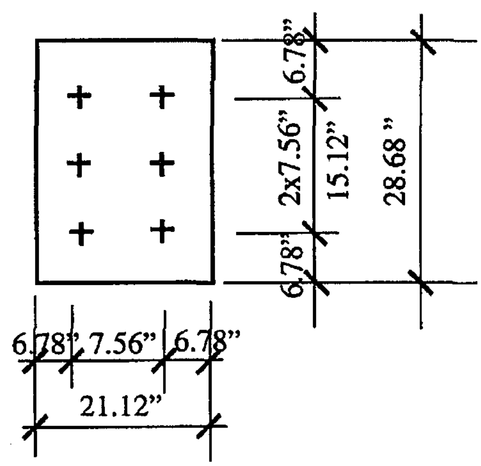


**Figure 6** - Duct cross-section properties

The duct cross-section properties are:

$$A = 21.12 \times 28.68 - 6 \times 5.563^2 \frac{\pi}{4} = 459.89 \text{ in}^2$$

$$y = 28.68/2 = 14.34 \text{ in}$$

$$I_{x,net} = 21.12x\frac{28.68^3}{12} - (0.785x2.7815^4)\ x\ 6\ -2x\frac{5.563^2\pi}{4}x7.56^2\ x2 =$$

$$I_{x,net} = 35680.75\ in^4$$

$$S_x = \frac{I_x}{25.5}x2 = \frac{35680.75x2}{28.68} = 2488.19\ in^3$$

Load on the duct: $H = 2.0'$ (min), Ref. 6.2.7

Earth load: $p_{e'} = 2.0'x120\ lb/ft^3 = 240\ lb/ft^2$
$p_e = 240 \times 21.12/12 = 422.4\ lb/ft = 35.2\ lb/in$

Transporter load: $q = 3.23\ kip/ft^2 = 3230.0/144 = 22.43\ lb/in^2$ (Section 8.1)
$p_t = 22.43x\ 21.12 = 473.72\ lb/in$

Total load on the duct: $q = p_e + p_t = 35.2 + 473.72 = 508.92\ lb/in$

Duct will be evaluated as an infinite beam, loaded along the total length of the Transporter truck surface.

As per Ref. 6.1.4, eqn. 7c: $M_c = \frac{q}{4\lambda^2}(B_{\lambda a} + B_{\lambda b})$, where:

$\lambda = \sqrt[4]{\frac{k_s}{4E_c I_{net}}}$, where $k_s = b \times k = 21.12" \times 400\ pci = 8448\ psi$ Ref. 6.1.4, pg.2, eqn (a)

$$\lambda = \sqrt[4]{\frac{8448}{4 \times 3.15 \times 10^6 \times 35680.75}} = 0.0117\ \frac{1}{in}$$

$\lambda l = 0.0117 \times 197" = 2.3 = 0.73\ \pi < \pi/4 = 0.785$. Therefore, the beam behaves as a minimum as a beam of medium length. The characteristic of this group of beams is that force acting at one end of the beam has a finite, not negligible effect on the forces at the other end. As described in Reference 6.1.4, the forces at one point counter the forces at other points, reducing the total load. Consequently, when the formula for an infinite beam is used to design a beam of medium length, the forces will be very conservatively determined.

| point | a | b | La | Lb | B(La) | B(Lb) | C(La) | C(Lb) | M | Q |
|---|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | | | | | | | (lb/in$^2$) | (lb/in) |
| 1 | 0 | 197 | 0.0000 | 2.3049 | 0.0000 | 0.0741 | 1.0000 | -0.1409 | 68844.44 | **12406.65** |
| 2 | 20 | 177 | 0.2340 | 2.0709 | 0.1835 | 0.1106 | 0.5863 | -0.1711 | 273370.14 | 8236.10 |
| | 29.5 | 167.5 | 0.3452 | 1.9598 | 0.2396 | 0.1304 | 0.4268 | -0.1838 | 343844.98 | 6639.56 |
| 3 | 30 | 167 | 0.3510 | 1.9539 | 0.2421 | 0.1314 | 0.4190 | -0.1844 | 347145.32 | 6561.89 |
| 4 | 40 | 157 | 0.4680 | 1.8369 | 0.2825 | 0.1537 | 0.2764 | -0.1956 | 405425.49 | 5132.78 |
| 5 | 50 | 147 | 0.5850 | 1.7199 | 0.3076 | 0.1771 | 0.1568 | -0.2037 | 450524.29 | 3920.56 |
| 6 | 60 | 137 | 0.7020 | 1.6029 | 0.3200 | 0.2012 | 0.0584 | -0.2077 | 484453.11 | 2893.16 |
| 7 | 70 | 127 | 0.8190 | 1.4859 | 0.3220 | 0.2255 | -0.0209 | -0.2063 | 508887.07 | 2015.54 |
| 8 | 80 | 117 | 0.9360 | 1.3689 | 0.3158 | 0.2492 | -0.0832 | -0.1982 | 525139.33 | 1250.47 |
| 9 | 90 | 107 | 1.0530 | 1.2519 | 0.3032 | 0.2715 | -0.1305 | -0.1819 | 534142.63 | 559.20 |
| 10 | 98.5 | 98.5 | 1.1525 | 1.1525 | 0.2886 | 0.2886 | -0.1603 | -0.1603 | **536510.49** | 0.00 |

**Table 19** - Duct cross-sectional forces

Concrete tensile stress: $\sigma = \dfrac{1.7M}{S} = \dfrac{1.7 \times 536510.49}{2488.19} = 366.56\ \text{lb/in}^2$ < Allowable rupture stress: $_{min}f_r = 8\sqrt{3000} = {}_{min}f_r = 438\ \text{lb/in}^2$ (Reference 6.1.5, Table 2.2, page 49). Based on Ref. 6.1.14, Table 1, Approximate range of Modulus of Rupture for 3000 psi concrete is 450-525 psi.

In addition, the duct is reinforced and the duct reinforcement tension capacity is assessed as follows:

$A_s = 3\text{x}\ 0.2\ \text{in}^2 = 0.6\ \text{in}^2$

$$a = \frac{A_s f_y}{0.85 f'_c b} = \frac{0.6\text{x}40.0\text{x}1000}{0.85\text{x}3000\text{x}21.12} = 0.445\ \text{in}$$

$d = 28.68 - 2.0 = 26.68\ \text{in}$

Reinforcement bending capacity:

$\Phi M = \Phi A_s f_y (d - \frac{a}{2}) = 0.9 \text{ x } 0.6\text{x } 40000 \ (26.68 - \frac{0.445}{2}) = 571475.16 \text{ lb-in} = 571.47 \text{ kip-in}$

$\Phi M = 571.47 \text{ kip-in} < 1.7 \text{ x } 536.51 \text{ kip}$

$\Phi = 0.9$, Ref. 6.3.1, Section 9.3

The section is not sufficiently reinforced, however, the concrete stress is not above the tension limit..

Shear check:

Max shear load: maxQ = 1.7 x 12406.65 = 21091.3 lb/in = 21.09 kip/in

Concrete section area: A = 459.89 $in^2$

Shear capacity for the net section:

$V_c = 2 \ \Phi \sqrt{3000}$, Ref. 6.3.1, Section 11.3.1.1

$\Phi = 0.85$, Ref. 6.3.1, Section 9.3.2.3

$V_c = 2 \text{ x } 0.85 \text{ x } \sqrt{3000} \text{ x } 459.89 \text{ in}^2 = 42821.66 \text{ lb} = 42.82 \text{ kip} > 2\text{x}21.09 = 42.18\text{kip}$

(Ref. 6.3.1, Section 11.3.1.1)

Duct is reinforced with #4 bars.

$Vs + V_c = > 21.09$ kip

Therefore, this duct is adequately reinforced and has sufficient capacity to resist the loads from the Transporter.

### 8.3.4 Underground Duct Located Along The Travel Route, Shown On Dwg. A207621-8 And A207622-1

(Ref. 6.2.6 and 6.2.7, Section N-N)

Depth below the grade: H = 2.0 ft

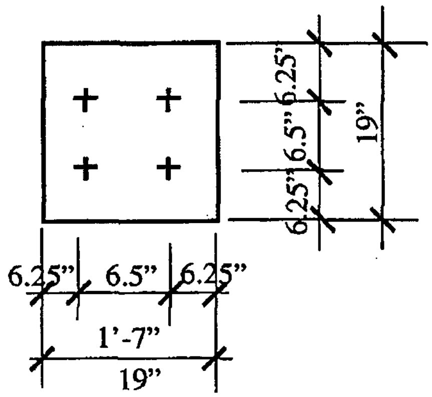


**Figure 7-** Duct cross-section properties

The 4" conduits have outer diameter of: OD = 4.5"

$$A_1 = 19^2 - 4x\ 4.5^2\ \frac{\pi}{4} = 297.38\ in^2$$

$$y = 19.0/2 = 9.5\ in$$

$$I_{x,net} = 19x\frac{19.0^3}{12} - (0.785x2.25^4)\ x\ 4\ -2x\frac{4.5^2\pi}{4}x\ 3.25^2\ x2 = 10860.08\ -80.47\ -671.96 =$$

$$I_{x,net} = 10107.65\ in^4$$

$$S_x = \frac{I_x}{19.0}x2 = \frac{10107.65\ x2}{19.0} = 1064\ in^3$$

Load on the duct: H = 2.0' (min), Ref. 6.2.2

Earth load: $p_{e'} = 2.0' \text{x} 120\ \text{lb/ft}^3 = 240\ \text{lb/ft}^2$
$p_e = 240 \text{ x } 19/12 = 380\ \text{lb/ft} = 31.67\ \text{lb/in}$

Transporter load: $q = 3.23\ \text{kip/ft}^2 = 3230.0/144 = 22.43\ \text{lb/in}^2$, which is found by the conventional 30° load spread method, defined in Section 8.1 of this calculation.

$p_t = 22.43 \text{x } 19 = 426.17\ \text{lb/in}$

Total load on the duct: $q = p_e + p_t = 31.67 + 426.17 = 457.84\ \text{lb/in}$

This load will be placed along the whole affected length of duct, which is taken as
L =2'+ 19/12 +2' = 5.58 ft = 67 in

Duct will be evaluated as an infinite beam, loaded along 5.58 feet of length with the load
q = 457.84 lb/in

As per Ref. 6.1.4, eqn. 7c and 7d respectively:

$M_c = \frac{q}{4\lambda^2}(B_{\lambda a} + B_{\lambda b})$, and $Q_c = \frac{q}{4\lambda}(C_{\lambda a} + C_{\lambda b})$ where:

q = 457.84 lb/in

$\lambda = \sqrt[4]{\frac{k_s}{4E_c I_{net}}}$, where $k_s = b \text{ x } k = 19'' \text{ x } 400\ \text{pci} = 7600\ \text{psi}$  Ref. 6.1.4, pg.2, eqn (a)

$$\lambda = \sqrt[4]{\frac{7600}{4 \text{ x } 3.15 \text{ x } 10^6 \text{ x } 10107.65}} = 0.0156\ \frac{1}{\text{in}}$$

$\lambda l = 0.0156 \text{ x } 67'' = 1.05 > \pi/4 = 0.785$. Therefore, the beam behaves as a beam of medium length. The characteristic of this group of beams is that force acting at one end of the beam has a finite, not negligible effect on the forces at the other end. As described in Reference 6.1.4, the forces at one point counter the forces at other points, reducing the total load. Consequently, when the formula for an infinite beam is used to design a beam of medium length, the forces will be very conservatively determined.

| POINT | a | b | L | La | Lb | B(La) | B(Lb) | C(La) | C(Lb) | M | Q |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | (in) | (in) | (1/in) | | | | | | | (lb/in$^2$) | (lb) |
| 1 | 0.00 | 67.00 | 0.0156 | 0.000 | 1.045 | 0.000 | 0.304 | 1.000 | -0.128 | 143056.82 | **8274.45** |
| 2 | 10.00 | 57.00 | | 0.156 | 0.889 | 0.133 | 0.319 | 0.712 | -0.060 | 212629.52 | 5667.73 |
| 3 | 20.00 | 47.00 | | 0.312 | 0.733 | 0.225 | 0.321 | 0.472 | 0.035 | 256885.27 | 3202.73 |
| 4 | 30.00 | 37.00 | | 0.468 | 0.577 | 0.283 | 0.306 | 0.276 | 0.164 | 276972.04 | 823.85 |
| 5 | 33.50 | 33.50 | | 0.523 | 0.523 | 0.296 | 0.296 | 0.218 | 0.218 | **278413.37** | 0.00 |

**Table 20** - Duct cross-sectional forces

Concrete tensile stress: $\sigma = \dfrac{1.7M}{S} = \dfrac{1.7 \times 278413.37}{1064} = 444.83$ lb/in$^2$ ~ Allowable rupture stress: $_{min}f_r = 8\sqrt{3000}$ = 438 lb/in$^2$ (Reference 6.1.5, Table 2.2, page 49). Based on Ref. 6.1.15, Table 1, Approximate range of Modulus of Rupture for 3000 psi concrete is 450-525 psi.

Shear check:
Max shear load: maxQ = 1.7 x 8274.45 = 14066.56 lb/in = 14.06 kip/in

Concrete section area: $A = 297.38$ in$^2$

Shear capacity for the net section:

$V_c = 2\,\Phi\sqrt{3000}$, Ref. 6.3.1, Section 11.3.1.1

$\Phi = 0.85$, Ref. 6.3.1, Section 9.3.2.3

$V_c = 2 \times 0.85 \times \sqrt{3000} \times 297.38$ in$^2$ = 27689.89 lb = 27.68 kip ≈ 2x14.06 = 28.13 kip
(Ref. 6.3.1, Section 11.3.1.1)
Duct is reinforced with #4 bars, hence: $V_s + V_c >$ 28.13 kip

Therefore, this duct has sufficient capacity to resist the loads from Transporter.

## 8.4 OTHER UNDERGROUND STRUCTURES

### 8.4.1 Pump and Valve Box and Sewer tank

The effect of the Transporter on the pump and valve box shown on Drawing No. A 217469-2 (Ref. 6.2.4) is evaluated based on the following facts:

- The plan showing the distance to the road edge is shown on **Figure 8:**

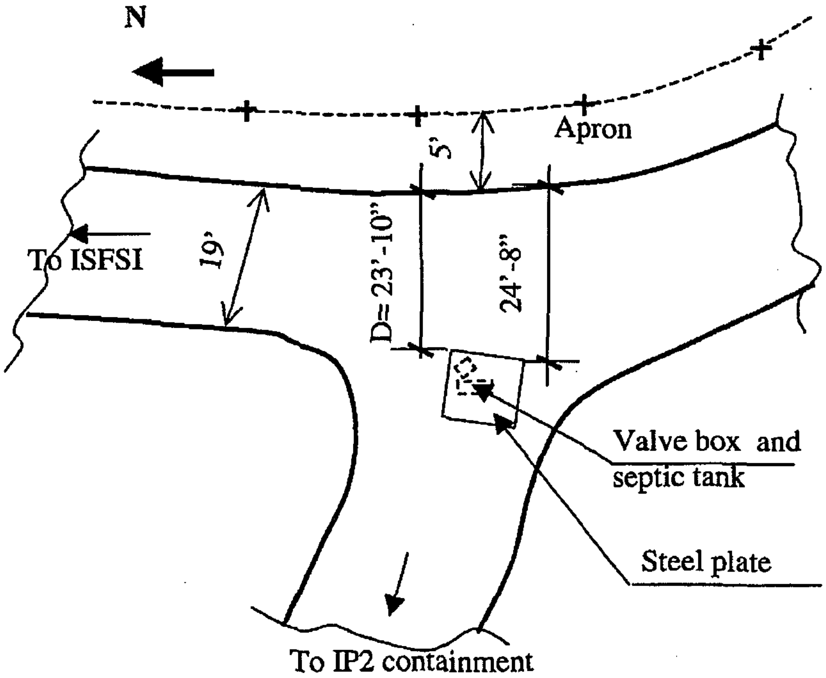


**Figure 8 -** Concrete pump box and sewer tank situation with respect to the road edge

- There is no available drawing that shows the exact distance between the box and the edge of the road. The compilation of the available topographic drawings and drawing Ref. 6.2.4 show that the distance is 23'-5".
- The measurement taken at the field states that the distance D = 25'-10".

Since the Transporter width is 17'-10" (Ref. 6.1.16), we have at least d = (25'-5" - 17'-10") = 5.58' of distance between the edge of road and the concrete box. In this case, from the edge of road to the steel plate, the distance is 25' -10 '-17' -10" = 8 feet. Conservatively, the stress in the soil will be found for the Transporter traveling at the distance of 5 feet from the edge of the box. This is considered as a conservative distance, with regard to both: evaluation of the stresses onto the concrete box and the Transporter situation along the haul path.

The horizontal stress onto the concrete box is found per eqn. 3.35 of Ref. 6.1.9:

$$\sigma_x = \frac{q}{\pi}[\alpha - \sin\alpha \times \cos(\alpha + 2\delta)],$$

where q, $\alpha$, and $\delta$ are parameters shown an **Figure 9**:

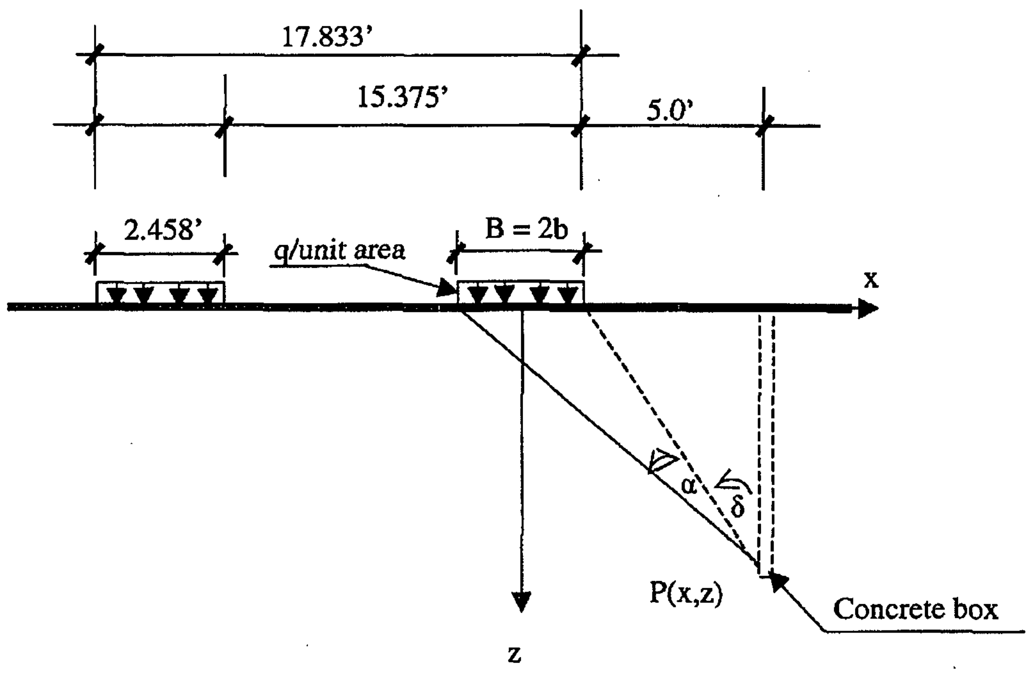


**Figure 9** - Uniform vertical load on an infinite strip

For load q = 50 psi = 7.2 ksf, calculation of the angles along with the final stress is tabulated in **Table 21**.

The concrete wall will be checked for the uniform load q applied over the entire surface.

Bending stress of the plate simply supported at all four edges is:

$\sigma = \dfrac{-\beta q b^2}{t^2}$ (Ref. 6.1.8, Table 26, case 1),

The maximum concrete tension is limited to ${}_{min}f_r = 8\sqrt{3000} = {}_{min}f_r = 438$ lb/in$^2$ (Ref. 6.1.5).
The actual load q will be limited to the intensity that results in a allowable concrete tension stress of 438 lb/in$^2$.

For: b = 4'-0" = 4.00 ft, a = 5.5' , a/b= 5.5/4.0 = 1.375
t = 6.0 " = 0.5 ft
$\beta = 0.443$ (interpolated value from Case No. 1 of Table 26 Ref. 6.1.8)
we have: $\sigma = \dfrac{-\beta q x 4.0^2}{0.5^2} = -64\ \beta q = -64 \text{ x } 0.443\ q = 28.35\ q$

resulting in the value of $q = \dfrac{\sigma}{28.35} = \dfrac{438}{28.35} = 15.44$ lb/in$^2$

This value will be factored down by the Live Load factor of 1.7, resulting in pressure
$q_{all} = 15.44$lb/in$^2$/1.7 = 9.1 lb/in$^2$

Shear Reaction: $R = \gamma q b$, where $\gamma = 0.475$ (interpolated value from Case No. 1 of Ref. 6.1.8)
Shear capacity of the unit length of the concrete unreinforced wall is:
$V_c = 2\ \Phi\sqrt{3000}\ Ac$, Ref. 6.3.1, Section 11.3.1.1 , $\Phi = 0.85$,Ref. 6.3.1, Section 9.3.2.3

$V_c = 2 \text{ x } 0.85 \text{ x } \sqrt{3000} \text{ x } 6.0 = 558.68$ lb/in
q = R/($\gamma$ b) = Vc/ 0.475x(4.0x12) = 558.68/0.475x(48.0) = 24.5 lb/ in$^2$

Therefore, bending requirement governs.
Tabulation of horizontal stresses affecting the concrete box due to the Transporter track loads is presented in **Table 21**:

Stress at point (x,y,z) due to the Transporter load @ 5'+15.375' away:

| point | x | y | z | $R = (x^2 + z^2)^{1/2}$ | q =7.2ksf | q/PI | tanb= 5.0/z | b | tan(a+b) = 5.0+29.5/12)/z | a +b | a | sina | cos(a+2b) | stress |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | (ft) | (ft) | (ft) | (ft) | (kip/ft²) | (kip/ft²) | | rad | | rad | rad | | | (kip/ft²) |
| 1 | 5.00 | 0.00 | 2.00 | 5.39 | 7.20 | 2.2930 | 2.50000 | 1.1903 | 3.7292 | 1.3088 | 0.1185 | 0.1182 | -0.8006 | 0.4888 |
| 2 | 5.00 | 0.00 | 3.00 | 5.83 | 7.20 | 2.2930 | 1.66667 | 1.0304 | 2.4861 | 1.1884 | 0.1580 | 0.1573 | -0.6035 | 0.5800 |
| 3 | 5.00 | 0.00 | 4.00 | 6.40 | 7.20 | 2.2930 | 1.25000 | 0.8961 | 1.8646 | 1.0785 | 0.1825 | 0.1815 | -0.3929 | 0.5819 |
| 4 | 5.00 | 0.00 | 5.00 | 7.07 | 7.20 | 2.2930 | 1.00000 | 0.7854 | 1.4917 | 0.9802 | 0.1948 | 0.1936 | -0.1936 | 0.5327 |
| 5 | 5.00 | 0.00 | 6.00 | 7.81 | 7.20 | 2.2930 | 0.83333 | 0.6947 | 1.2431 | 0.8933 | 0.1986 | 0.1973 | -0.0173 | 0.4632 |

Stress at point (x,y,z) due to the Transporter load @ 5'+15.375' away:

| point | x | y | z | $R = (x^2 + z^2)^{1/2}$ | q =7.2ksf | q/PI | tanb = (5.0+15.375)/z | b | tan(a+b) = 5.0+17.833/z | a +b | a | sina | cos(a+2b) | stress |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | (ft) | (ft) | (ft) | (ft) | (kip/ft²) | (kip/ft²) | | rad | | rad | rad | | | (kip/ft²) |
| 1 | 20.38 | 0.00 | 2.00 | 20.47 | 7.20 | 2.2930 | 10.18750 | 1.4730 | 11.4165 | 1.4834 | 0.0105 | 0.0105 | -0.9829 | 0.0476 |
| 2 | 20.38 | 0.00 | 3.00 | 20.59 | 7.20 | 2.2930 | 6.79167 | 1.4246 | 7.6110 | 1.4402 | 0.0155 | 0.0155 | -0.9619 | 0.0699 |
| 3 | 20.38 | 0.00 | 4.00 | 20.76 | 7.20 | 2.2930 | 5.09375 | 1.3769 | 5.7083 | 1.3974 | 0.0204 | 0.0204 | -0.9333 | 0.0906 |
| 4 | 20.38 | 0.00 | 5.00 | 20.98 | 7.20 | 2.2930 | 4.07500 | 1.3302 | 4.5666 | 1.3552 | 0.0251 | 0.0251 | -0.8977 | 0.1091 |
| 5 | 20.38 | 0.00 | 6.00 | 21.24 | 7.20 | 2.2930 | 3.39583 | 1.2844 | 3.8055 | 1.3138 | 0.0294 | 0.0294 | -0.8560 | 0.1252 |

**Table 21** - Horizontal stresses at the point due to vertical load on an infinite strip

Total stress at the point due to the load from both tracks is summarized in **Table 22**:

| point | stress | stress | sum |
|---|---|---|---|
| | (kip/ft2) | (kip/ft2) | (kip/ft2) |
| 4 | 0.4888 | 0.0476 | 0.54 |
| 1 | 0.58 | 0.0699 | 0.65 |
| 2 | 0.5819 | 0.0906 | 0.67 |
| 3 | 0.5327 | 0.1091 | 0.64 |
| 5 | 0.4632 | 0.1252 | 0.59 |

**Table 22** - Sum of stresses due to both Transporter track loads

Maximum horizontal load affecting the concrete box due to the Live load is found to be $q = 0.67$ kip/ft$^2$

Horizontal stress due to the soil overburden:
$q_s = \gamma H k = 0.12 \times 5.0 \times 0.35 = 0.21$ ksf
(k = 0.35 , conservatively taken soil coefficient, Ref. 6.1.5, Chapter 19)

$q_t = q + q_s = 0.67 \text{ kip/ft}^2 + 0.21 = 0.88 \text{ kip/ft}^2 = 0.88 \times 1000/144 \text{ lb/in}^2 = 6.11 \text{ lb/in}^2 < q_{all} = 9.1 \text{ lb/in}^2$

Therefore, a distance of 5 feet between the Transporter and the Pump and Valve Box should be maintained.

### 8.4.2 Utility Tunnel

Based on Ref. 6.2.1, Attachment A, the tunnel ceiling is located at 12' below the grade.

Vertical load on the tunnel is found from **Table 2** as: $p_v = 1.15 \text{ kip/ft}^2$ for the depth of 12':

Total vertical load on the tunnel ceiling is:
$W_u = 1.4(1.5 x .15 + 0.12 x 12) + 1.7 \times 1.15 =$
$W_u = 2.38 + 1.955 = 4.338 \text{ kip/ft}^2 = 4.34 \text{ kip/ft}^2$

Bending moment for the tunnel ceiling dimensions defined on pg. 31 of Attachment A to Ref. 6.2.1:
Conservatively assessed: $M = W_u \times l^2/8 = 4.34 \times 9.0^2/8 = 43.94$ kip-ft

Bending capacity: M = 96.8 ft-kip ( Ref. 6.2.1) > 43.94 kip-ft

This bending capacity is based on the actual reinforcement consisting of #9 @ 6". Therefore, the reinforcing ratio is $\rho = 2.0/12x17.4375 = 0.009$. This reinforcement ratio is almost two times higher than the minimum used in the Westinghouse calculation.

Hence, the concrete top has adequate bending capacity to sustain the applied load (mostly the weight of soil and the own weight; at this depth the surcharge load is less than the soil weight)

For d = 17.4375 in (Ref. 6.2.1) we have:

Shear load: Vu = 4.34 x 8/2 - 4.34 x 17.4375/12 = 17.36 - 6.306 = 11.05 kip/ft

Shear capacity: $V = 2x0.8x\sqrt{3000}\,x\ 12x17.4375 = 24.59$ kip

The top of the tunnel has adequate capacity to sustain the applied load.

Tunnel walls analysis:

Analysis for the configuration shown on **Figure 9** is applicable. However, it will conservatively be taken that the Transporter travels along the edge of the Utility tunnel wall, hence, x = 0.

| point | x | y | z | $R = (x^2 + z^2)^{1/2}$ | q =7.2ksf | q/PI | tanb= 0.0/z | b | tan(a+b) =(29.5/12)/z | a +b | a | sina | cos(a+2 b) | stress |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | (ft) | (ft) | (ft) | (ft) | (kip/ft²) | (kip/ft²) | | rad | | rad | rad | | | (kip/ft²) |
| 0 | 0.000 | 0.000 | 1.000 | 1.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 2.4583 | 1.1845 | 1.1845 | 0.9263 | 0.3768 | 1.9156 |
| 1 | 0.000 | 0.000 | 2.000 | 2.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 1.2292 | 0.8878 | 0.8878 | 0.7757 | 0.6311 | 0.9133 |
| 2 | 0.000 | 0.000 | 3.000 | 3.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.8194 | 0.6865 | 0.6865 | 0.6338 | 0.7735 | 0.4500 |
| 3 | 0.000 | 0.000 | 4.000 | 4.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.6146 | 0.5511 | 0.5511 | 0.5236 | 0.8520 | 0.2407 |
| 4 | 0.000 | 0.000 | 5.000 | 5.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.4917 | 0.4570 | 0.4570 | 0.4412 | 0.8974 | 0.1399 |
| 5 | 0.000 | 0.000 | 6.000 | 6.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.4097 | 0.3889 | 0.3889 | 0.3791 | 0.9253 | 0.0872 |
| 6 | 0.000 | 0.000 | 7.000 | 7.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.3512 | 0.3377 | 0.3377 | 0.3314 | 0.9435 | 0.0576 |
| 7 | 0.000 | 0.000 | 8.000 | 8.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.3073 | 0.2981 | 0.2981 | 0.2937 | 0.9559 | 0.0398 |
| 8 | 0.000 | 0.000 | 9.000 | 9.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.2731 | 0.2666 | 0.2666 | 0.2635 | 0.9647 | 0.0286 |
| 9 | 0.000 | 0.000 | 10.000 | 10.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.2458 | 0.2411 | 0.2411 | 0.2387 | 0.9711 | 0.0212 |
| 10 | 0.000 | 0.000 | 11.000 | 11.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.2235 | 0.2199 | 0.2199 | 0.2181 | 0.9759 | 0.0161 |
| 11 | 0.000 | 0.000 | 12.000 | 12.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.2049 | 0.2021 | 0.2021 | 0.2007 | 0.9797 | 0.0125 |
| 12 | 0.000 | 0.000 | 13.000 | 13.000 | 7.20 | 2.2930 | 0.00000 | 0.0000 | 0.1891 | 0.1869 | 0.1869 | 0.1858 | 0.9826 | 0.0099 |

**Table 23** - Total horizontal stress at point

**Table 23** shows that the horizontal pressure due to the surcharge load at the depth of twelve feet is negligible. Consequently, the results of analysis conducted in Ref. 6.2.1, Attachment A, pg. 32 and 33 are applicable and the tunnel is adequate for the applied loads.

### 8.4.3 Manholes

8.4.3.1 THE MANHOLE POSITIONED AT THE ROAD SECTION LEADING FROM THE IP3 FSB TOWARD THE SECURITY GATE

The edge of the manhole cover is positioned 5' away from the edge of the road (measured at walkdown).

The position of the manhole with respect to the road-edge is depicted on **Figure 10.**

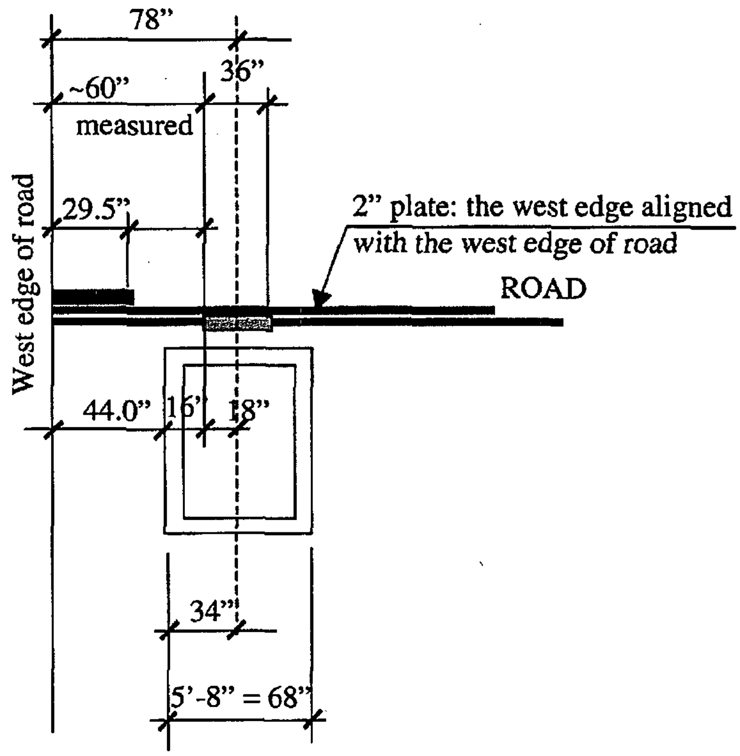


**Figure 10** - Manhole cover location with respect to the edge of the existing road

The distance from the edge of the manhole cover to the west edge of the road is found to be 5'.

Based on the road dimension, the present configuration of the Transporter path leads over the manhole structure.

Per Ref. 6.2.1, Attachment A, pg. 41 and 49, we have the following conclusions applicable for the manhole cover:

- The depth below the grade is 1.5' ( H = 18")
- The cover reinforcement bending capacity is M = 11.2 ft-kip/ft

The pressure will be checked at depth h = 1.5 ft = 18" under the surface, with 2" plate engaged in the load spread using one-way capacity (The ratio of long to short dimension is: a/b = 12'-9"/5'-8"= 12.75/5.67 =2.25 > 2.0, therefore, the load is carried in short direction):

Area:

$l_{18"} = l + 2xh \tan30° = 197 + 2x18x \tan30° = 217.78$ in

$w_{18"} = w + 2xh \tan30° = 29.5 + 2x18x \tan30° = 50.28$ in

$A_{18"} = l_{18"} \times w_{18"} = 217.78 \times 50.28 = 10950.98$ in$^2$

The cover width is 4'-6" = 54" (Ref. 6.2.1, Attachment A, pg. 47). Note that the width of cover dimension is obtained from the scaled drawing. Obviously, the width of the load spread matches the width of cover.

Weight of soil: $p_e = 1.5x120 = 180.0$ lb/ft$^2$

Cover own weight: $DW = 150x\frac{8}{12} = 100$ lb/ft$^2$

Check the pressure at the grade due to the 2" plate load distribution:

p = 16.53 psi as determined in Section 8.3.2.1.

This pressure over the width of truck is: p = 16.53 psi

The pressure at 18" below the grade, distributed at 30°:

Area:

$l_{18"} = l + 2xh \tan30° = 197 + 2x18x \tan30° = 217.78$ in

$w_{18"} = w + 2xh \tan 30° = 29.5 + 2x18x \tan 30° = 50.28\ in$

$A_{18"} = l_{18"} \times w_{18"} = 217.78 \times 50.28 = 10950.98\ in^2$

The cover width is 4'-6" = 54" (Ref. 6.2.1, Attachment A, pg. 47). Note that the width of cover dimension is obtained from the scaled drawing. (Obviously, the width of the load spread matches the width of cover.)

$$p_{18"} = 16.53 \times \frac{197x29.5}{A_{18"}} = 16.53 \frac{5811.5}{10950.98} = 8.77 \text{ psi} = 1263.2\ \text{lb/ft}^2$$

Total load on the cover: $l = 1.4x100 + 1.7(1263.2 + 180) = 2593.44\ \text{lb/ft}^2$

This load acting over the width of cover results in a bending moment of:

$$M = \frac{2593.44\ x4.5^2}{8} = 6564.65 \text{ lb-ft} = 6.56 \text{ kip-ft}$$

Based on Dwg. A2207622-1 (Ref. 6.2.7) and Ref. 6.2.1, Attachment A, pg. 49, the reinforcement capacity of the concrete cover is M = 11.2 kip-ft/ft.

The manhole cover can withstand the load imposed by the transporter when it is covered with the 2" thick plate.

Determine the distance from the cover to the edge of the plate:

Semi-infinite condition, i.e. the condition when Transporter travels at the edge of plate:

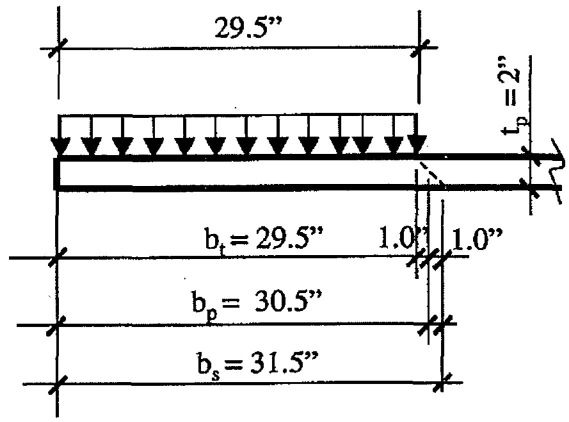


**Figure 11** - Semi-infinite plate condition

Longitudinal beam cross-section properties:

The beam width: $b_p = 29.5 + 1.0 = 30.5$ in

The width of the engaged soil: $b_s = 29.5 + 2.0 = 31.5$ in

$k = b_s \times k_0 = 31.5 \times 350 = 11025$ psi

$$I_c = \frac{b_p x h^3}{12} = \frac{30.5x2.0^3}{12} = 20.33 \text{ in}^4 \qquad S_p = 20.33 \text{ in}^3$$

$$\lambda = \sqrt[4]{\frac{k}{4E\ I_p}} = \sqrt[4]{\frac{11025}{4x29x10^6 x\ 20.33}} = 0.0465 \text{ 1/in}$$

For say 5' long plate we have:

$\lambda l = 0.0465x\ 60 = 2.79 < \pi$. If the plate is 5' long, the analysis based on the Long Beam behavior is conservative, for plates more than 5' ($\lambda l > \pi$), it is adequate.

Deflection for this beam is found by the methodology described above, for the one-way plate.

In the mid-point (a = 98.5 in, b = 98.5 in): $\lambda a = \lambda b = 0.0465 \times 98.5 = 4.58$
From Table I of 6.1.4,: $D_{\lambda a} = D_{\lambda b} = -0.0014$

The deflection equation with the consideration for the load distributed onto the Transverse beam is:

$$y_L = \frac{\alpha q}{2k}(2 - D_{\lambda a} - D_{\lambda b}) = \frac{\alpha p x 29.5}{2x11025}(2 + 2x0.0014) = 2679.48 \times 10^{-6} \times \alpha p$$

The Transverse edge beam has the cross-sectional properties determined earlier in this Section:

$$\lambda = 0.0457 \frac{1}{in}, \qquad k = 11725 \text{ lb/in}^2$$

The beam deflection is found as:

$$y = \frac{2P\lambda}{k} D_{\lambda a}, \quad \text{Ref. 6.1.4, eq. 19a}$$

where:

$P = (1-\alpha)\ p \times 29.5\text{"} \times 29.5\text{"} = 870.25 \times (1-\alpha)\ p$
$D_{\lambda a} = 1.0$ for the edge condition with a = 0 (Table I of Ref. 6.1.4)

$$y_T = \frac{2x870.25(1-\alpha)px0.0457}{11725} \times 1.0 = 6783.87 \times 10^{6} (1-\alpha)\ p$$

The condition is represented on **Figure 12**:

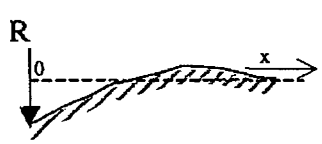


**Figure 12** - End loading of the Semi-Infinite Beam

Deflection "y" will be calculated for both beams, and the coefficient $\alpha$ which distributes the load according to the beam stiffness, will be determined:

$$y_L = 2679.48 \times 10^{-6} \times \alpha p = 6783.87 \times 10^{-6} (1-\alpha) p$$

$$2679.48 \times \alpha p = 6783.87 \times (1-\alpha) p, \qquad \alpha = 0.717$$

Deflection at the track mid-point:

$$y_T = 6783.87 \times 10^{-6} \times (1-\alpha) p = 6783.87 \times 10^{-6} \times (1-0.717) \times 50.0 = 0.096 \text{ in}$$

This deflection results in a pressure p' = 0.096xk = 0.096x 11725 = 1125.6 lb/in

Pressure over the length of beam: p = 1125.6/33.5 = 33.6 psi

$$p_{18''} = \frac{(197 + 2x2.0)x33.5}{10950.98} x\ 33.6 = 20.66 \text{ psi} = 2975.01 \text{ lb/ft}^2$$

Weight of soil: $p_e = 1.5x120 = 180 \text{ lb/ft}^2$

Cover own weight: $DW = 150 \times \frac{8}{12} = 100 \text{ lb/ft}^2$

Total load on the cover: $l = 1.4x100 + 1.7(2975.01+180) = 5503.53 \text{ lb/ft}^2$

This load acting over the width of cover results in a bending moment of:

$$M = \frac{5503.53x4.5^2}{8} = 13930.8 \text{ lb-ft} = 13.9 \text{ kip-ft} > M_n = 11.2 \text{ kip-ft.}$$

This means that the manhole top cover edge can not be aligned with the edge of plate.

The plate should be positioned in such a way that the Transporter travels at a distance "d" away from the edge of plate. That distance is determined in a following way:

The acceptable deflection (i.e. the deflection which results in a pressure that the cover can withstand) is found to be y = 0.05 in ( It has been shown that for deflection of 0.048 in the cover is adequate)

The distance at which the Transverse beam deflects this much is found as:

$$0.05 = \frac{2P\lambda}{k} D_{\lambda a} = 2x\frac{870.25 \text{ x } (1-\alpha) \text{ px}0.0457}{11725} D_{\lambda a} = 2x\frac{870.25x(1-0.717)x50.0x0.0457}{11725} x D_{\lambda a}$$

$D_{\lambda a} = 0.05/0.09599 = 0.521$. From Table I of Ref. 6.1.4 we have that $\lambda a = 1.94$

$a = 1.94/\lambda = 1.94/0.0457 = 42.45 \text{ in} = 3.54 \text{ ft}$

Investigation at the location of electrical manhole should be performed in order to determine the manhole orientation with regard to the haul path.

The worst possible case anticipated is shown on **Figure 13**:

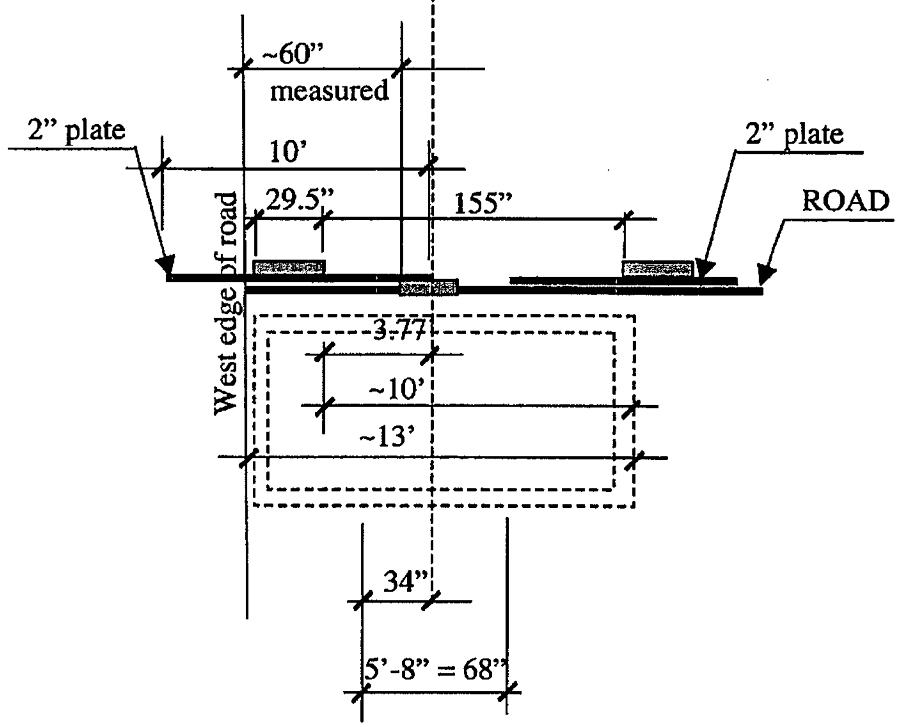


**Figure 13 -** Potential manhole position with respect to the haul road

The Transporter travel path should be placed a minimum of 3.5 feet away from the edge of the 2" steel plate.

If the manhole top cover length is positioned in a way that the other Transporter track is positioned above the wall, the 2" plate should be positioned under the other track surface for protection of the manhole cover. The cover plate should cover 3.5 feet beyond each of the Transporter track surfaces.

### 8.4.3.2 OTHER MANHOLES

Typical manhole/catchbasin wall (Ref. 6.2.1, Attachment A, pg. 41- pg. 44) has been evaluated for the horizontal pressure of the intensity of 1.0 kip/ft$^2$. The analysis concludes that the walls are adequate for this load.

For the case of Transporter traveling over the plates, walls will be exposed to the load intensity found in **Table 24.** The pressure onto the ground under the 2" plate is found to be:

p = 16.74 psi = 2.41 kip/ft$^2$

| point | x | y | z | R | q | q/PI | tanb= 0.0/z | b | tan(a+b) =(29.5/12)/z | a +b | a | sina | cos(a+2b) | stress |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | (ft) | (ft) | (ft) | (ft) | (kip/ft$^2$) | (kip/ft$^2$) | | rad | | rad | rad | | | (kip/ft$^2$) |
| 0 | 0.00 | 0.00 | 1.00 | 1.00 | 2.410 | 0.768 | 0.000 | 0.000 | 2.458 | 1.184 | 1.184 | 0.926 | 0.377 | 0.641 |
| 1 | 0.00 | 0.00 | 2.00 | 2.00 | 2.410 | 0.768 | 0.000 | 0.000 | 1.229 | 0.888 | 0.888 | 0.776 | 0.631 | 0.306 |
| 2 | 0.00 | 0.00 | 3.00 | 3.00 | 2.410 | 0.768 | 0.000 | 0.000 | 0.819 | 0.686 | 0.686 | 0.634 | 0.773 | 0.151 |
| 3 | 0.00 | 0.00 | 4.00 | 4.00 | 2.410 | 0.768 | 0.000 | 0.000 | 0.615 | 0.551 | 0.551 | 0.524 | 0.852 | 0.081 |
| 4 | 0.00 | 0.00 | 5.00 | 5.00 | 2.410 | 0.768 | 0.000 | 0.000 | 0.492 | 0.457 | 0.457 | 0.441 | 0.897 | 0.047 |
| 5 | 0.00 | 0.00 | 6.00 | 6.00 | 2.410 | 0.768 | 0.000 | 0.000 | 0.410 | 0.389 | 0.389 | 0.379 | 0.925 | 0.029 |
| 6 | 0.00 | 0.00 | 7.00 | 7.00 | 2.410 | 0.768 | 0.000 | 0.000 | 0.351 | 0.338 | 0.338 | 0.331 | 0.944 | 0.019 |

**Table 24** - Horizontal pressure under the 2" plate

The gate width has been found to be about 17'. The transporter truck outer edge dimension is 17'-10". Hence, the gate is not functional in this application and should be modified, i. e. expanded. However, due to the wider space for the Transporter, it is anticipated that the transporter will cross over the existing plates. Hence, the top cover is evaluated for the load due to the Transporter crossing over the manholes

The wall bearing capacity is: $\sigma = \Phi x 0.85 f_c^{'}$ ( Ref. 6.3.1, eqn. 10.15.1)
$\Phi = 0.7$,
$\sigma = 0.7x0.85x3000 = 1785$ psi
For 6" thick wall: $R_n = 1785x6 = 10710$ lb/in = 10.71 kip/in

Reaction R from 4-edge supported plate is: $R = \gamma qb$, (Ref. 6.1.8, Table 26,1) where $\gamma = 0.428$
Conservatively take that b = 48" on all four sides:

$R = 1.7x\ 0.428\ x\ 50x\ (48) = 1.7x1027.2 = 1746.24$ lb/in = 1.75 kip/in $< R_n < 10.71$ kip/in

Design axial load strength of the wall is found by empirical design method, Ref. 6.3.1 eqn. 14-1:

$$\Phi P_{nw} = 0.55\Phi\ f_c^{'} A_g \left[1-\left(\frac{kl_c}{32h}\right)^2\right]$$

where: $A_g = 12.0 \text{ x } b = 12.0 \text{ x } 6.0 = 72.0 \text{ in}^2$
$k = 0.8$, Ref. 6.3.1, Section 14.5.2
$l_c$ = say 4' = 48" as a minimum, = 48"
h = 6"overall thickness of member
$\Phi = 0.7$, Ref. 6.3.1, Section 9.3.2.2

$$P_{nw} = 0.55x0.7x3000x72.0\ x\left[1-\left(\frac{0.8x48}{32x6.0}\right)^2\right] = 0.55\ x\ 0.70x\ 3000x72.0x\ 0.96 = 79833.6 \text{ lb} = 79.83 \text{ kip/ft}$$

$P_{nw} = 6.65$ kip/in
$P_{nw} = 6.65$ kip/in $> R = 1.75$ kip/in

In the case of deteriorated manhole cover, the concrete wall can withstand the load from the Transporter, when covered with the 2" plate. Please refer to the next Section for the 2" plate bending evaluation.

### 8.4.4.1 Drain Inlets

DI #9 (Ref. 6.2.10)

Drawing No. 9321-F-40143: Yard Storm Drains Sections and Details (Ref. 6.2.9) provides the information regarding the width of the manholes. Based on the provided Table, the biggest manhole dimension is 48"x48". The walls are placed under the layer of bricks. Based on Ref. 6.1.20, Table 1 of Section 15, the brick compressive strength is at least 1250 psi.

The cover of the inlet is deeply recessed (at least 2 inches). The recess dimension is 4' both ways. Therefore, the 2" plate will literally bridge the cover of the inlet.

Bending stress of the plate is found to be: $\sigma = \frac{\beta q b}{t^2}$ (Ref. 6.1.8, Table 26, case 1)

Where $\beta = 0.2874$ (for a/b = 1.0).

$$\sigma = \frac{0.2874 x 50.0 x 48^2}{2^2} = 8277.12 \text{ lb/in}^2 = 8.28 \text{ ksi} < \text{bending capacity: } {}_{all}\sigma = 27.0 \text{ ksi}$$

Plate deflection: ${}_{max}y = \frac{\alpha q b^4}{E t^3}$ (Ref. 6.1.8, Table 26, case 1)

$\alpha = 0.0444$ $\qquad {}_{max}y = \frac{\alpha q b^4}{E t^3} = \frac{0.0444 x 50.0 x 48.0^4}{29 x 10^6 x 2.0^3} = 0.05 \text{ in}$

The bricklayer is placed on a top of the concrete walls. The brick compressive strength of minimum of 1250 psi is adequate to resist the pressure from the top.

Side loads: For 1.92 ksf side pressure (**Table 23**) and the 4' span of the wall, the maximum bending moment is: $q = 1.4x(1.92 + \underbrace{130x1.0x0.35/1000}_{\text{soil pressure}}) = 1.4\,x(1.92+0.046) = 2.75 \text{ kip/ft}^2$

$M = 2.75 \text{ x } 4.0^2/8 = 5.5 \text{ kip-ft}$

The #6 bars @9" capacity is:

$As = 0.59\ in^2/ft$

$$a = \frac{A_s f_y}{0.85fc'b}$$ (Ref. 6.1.5, eqn. 3.3.1)

$$a = \frac{0.59x40000}{0.85x3000x12} = 0.77\ in$$

For 8" *concrete wall, conservatively determined d is* $d = 8.0 - 3.0 - 0.75/2 = 4.6$ *in*

Bending capacity: $Mn = A_s f_y(d-\frac{a}{2}) = 0.59 \text{ x } 40000x(4.6-0.77/2) = 99813.34$ lb-in = 8.3 kip-ft > Mn = 8.3 kip-f > 5.5 kip-ft

The manhole wall can withstand the side pressure imposed by the Transporter.

*DI #10* (Ref. 6.2.10)

The 2" plate will bridge the inlet cover. However, since the inlet protrudes above the road surface, the road will be grouted in such a way to fill the gap in the road and to provide a level surface for the plate. In addition, the grout for the platform for the plate should have a 0.25 inch distance to the top of inlet cover (max. deflection as determined above is $_{max}y = 0.05$ in).

## 8.5 Haul Road Evaluation

The road section shown as a "Typical Road Bed Construction" detail on Drawing No. 9321-F-1004-1: Plan of Entrance Roads, Units No. 1, 2 &3" (Ref. 6.2.5) has been evaluated for the loads imposed by the Steam Generator Transporter and Prime Mover. These loads are of lower intensity than the bearing pressure of the Transporter, determined as 50 psi. (The shadow ground load, specified on pg. 1 of Attachment B to the Calculation SGRP-C-003, Rev. 0 (Ref. 6.2.1) is 1147 lb/ft$^2$ = 7.96 psi)

The Army Corps of Engineers Manual EM-1110-3-141: Airfield Flexible Pavement Mobilization Construction (Ref. 6.1.18) has served as the guiding document to develop the engineering and design basis for the evaluation of the road. To address the issues of frost susceptibility The Pavement Criteria for Seasonal Frost Conditions, Manual EM-1110-3-138 (Ref. 6.1.17) was used.

The methodology developed in these references provides adequate thickness of quality pavement components above the sub-grade to prevent detrimental sub-grade deformation, excessive deflection of the pavement surface and excessive tensile strain in the bituminous pavement material under traffic.

The thickness design procedures for conventional flexible pavement construction, presented in Ref. 6.1.17 and 6.1.18 are based on CBR design method developed for airfields.

Based on Ref. 6.2.1, Attachment A, pg. 66, the sub-grade soil near the surface is primarily medium to fine send with traces of silt and gravel. Based on Table 2-1 of Ref. 6.1.17, sandy soil is classified as S1 to S2 group. Therefore, this type of soil has very low to medium frost-susceptibility and even is considered suitable for subbase material.
Based on Table 3-1 of Ref. 6.1.19, the range of CBR values for sand - gravel mixes is not lower than 10, and could be as high as 40. This type of soil is not frost susceptible and will not retain enough moisture to cause significant frost heave and thaw weakening. In addition,
since the road is relatively narrow (portions are only 19 feet wide), and since good drainage conditions are utilized (minimum12 inches deep side ditches), less intense ice segregation is present.

For the case where frost penetration conditions are not applicable (non-frost design), based on Ref. 6.1.18, Figure 7-6-b: Flexible Pavement Design Curves, Air Force Heavy-Load Pavement, Types B, C, and D Traffic Areas and Overruns, we have that the sub-grade of CBR = 10 results in the pavement of 24 inches. Sub-grade of CBR of 40 would give the value of 8.5 in, and sub-grade of 20 would require the total thickness of 13 inches of pavement. The pavement total thickness, based on Ref. 6.2.5 and 6.2.1, Attachment A, pg. 68, is:
Minimum surface thickness (Table 5-2 of Ref. 6.1.18):

- 3" with 100 CBR base
- 6" of base thickness

These conditions are fulfilled since we have a total of (2"+3" =) 5" of top layer over a 6 inches thick layer of granular stone base course, classified as CBR =100.

With 6" of subbase layer, the total thickness is: p = 5"+ 6"+ 6"= 17", what is adequate for the load and for the subgrade of CBR of about 15 (Figure 7-6b of Ref. 6.1.18). This value of the subgrade CBR is most likely exceeded. Hence, the thickness of the road fulfills the more rigorous subgrade conditions.

The standard road building and design practice requires the following:

- all stones more than 6 inches in diameter are removed (to prevent boulder heaves from damaging the pavement)
- minimum of 12 inches of top soil material is removed (confirmed by Ref. 6.2.4),
- minimum of four inches of sub-grade are compacted .

With addition of the service history of the road as good (no history of work on the road is readily available), we have the basis for the favorable judgment about the capacity of the portion of the existing road that will be utilized during the fuel transfer. Based on the present condition and the past servicing capacity, the road has been evaluated as adequate for this task.

## 9.0 CONCLUSION

The following items need to be modified/protected:

9.1 The present Security Gate needs to be modified to reflect the Transporter dimensions. The present Gate width is not sufficient to allow the Transporter to pass through. In addition, the Gate height is not adequate for the height of the cask. These facts result in a need to modify the Security Gate.

9.2 The electrical manhole cover can not withstand the load imposed by the Transporter. Hence the total area of the manhole cover should be covered with the 2" thick plate.
Investigation at the location of each manhole should be performed in order to determine how the manhole is positioned with regard to the haul path.
The Transporter should travel along the middle of the 10' wide plate. That means that 3.5 feet distance should exist between the edge of plate and the Transporter. Two plates should be placed, each under one track surface.

9.3 The duct shown on Section J-J of Ref. 6.2.6 with 5" PVC conduits needs protection along the affected length of the haul route. The protection is 2" plate placed at the grade above the duct run.

9.4 All manholes presently covered and Drain inlets should be covered with 2" thick plates. In addition, Drain inlet DI-10 should have a road surface grouted to achieve the level plate position. as well as 0.25-inch plate distance to the top of the manhole cover.

Necessary modifications for the haul path lay-out based on presently assessable road are depicted on **Figure 14:**

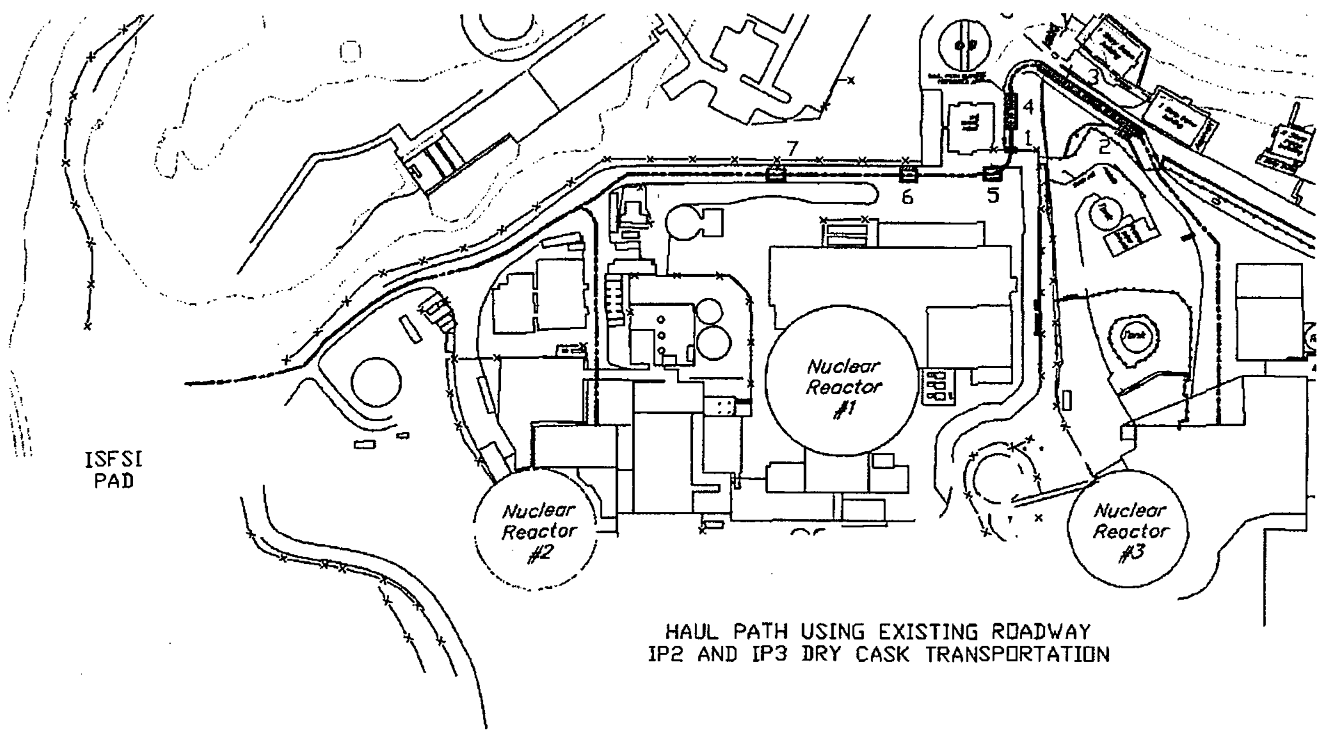


**Figure 14**
Holtec Transporter Haul Path based on existing road lay-out

LEGEND of necessary modifications for the haul path depicted on **Figure 14**:

1 Security Gate modification in order to accommodate the Transporter dimensions

2 Electrical manhole covered with two 10'x10'x2" plates, placed in such a way that 3.5 feet of plate exist form the edge of the Transporter track

3 Duct along the route covered with 2" steel plates.

4 Manholes in front of the Security Gate, marked as MH-1, MH-2, MH-3 on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10). All covered with 2" plates.

5 DI-7 Drain Inlet marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates.

6 DI-9 marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates.

7 DI-10 marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates. Grout road surface around the opening with high strength grout (Five Star Grout or similar).

**Recommendations:**

The route shown on Steam Generator Haul Route & Upgrades Drawing (Ref. 6.2.10) reflects section of the current haul road lay-out. The integrated IP2 and IP3 Haul Path is depicted on **Figure 14.**

The need to modify the Security Gate for clearance to enable the Transporter to pass through is identified in this calculation. This information should be transmitted for input to the Security modification upgrade, making known that fuel can not be transmitted until the Gate Modification is complete.

The following Plant security modifications are recommended:

**a)** Removing the fence presently positioned along the West edge of the IP3 road to the East Side of the road. This would result in a possibility to **b)** lay-out the haul road as shown on **Figure 15.** With this proposed lay-out the following protection identified as necessary for the present road lay-out, would not be needed:

- Electrical manhole would not be crossed-over by the Transporter since it would not be located along the haul path. Hence, the 2" plates covering the road will not be needed.

- Underground duct with the PVC pipes would not be located along the haul path in the full length. Potentially, only one plate placed at the start of the future haul path curvature would resolve the overstressed problem.

- The manhole West from the present Security Gate would not need protection.

The present road configuration with the sharp turns before and after the Security Gate is judged to be very challenging for the loaded Transporter. The proposed route has a smooth curvature that is not a challenge for this vehicle.

Hence, the Plant Security Upgrade in coordination with the ISFSI Project would contribute in a mutually satisfactory solution with less interference.

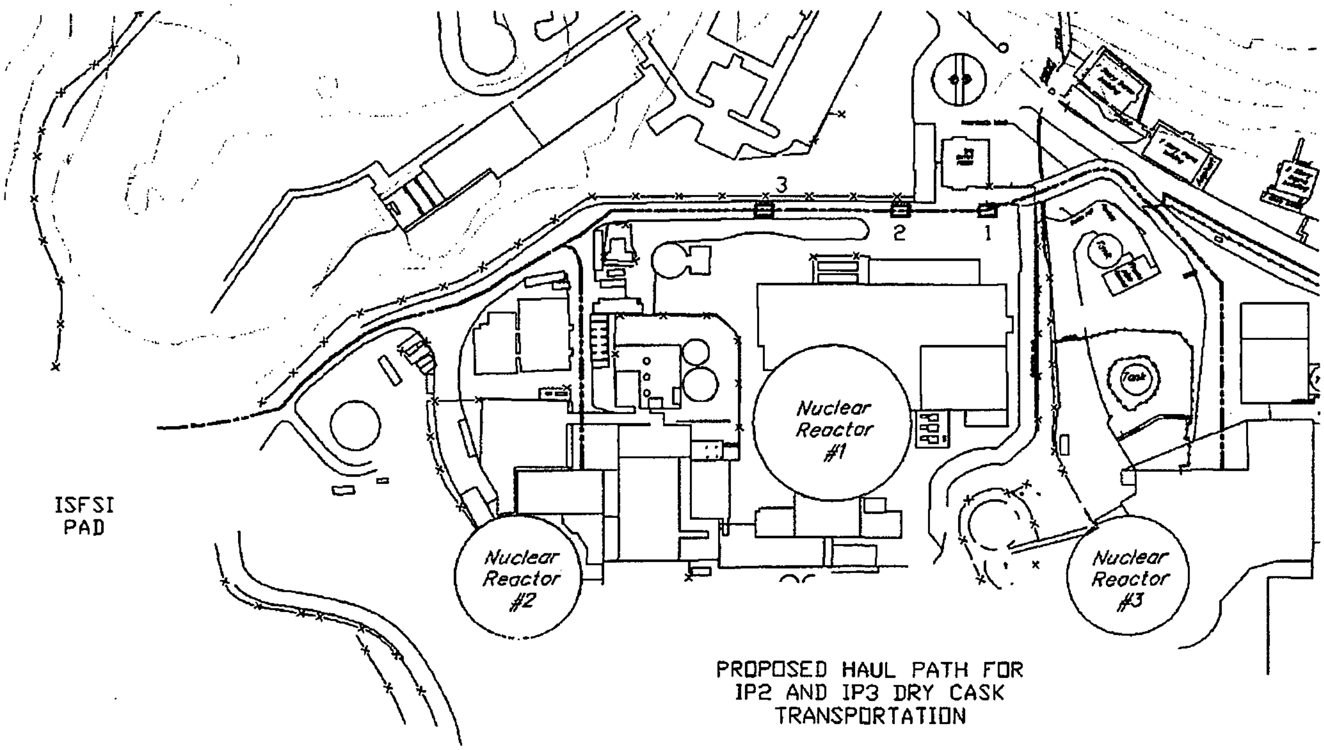


**Figure 15**
Proposed Haul Path for IP2 and IP3 Dry Cask Transportation

LEGEND of necessary modifications for the haul path depicted on **Figure 15**:

1 DI-7 Drain Inlet marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates. This is a potential modification, it depends on the roadway juncture between the existing and the proposed spur. Most likely it would not be needed.

2 DI-9 marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates.

3 DI-10 marked on Steam Generator Haul Route & Upgrades Drawing, No. WP-2090-001 (Ref. 6.2.10), covered with 2" plates. Grout road surface around the opening with high strength grout (Five Star Grout or similar).

ATTACHMENT F

to

ATTACHMENT 1

**Holtec Report HI-2032977, "Generic Crawler Specification"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT G

to

ATTACHMENT 1

**Drawing 6013 Rev 7, "Indian Point Unit 3 Shielded Transfer Canister Assembly; Licensing Drawing – General Arrangement"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT H

to

ATTACHMENT 1

**Drawing 6015 Rev 4, "Indian Point Unit 3 Shielded Transfer Canister Basket Assembly; Licensing Drawing – General Arrangement"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT I

to

ATTACHMENT 1

**Drawing 6571 Rev 2, "Indian Point Unit 3 HI-TRAC Transfer Cask Top Lid Assembly; Licensing Drawing – General Assembly"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT J

to

ATTACHMENT 1

**Holtec Report HI-2084176 Rev 2, "Criticality Safety Evaluation of the IP-3 Shielded Transfer Canister"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT K

to

ATTACHMENT 1

**Holtec Report HI-2084109 Rev 4, "Shielded Transfer Canister Shielding Report"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT L

to

ATTACHMENT 1

**Holtec Report HI-2094345, Rev 0, "Analysis of a Postulated HI-TRAC 100D Drop Accident During Spent Fuel Wet Transfer Operation"**

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT M

to

ATTACHMENT 1

# IPEC HI-STORM 100 Cask System 72.212 Evaluation Report

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

**ENTERGY NUCLEAR**

**10 CFR 72.212 Evaluation Report**

**for Independent Spent Fuel Storage Installations**

**Utilizing the Holtec International HI-STORM 100 Cask System**

**Revision 7**

**APPLICABLE SITES**

| | | | | | |
|---|---|---|---|---|---|
| ANO Unit 1: | X | GGNS: | X | W-3: | |
| ANO Unit 2: | X | RBS: | X | ECH: | |
| IPEC Unit 1: | X | JAF: | | PNPS: | |
| IPEC Unit 2: | X | VY: | X | WPO: | |
| IPEC Unit 3: | X | | | | |

Safety-Related: _____ Yes
X No

Prepared by: [signature] Responsible Engineer Date: 10/17/08

Reviewed by: [signature] Reviewer Date: 10/17/08

Approved by: M. Jerrell Campbell Dry Fuel Storage Project Manager Date: 10/20/08

## RECORD OF REVISION

| Revision # | Reason for Revision |
|---|---|
| 0 | Initial issuance of report. |
| 1 | Reflect changes to procedure numbers in Appendix A. |
| 2 | Reflect changes to LI-115 and update for changes to Appendix B (ANO) Revision 2. |
| 3 | Incorporate HI-STORM 100 CoC Revision 2 and various editorial changes. |
| 4. | Incorporate updated information. |
| 5. | Incorporate updated information and Appendix C for GGNS. |
| 6. | Incorporate updated information and reflect application to IPEC Unit 1, Units 2 and -3, and VY in accordance with NMM Procedure EN-LI-115. Establish Appendices E, F, and G for IPEC-1, IPEC-2 & 3, and VY, respectively. |
| 7. | Adopt HI-STORM CoC Amendment 5 and FSAR Revision 7 and make a small number of corrections to the regulatory requirements. |

**TABLE OF CONTENTS**

# I. INTRODUCTION

The HI-STORM 100 10 CFR 72.212 Evaluation Report (212 Report) documents reviews and approvals required by 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*, for conducting spent fuel storage activities at nuclear facilities operated by Entergy Operations, Inc. (Entergy) using the Holtec International HI-STORM 100 Cask System. The NRC issued Certificate of Compliance (CoC) Number 1014 (Docket Number 72-1014) to Holtec International for the HI-STORM 100 System in June 2000. The HI-STORM 100 212 Report is maintained in accordance with NMM Procedure EN-LI-115, *Independent Spent Fuel Storage Installation (ISFSI) Licensing Document Preparation and Control.*

Each facility has established or will establish Independent Spent Fuel Storage Installations (ISFSIs) for storing spent fuel at Arkansas Nuclear One - Units 1 and 2 (ANO), Grand Gulf Nuclear Station (GGNS), River Bend Station (RBS), Indian Point Energy Center (IPEC) -Units 1, 2, and 3, and Vermont Yankee (VY). (Although this document contains information relevant to IPEC and VY, it is not yet effective at these facilities.)

Entergy currently operates ISFSIs at ANO utilizing the Pacific Sierra Nuclear Associates' VSC-24 cask system and James A. FitzPatrick (JAF) utilizing the Holtec International HI-STORM 100 cask system. Rather than this 212 Report, the ANO VSC-24 ISFSI is controlled as a separate licensing basis document in accordance with NMM Procedure EN-LI-113, *Licensing Basis Document Change Process,* while the JAF HI-STORM 100 ISFSI is controlled in accordance with JAF Engineering Report RPT-SFS-04329, *Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation Report.*

Indian Point Unit 1 (IPEC-1) transferred all of their fuel from the spent fuel pool to the IPEC ISFSI in a single loading campaign. The IPEC-1 loading campaign was conducted in accordance with Amendment 4 to the HI-STORM CoC. The changes in Amendment 4 to the CoC apply exclusively to IPEC-1. Further, none of the changes in CoC Amendment 4 were maintained in CoC Amendment 5 (Ref: Email from T. Morin, Holtec, to J. Campbell, Entergy, dated August 5, 2008). Therefore, this "system-wide" 212 document was not revised to reflect use of CoC Amendment 4 at IPEC-1. All IPEC-1-unique requirements from CoC Amendment 4 are addressed in Appendix E to this report.

# II. ORGANIZATION

The 212 Report is organized to present information common to the facilities while also functioning as a central repository for site-specific information. Appendix A contains a list of corporate policies and procedures that support spent fuel storage activities common among the Entergy ISFSIs. Appendices B, C, D, E, F, and G contain site-specific information for the ISFSIs at ANO, GGNS, RBS, IPEC Unit 1 (IPEC-1), IPEC Units 2 and 3 (IPEC-2&3), and VY, respectively.

The site-specific appendices will be developed and controlled via separate supporting engineering controlling documents rather than via the 212 Report directly. These site-specific documents are:

- **ANO** – Engineering Report ER-ANO-2000-3333-000
- **GGNS** – Engineering Report ER-GGNS-2003-0018-000
- **RBS** – Engineering Report ER-RBS-2000-0001-000
- **IPEC-1** – N/A
- **IPEC-2&3** – N/A
- **VY** – Engineering Report ER-VY-08-00001

## III. BACKGROUND

In order to provide adequate spent fuel storage capacity for ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY, Entergy has or will construct an ISFSI at each site. Each ISFSI will be located within the site's protected area to consist of a single or multiple concrete pads as required.

Each ISFSI operates under the conditions of the general license in accordance with 10 CFR 72.210, *General License Issued*. The spent fuel storage cask designs approved for use under a general license are listed in 10 CFR 72.214. Entergy is using the Holtec International HI-STORM 100 Cask System, which is listed in 10 CFR 72.214 with certificate number 1014 and docket number 72-1014. The CoC was originally issued in 2000 and has been amended several times since then. The effective date of each amendment is listed in 10 CFR 72.214. Each plant decides individually which amendment they wish to use for loading casks based on their individual needs. The design basis for the HI-STORM 100 Cask System is documented in the Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System (CFSAR), as updated.

Revision 7 to this 212 Report documents the adoption of a later amendment of the HI-STORM CoC (Amendment 5). However, because HI-STORM casks at Entergy ISFSIs have been loaded under previous amendments to the CoC, information from those previous CoC amendments remains applicable for those casks. Thus, certain licensing basis information pertaining to casks loaded under previous amendments to the CoC is retained in this report. Requirements that have changed in the later amendments will be discussed in both forms to the extent that information meets the level of detail retained in this report. See the introduction to Section VI of this report for additional discussion.

It is incumbent upon the personnel responsible for dry storage operations at each site to track the applicable licensing basis for each loaded cask such that the appropriate CoC and FSAR requirements and acceptance criteria are applied cask-specifically. In some cases in this report, the language has been broadened to be less specific in order to minimize this double-tracking of similar licensing basis requirements where possible.

## IV. SYSTEM COMPONENTS

The basic HI-STORM 100 System consists of the following components:

- An interchangeable multi-purpose canister (MPC) providing a confinement boundary for BWR or PWR spent nuclear fuel.
- A storage overpack (HI-STORM) providing a structural and radiological boundary for long-term storage of the MPC placed inside it.
- A transfer cask (HI-TRAC) providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the HI-STORM storage overpack.

In most cases, MPCs have identical exterior dimensions, which render them interchangeable. A single HI-STORM overpack design is capable of storing each type of MPC. The exception involves the IPEC-1 MPCs; specifically, the five IPEC-1 MPCs are approximately 33 inches shorter than the standard Holtec design.

Necessary auxiliaries required to deploy the HI-STORM 100 System for storage are:

- A vacuum drying system or forced helium dehydrator (for loading high burn-up fuel, or high-heat load casks as specified in the CoC technical specifications)
- A helium (He) backfill system with leakage detector
- HI-TRAC annulus supplemental cooling system (for loading high burn-up fuel or high heat load casks as specified in the CoC technical specifications)
- Lifting and handling systems
- Welding equipment
- Transfer vehicles/trailer
- The ISFSI concrete storage pad on which the HI-STORM casks are placed

Additional information is documented in Section 1.2, *General Description of HI-STORM 100 System*, of the HI-STORM CFSAR.

## V. 10 CFR PART 72 REQUIREMENTS

The regulations in 10 CFR Part 72 establish requirements, procedures, and criteria for issuing licenses to receive, transfer, and possess power reactor spent fuel and other radioactive materials associated with spent fuel storage in an ISFSI. They also establish the terms and conditions under which the NRC will issue these licenses. These regulations also establish requirements, procedures, and criteria for issuing Certificates of Compliance approving spent fuel storage cask designs. 10 CFR 72.13(c) identifies those regulations that are specifically applicable to a general licensee.

The following table identifies the requirements of 10 CFR Part 72 and documents how Entergy complies with these requirements, as applicable to a general licensee.

| 10 CFR PART 72 REQUIREMENTS | | ASSESSMENT OF COMPLIANCE |
|---|---|---|
| **Subpart A<br>General Provisions** | | |
| §72.1 | Purpose. | None required. |
| §72.2 | Scope. | |
| | (a)(1) | None required. |
| | (a)(2) | Not required for a general license per §72.13(c). |
| | (b)<br>(c) | None required. |
| | (d) | Not required for a general license per §72.13(c). |
| | (e) | None required. |
| | (f) | Not required for a general license per §72.13(c). |
| §72.3 | Definitions. | None required. |
| §72.4 | Communications. | All communications and reports are sent ATTN: Document Control Desk, Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and as specified by 10 CFR 72 requirements.<br>Entergy nuclear facilities have been assigned the following docket numbers:<br>Arkansas Nuclear One (ANO) – 72-13<br>Grand Gulf Nuclear Station (GGNS) – 72-50<br>River Bend Station (RBS) – 72-49<br>Indian Point Energy Center (IPEC) – 72-51<br>Vermont Yankee (VY) – 72-59 |

| 10 CFR PART 72 REQUIREMENTS | | ASSESSMENT OF COMPLIANCE |
|---|---|---|
| §72.5 | Interpretations. | None required. |
| §72.6 | License required; types of licenses. | |
| | (a)<br>(b)<br>(c)(1) | None required. In §72.6(b), "specific license" refers to the 10 CFR 50 license held by the facility. |
| | (c)(2) | Not required for a general license per §72.13(c). |
| §72.7 | Specific exemptions. | None required. |
| §72.8 | Denial of licensing by Agreement States. | None required. |
| §72.9 | Information collection requirements: OMB approval. | None required. |
| §72.10 | Employee protection. | Entergy Nuclear Management Manual (NMM) Procedure EN-EC-100, *Guidelines for Implementation of the Employee Concerns Program*, encompasses the requirements of this article. No further action is required. |
| §72.11 | Completeness and accuracy of information. | NMM Procedure EN-LI-106, *NRC Correspondence*, specifies the requirements of this article. Also, NRC Form 3 Notice To Employees contains relevant information. No further action is required. |
| §72.12 | Deliberate misconduct. | NMM Procedure ENS-HR-135, *Disciplinary Action*, encompasses the requirements of this article. (A fleet procedure pertaining to this issue will be developed in the future.) No further action is required. |
| §72.13 | Applicability | §72.13 identifies those sections under Part 72 that apply to the activities associated with a specific license [§72.13(b)], a general license [§72.13(c)], or a CoC [§72.13(d)]. Sections not applicable to a general license per §72.13(c) are so noted in this matrix. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| **Subpart B**<br>**License Application, Form, and Contents** | |
| §72.16 Filing of application for specific license. | Not required for a general license per §72.13(c). |
| §72.18 Elimination of repetition. | Not required for a general license per §72.13(c). |
| §72.20 Public inspection of application. | Not required for a general license per §72.13(c). |
| §72.22 Contents of application: General and financial information. | Not required for a general license per §72.13(c). |
| §72.24 Contents of application: Technical information. | Not required for a general license per §72.13(c). |
| §72.26 Contents of application: Technical specifications. | Not required for a general license per §72.13(c). |
| §72.28 Contents of application: Applicant's technical qualifications. | Not required for a general license per §72.13(c). |
| §72.30 Financial assurance and recordkeeping for decommissioning. | |
| (a)<br>(b) | Not required for a general license per §72.13(c). |
| (c)<br>(d) | To be submitted with each site's decommissioning plan five or more years prior to decommissioning in accordance with §§50.54(bb), 50.82, and 72.218. |
| §72.32 Emergency plan. | |
| (a)<br>(b) | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (c)<br>(d) | Each facility's emergency plan meets the requirements of §50.47. Entergy has evaluated or will evaluate the impact of dry fuel storage activities on each facility's emergency plan. The results of these reviews have been or will be documented in the site-specific appendices (applicable Section X.3.13) of this Report as completed.<br>[§72.32(c) only, also see discussion under §72.212(b)(6).] |
| §72.34 Environmental report. | Not required for a general license per §72.13(c). |
| **Subpart C**<br>**Issuance and Conditions of License** | |
| §72.40 Issuance of license. | Not required for a general license per §72.13(c). |
| §72.42 Duration of license; renewal. | Not required for a general license per §72.13(c). |
| §72.44 License conditions. | |
| (a) | Not required for a general license per §72.13(c). |
| (b) | None required; covered in Subpart K; specifically, §72.212(b)(2)(i)(A) and conformity with the CoC. See additional discussion on training [in accordance with §72.44(b)(4)] in §72.190. Bankruptcy reporting requirements are covered by §50.54(cc). |
| (c)<br>(d)<br>(e) | Not required for a general license per §72.13(c). |
| (f) | None required; covered in Subpart K; specifically, §72.212(b)(2)(i)(A) and conformity with the CoC. See additional discussion on training [in accordance with §72.44(b)(4)] in §72.190. Bankruptcy reporting requirements are covered by §50.54(cc). |
| (g) | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| §72.46 Public hearings. | Not required for a general license per §72.13(c). |
| §72.48 Changes, tests, and experiments. | Entergy will evaluate any proposed cask design changes and ensure the HI-STORM CoC envelopes them. Entergy will report any changes evaluated under this section in periodic submittals in compliance with NMM Procedure EN-LI-112, *10 CFR 72.48 Review Program*, and NMM Procedure EN-EV-115, *Environmental Reviews and Evaluations.* |
| §72.50 Transfer of license.<br>(a) | None required at this time. |
| (b)<br>(c) | Not required for a general license per §72.13(c). |
| §72.52 Creditor regulations. | |
| (a)<br>(b) | Compliance through 10 CFR 50.81. |
| (c) | Not required for a general license per §72.13(c). |
| (d)<br>(e) | Compliance through 10 CFR 50.81. |
| §72.54 Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas. | Not required for a general license per §72.13(c). |
| §72.56 Application for amendment of license. | Not required for a general license per §72.13(c). |
| §72.58 Issuance of amendment. | Not required for a general license per §72.13(c). |
| §72.60 Modification, revocation, and suspension of license. | None required. |
| §72.62 Backfitting. | NRC process, no action required by Entergy at this time. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| **Subpart D<br>Records, Reports, Inspections, and Enforcement** | |
| §72.70 Safety analysis report updating. | Not required for a general license per §72.13(c). |
| §72.72 Material balance, inventory, and records requirements for stored material. | NMM Procedure EN-NF-104, *Special Nuclear Materials Program*, contains inventory, transfer, and storage requirements for dry fuel storage. A documentation review of fuel in dry storage is conducted yearly in accordance with this procedure. |
| §72.74 Reports of accidental critically or loss of special nuclear material. | NMM Procedure EN-LI-108, *Event Notification and Reporting*, requires compliance with 10 CFR 72.74. |
| §72.75 Reporting requirements for specific events and conditions. | NMM Procedure EN-LI-108, *Event Notification and Reporting*, includes the reports required by §72.75. |
| §72.76 Material status reports. | NMM procedures EN-NF-104, *Special Nuclear Materials Program*, EN-NF-200, *Special Nuclear Material Control,* and EN-NF-201, *Special Nuclear Materials Reporting*, govern material status reporting requirements as required by §74.13. |
| §72.78 Nuclear material transfer reports. | NMM procedures EN-NF-104, *Special Nuclear Materials Program*, EN-NF-200, *Special Nuclear Material Control, and* EN-NF-201, *Special Nuclear Materials Reporting*, govern nuclear material transfer reporting requirements as required by §74.15.<br><br>Each procedure requires material transfer reports to be submitted in accordance with DOE/NRC 741 (new number for the report is NUREG/BR-0006 and NMMSS *Personal Computer Data Input for NRC Licensees*).<br><br>However, movement of fuel from the spent fuel pool to dry storage within the protected area is not considered "transfers" under §74.15 requirements. |
| §72.80 Other records and reports. | |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (a)<br>(b)<br>(c)<br>(d)<br>(e)<br>(f) | Records will be maintained in accordance with site procedures and NMM procedures that meet the Entergy Quality Assurance Program Manual (QAPM) document control requirements for Engineering Reports (ERs), department procedures, and licensing documents. Financial report requirements are met by compliance with §50.71(b). |
| (g) | Not required for a general license per §72.13(c). |
| §72.82 Inspections and tests. | The requirements of §72.82 are covered by §50.70. |
| §72.84 Violations. | None required. |
| §72.86 Criminal penalties. | None required. |
| **Subpart E**<br>**Siting Evaluation Factors** | |
| §72.90 General considerations. | Not required for a general license per §72.13(c). |
| §72.92 Design basis external natural events. | Not required for a general license per §72.13(c). |
| §72.94 Design basis external man-induced events. | Not required for a general license per §72.13(c). |
| §72.96 Siting limitations. | Not required for a general license per §72.13(c). |
| §72.98 Identifying regions around an ISFSI or MRS site. | Not required for a general license per §72.13(c). |
| §72.100 Defining potential effects of the ISFSI or MRS on the region. | Not required for a general license per §72.13(c). |
| §72.102 Geological and seismological characteristics. | Not required for a general license per §72.13(c). |
| §72.103 Geological and seismological characteristics for applications for dry cask modes of storage on or after December 16, 2003. | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| §72.104 Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS. | Addressed by §72.212(b)(2)(i)(C) with required evaluation described in the site-specific appendices (applicable Section X.3.9) of this report as completed and site compliance with 10 CFR 50 requirements for control of direct radiation. In addition, see the assessment of compliance discussion for §72.106, below. |
| §72.106 Controlled area of an ISFSI or MRS. | Site-specific drawings show or will show the distances from the ISFSI to the site boundary. Detailed descriptions will be provided in the site-specific appendices (applicable Section X.3.1) of this report, as completed. Dose calculations will also be described in the site-specific appendices (applicable Section X.3.9) of this report, as completed. |
| §72.108 Spent fuel for high-level radioactive waste transportation. | Not required for a general license per §72.13(c). |
| **Subpart F<br>General Design Criteria** | |
| §72.120 General considerations. | Not required for a general license per §72.13(c). |
| §72.122 Overall requirements. | See detailed information in site-specific appendices (applicable Section X.3.2) of this report, as completed. |
| §72.124 Criteria for nuclear criticality safety. | See detailed information in site-specific appendices (applicable Section X.3.3) of this report, as completed. |
| §72.126 Criteria for radiological protection. | See detailed information in site-specific appendices (applicable Section X.3.4). |
| §72.128 Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling. | Not required for a general license per §72.13(c). |
| §72.130 Criteria for decommissioning. | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| **Subpart G<br>Quality Assurance** | |
| §72.140 Quality assurance requirements. | The Entergy Quality Assurance (QA) Program, which satisfies applicable criteria of Appendix B to 10 CFR 50, has been approved by the NRC (reference CNRI 2001-00007, dated 12/5/01). Entergy notified NRC of its intent to use the Entergy program for fuel loading activities at its ISFSIs as required in section 72.140(d) via the following letters:<br>**ANO** – Letter 0CAN109511 dated 10/24/95<br>**GGNS** – Letter CNRO-2003-00026 dated 6/19/03.<br>**RBS** – Letter CNRO-2003-00026 dated 6/19/03.<br>**IPEC** – Letter ENOC-07-00014 dated May 2, 2007<br>**VY** – Letter BVY-07-031 dated 4/18/07<br>The Holtec Quality Assurance Program is described in Section 13.3 of the HI-STORM 100 CFSAR, which has been approved by the NRC. |
| §72.142 Quality assurance organization. | Covered in the Entergy QA Program |
| §72.144 Quality assurance program. | The Entergy QA Program, which satisfies applicable criteria of Appendix B to 10 CFR 50, was approved by the NRC (reference letter CNRI-2001-00007 dated 12/5/01). IPEC Units 1, 2, and 3, and VY adopted the Entergy QA Program with QAPM Revisions 7 and 8, respectively.<br>Indoctrination and training of individuals involved in fuel loading activities, as required in section 72.144(d), will be accomplished prior to first cask use at each Entergy facility. |
| §72.146 Design control. | Covered in the Entergy QA Program |
| §72.148 Procurement document control. | Covered in the Entergy QA Program |
| §72.150 Instructions, procedures, and drawings. | Covered in the Entergy QA Program |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
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| §72.152 Document control. | Covered in the Entergy QA Program |
| §72.154 Control of purchased material, equipment, and services. | Covered in the Entergy QA Program |
| §72.156 Identification and control of materials, parts, and components. | Covered in the Entergy QA Program |
| §72.158 Control of special processes. | Covered in the Entergy QA Program |
| §72.160 Licensee and certificate holder inspection. | Covered in the Entergy QA Program |
| §72.162 Test control. | Covered in the Entergy QA Program |
| §72.164 Control of measuring and test equipment. | Covered in the Entergy QA Program |
| §72.166 Handling, storage, and shipping control. | Covered in the Entergy QA Program |
| §72.168 Inspection, test, and operating status. | Covered in the Entergy QA Program |
| §72.170 Nonconforming materials, parts, or components. | Covered in the Entergy QA Program |
| §72.172 Corrective action. | Covered in the Entergy QA Program<br>The Entergy Corrective Action Program is controlled in accordance with NMM Procedure EN-LI-102, *Corrective Action Process.* |
| §72.174 Quality assurance records. | Covered in the Entergy QA Program |
| §72.176 Audits. | Covered in the Entergy QA Program |
| **Subpart H**<br>**Physical Protection** | Compliance with this Subpart is covered by §72.212(b)(5) and is accomplished via each facility's security plan. |
| §72.180 Physical security plan. | Not required for a general license per §72.13(c). |
| §72.182 Design for physical protection. | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| §72.184 Safeguards contingency plan. | Not required for a general license per §72.13(c). |
| §72.186 Changes to physical security and safeguards contingency plans. | Not required for a general license per §72.13(c). |
| **Subpart I**<br>**Training and Certification of Personnel** | |
| §72.190 Operator requirements. | Training is required in accordance with the HI-STORM 100 CoC. Personnel involved in storage activities are certified by a combination of classroom training, and experience and qualification cards. See detailed information in site-specific appendices (applicable Section X.3.5) of this report, as completed |
| §72.192 Operator training and certification program. | Not required for a general license per §72.13(c). |
| §72.194 Physical requirements. | Physical condition and health of personnel involved with operation of equipment under this section are subject to requirements of:<br>• NMM Procedure EN-NS-102, *Fitness For Duty Program*<br>• NMM Procedure EN-NS-112, *Medical Program*<br>• NMM Procedure EN-MA-119, *Material Handling Program* |
| **Subpart J**<br>**Provision of MRS Information to State Governments And Indian Tribes** | |
| §72.200 Provision of MRS information. | Not required for a general license per §72.13(c). |
| §72.202 Participation in license reviews. | Not required for a general license per §72.13(c). |
| §72.204 Notice to States. | Not required for a general license per §72.13(c). |
| §72.206 Representation. | Not required for a general license per §72.13(c). |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
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| **Subpart K**<br>**General License for Storage of Spent Fuel at Power Reactor Sites** | |
| §72.210 General license issued.<br><br>A general license is hereby issued for the storage of spent fuel in an independent spent fuel storage installation at power reactor sites to persons authorized to possess or operate nuclear power reactors under part 50 of this chapter. | The NRC issued General License CoC 72-1014 for the HI-STORM 100 Cask System effective June 1, 2000.<br><br>Entergy is authorized to possess nuclear fuel in accordance with the Part 50 licenses issued by NRC. The Part 50 licenses are:<br><br>**ANO-1** – DPR-51<br><br>**ANO-2** – NPF-6<br><br>**GGNS** – NPF-29<br><br>**RBS** – NPF-47<br><br>**IPEC- 1** – DPR-5<br><br>**IPEC-2** – DPR-26<br><br>**IPEC-3** – DPR-64<br><br>**VY** – DPR-28<br><br>The ISFSI for each site is or will be located within the protected area for each site. Therefore, Entergy is licensed to store spent fuel in accordance with a general license authorized via §72.210. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| §72.212 Conditions of general license issued under §72.210. | |
| (a)(1) The general license is limited to that spent fuel which the general licensee is authorized to possess at the site under the specific license for the site. | Entergy is authorized to possess nuclear fuel in accordance with the Part 50 licenses issued by NRC. The Part 50 licenses are:<br>**ANO-1** – DPR-51<br>**ANO-2** – NPF-6<br>**GGNS** – NPF-29<br>**RBS** – NPF-47<br>**IPEC- 1** – DPR-5<br>**IPEC-2** – DPR-26<br>**IPEC-3** – DPR-64<br>**VY** – DPR-28 |
| (a)(2) This general license is limited to storage of spent fuel in casks approved under the provisions of this part. | The HI-STORM 100 System is an approved spent fuel storage cask and is listed in 10 CFR 72.214. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (a)(3) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance terminates 20 years after the date that the particular cask is first used by the general licensee to store spent fuel, unless the cask's Certificate of Compliance is renewed, in which case the general license terminates 20 years after the cask's Certificate of Compliance renewal date. In the event that a cask vendor does not apply for a cask model re-approval under §72.240, any cask user or user's representative may apply for a cask design re-approval. If a Certificate of Compliance expires, casks of that design must be removed from service after a storage period not to exceed 20 years. | Each site will begin cask license review prior to license expiration as identified below:<br><br>**ANO** – ANO procedure OP-1022.012 , *Storage, Control, & Accountability of Nuclear Fuel*, requires annual review of cask license status and cask unloading if the license not renewed.<br><br>**GGNS** –<br><br>**RBS** - RBS procedure REP-0029, *Fuel Movement*, requires a review 15 years from the first cask loading to determine the need for license renewal.<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2&3** - To Be Determined<br><br>**VY** – To Be Determined |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
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| (b) The general licensee shall: | |
| (b)(1)(i) Notify the Nuclear Regulatory Commission using instructions in §72.4 at least 90 days prior to first storage of spent fuel under this general license. The notice may be in the form of a letter, but must contain the licensee's name, address, reactor license and docket numbers, and the name and means of contacting a person responsible for providing additional information concerning spent fuel under this general license. A copy of the submittal must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office listed in appendix D to part 20 of this chapter. | Entergy has notified the NRC of its intent to store spent fuel using the HI-STORM 100 storage cask at ANO and RBS as documented in the following letters:<br>**ANO** – Letter 0CAN010303 dated January 31, 2003<br>**GGNS** – Letter GNRO-2006/00033 dated May 16, 2006<br>**RBS** – Letter RBG-46429 dated April 26, 2005<br>**IPEC**– Letter NL-03-190 dated December 29, 2003<br>**VY** – Letter BVY 07-013 dated March 13, 2007 |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(1)(ii) Register use of each cask with the Nuclear Regulatory Commission no later than 30 days after using that cask to store spent fuel. This registration may be accomplished by submitting a letter using instructions in §72.4 containing the following information: the licensee's name and address, the licensee's reactor license and docket numbers, the name and title of a person responsible for providing additional information concerning spent fuel storage under this general license, the cask certificate and model numbers, and the cask identification number. A copy of each submittal must be sent to the administrator of the appropriate Nuclear Regulatory Commission regional office listed in appendix D to part 20 of this chapter. | Entergy will register each cask within 30 days following loading in accordance with its site procedure pertaining to spent fuel removal and dry storage operations. These procedures are:<br>**ANO** – OP-3403.005, *HI-STORM 100 System Loading Operations*<br>**GGNS** – 20-S-01-003<br>**RBS** – DFS-0003, *MPC Transfer Operations and HI-STORM Transport*<br>**IPEC-1** – To be Determined<br>**IPEC-2&3** - To be Determined<br>**VY** – To be Determined |
| (b)(1)(iii) Fee. Fees for inspections related to spent fuel storage under this general license are those shown in §170.31 of this chapter. | Fees associated with inspections are incurred on a case-by-case basis. |
| (b)(2)(i) Perform written evaluations, prior to use, that establish that: | |
| (b)(2)(i)(A) conditions set forth in the Certificate of Compliance have been met; | See Section VI; also see site-specific appendices (applicable Section X.3.7) of this report, as completed. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(2)(i)(B) cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks considering potential amplification of earthquake through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion; and | Each facility's ISFSI cask storage pad has been or will be designed to adequately support the static and dynamic loads of the storage casks in accordance with CFSAR requirements, including the effects of soil-structure interaction and potential soil liquefaction. The storage pads were/will be constructed to the facility's construction specifications. Site-specific appendices (applicable Section X.3.8) of this report provide/will provide more information pertaining to the ISFSI pads. Engineering evaluations are/will be summarized in each appendix to document the structural adequacy of each facility's loading area and roadway to support the static and dynamic loads of the casks, including soil-structure interaction and soil liquefaction. |
| (b)(2)(i)(C) The requirements of §72.104 have been met. A copy of this record must be retained until spent fuel is no longer stored under the general license issued under §72.210. | Dose rates calculations for each facility's ISFSI demonstrate or will demonstrate that the combined plant and ISFSI dose rates satisfy the criteria. The calculations will be documented in site-specific appendices (applicable Section X.3.9) of this report, as completed. |
| (b)(2)(ii) The licensee shall evaluate any changes to the written evaluations required by this paragraph using the requirements of §72.48(c). A copy of this record shall be retained until spent fuel is no longer stored under the general license issued under §72.210. | Entergy evaluates any proposed cask design changes and ensure by evaluation that all changes are enveloped by the HI-STORM CoC. Entergy reports any changes evaluated under this section in periodic submittals in compliance with NMM Procedure EN-LI-112, *10 CFR 72.48 Review Program.* |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(3) Review the Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(2) of this section. | Entergy has reviewed the appropriate revisions of the HI-STORM CFSAR and the NRC Safety Evaluation Reports (SERs) to ensure that site parameters are enveloped. Each site will document the results of these reviews in site-specific appendices (applicable Section X.3.10) of this report, as completed. |
| (b)(4) Prior to use of this general license, determine whether activities related to storage of spent fuel under this general license involve a change in the facility technical specifications or require a license amendment for the facility pursuant to §50.59(c)(2) of this chapter. Results of this determination must be documented in the evaluation made in paragraph (b)(2) of this section. | Entergy has reviewed or will review each facility's TS to determine if changes to TS is required in accordance with §50.59 and NMM Procedures EN-LI-100, *Process Applicability Determination,* and EN-LI-101, *10 CFR 50.59 Review Program.* The results of these reviews are documented in site-specific appendices (applicable Section X.3.11) of this report, as completed. |
| (b)(5) Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to §73.55 of this chapter with the following additional conditions and exceptions. | The results of these reviews are documented in site-specific appendices (applicable Section X.3.12) of this report, as completed. |

| 10 CFR PART 72 REQUIREMENTS | | ASSESSMENT OF COMPLIANCE |
|---|---|---|
| (b)(5)(i) | The physical security organization and program for the facility must be modified as necessary to assure that activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with §73.55 of this chapter. | Each ISFSI is or will be located inside each facility's protected area. Spent fuel movement activities are performed inside the protected area except for HI-STORM overpack transport at RBS. Due to the plant protected area configuration and the location of the ISFSI, the loaded HI-STORM will be moved outside the protected area on a section of the path between the fuel building and the storage pad.<br><br>Each facility's security plan has been or will be updated as needed to include the ISFSI in accordance requirements of §50.54. Any future changes to the plan will continue to be made in accordance with §50.54. Reviews of each facility's security plan will be documented in site-specific appendices (applicable Section X.3.12) of this report, as completed. |
| (b)(5)(ii) | Storage of spent fuel must be within a protected area, in accordance with §73.55(c) of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with this general license. | Each ISFSI is or will be located within the protected area of the facility. IPEC-1 and IPEC-2&3 share a common ISFSI within the protected area of the site. |
| (b)(5)(iii) | For purposes of this general license, searches required by §73.55(d)(1) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosives detection equipment. | Each ISFSI is or will be located within the protected area of the facility. Each facility's security plan governs searches prior to entry into the protected area. Reviews of each facility's security plan are or will be documented in site-specific appendices (applicable Section X.3.12) of this report, as completed. |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(5)(iv) The observational capability required by §73.55(h) (6) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of closed circuit television. | Each ISFSI is or will be located within the protected area of the facility. Each facility's security plan specifies observation requirements in the protected area. |
| (b)(5)(v) For the purpose of this general license, the licensee is exempt from §§73.55(h)(4)(iii)(A) and 73.55(h)(5) of this chapter. | No action required. |
| (b)(6) Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals. | Entergy has evaluated or will evaluate the impact of dry fuel storage activities on each facility's emergency plan, quality assurance program, training program, and radiation protection program. Entergy has determined there is no adverse impact on the effectiveness of these programs at ANO and RBS. The results of these reviews are or will be documented in site-specific appendices (applicable Section X.3.13) of this report, as completed. |
| (b)(7) Maintain a copy of the Certificate of Compliance and documents referenced in the certificate for each cask model used for storage of spent fuel, until use of the cask model is discontinued. The licensee shall comply with the terms and conditions of the certificate. | A controlled copy of the HI-STORM 100 CoC and related documents are maintained in accordance with the Entergy QA Program, NMM Procedures EN-AD-103, *Document Control and Records Management Activities*, and EN-LI-113, *License Basis Document Change Process*. |
| (b)(8)(i) Accurately maintain the record provided by the cask supplier for each cask that shows, in addition to the information provided by the cask vendor, the following: | |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(8)(i)(A) The name and address of the cask vendor or lessor; | Entergy uses the Holtec International HI-STORM 100 Cask System at ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY. The vendor address is:<br><br>Holtec International<br>Holtec Center<br>555 Lincoln Drive West<br>Marlton, NJ 08053 |
| (b)(8)(i)(B) The listing of spent fuel stored in the cask; and | Entergy will validate and document spent fuel storage in each cask in accordance with corporate and site procedures. These procedures are:<br><br>**Corporate** – EN-NF-200, *Special Nuclear Material Control*<br><br>**ANO** – OP-1022.012, *Storage, Control, and Accountability of Special Nuclear Material*<br><br>**GGNS** – 17-S-02-300<br><br>**RBS** - REP-0029, *Fuel Movement*<br><br>**IPEC-1** – 0-NF-203, *Internal Transfer of Fuel Assemblies and Inserts*<br><br>**IPEC-2&3** - 0-NF-203, *Internal Transfer of Fuel Assemblies and Inserts*<br><br>**VY** – To be Determined |

| 10 CFR PART 72 REQUIREMENTS | ASSESSMENT OF COMPLIANCE |
|---|---|
| (b)(8)(i)(C) Any maintenance performed on the cask. | Any work performed on a cask will be controlled in accordance with corporate and facility maintenance program procedures. These procedures are:<br><br>**Corporate** – NMM Procedure EN-MA-101, *Conduct of Maintenance*<br><br>**ANO** – OP-1000.024, *Control of Maintenance*<br><br>**GGNS** – 07-S-01-205, *Conduct of Maintenance Activities*<br><br>**RBS** – NMM Procedure EN-MA-101, *Conduct of Maintenance*<br><br>**IPEC-1** – To be Determined<br><br>**IPEC-2&3** - To be Determined<br><br>**VY** – To be Determined |
| (b)(8)(ii) This record must include sufficient information to furnish documentary evidence that any testing and maintenance of the cask has been conducted under an NRC-approved quality assurance program. | Plant/System Engineering at each facility has detailed cask fabrication information.<br><br>Each facility's component/equipment database provides or will provide the safety-related component classification for input into any maintenance required and controlled under facility control of maintenance programs. |
| (b)(8)(iii) In the event that a cask is sold, leased, loaned, or otherwise transferred to another registered user, this record must also be transferred to and must be accurately maintained by the new registered user. This record must be maintained by the current cask user during the period that the cask is used for storage of spent fuel and retained by the last user until decommissioning of the cask is complete. | None required or anticipated until transfer of the fuel to the Department of Energy. |

| 10 CFR PART 72 REQUIREMENTS | | ASSESSMENT OF COMPLIANCE |
|---|---|---|
| | (b)(9) Conduct activities related to storage of spent fuel under this general license only in accordance with written procedures. | Safety-related procedures controlled under the Entergy QAPM have been or will be developed for cask preparation, cask loading, unloading, and storage activities. These procedures are/will be identified in site-specific appendices (applicable Section X.2) of this report, as completed. |
| | (b)(10) Make records and casks available to the Commission for inspection. | None required at this time. The records and casks will be available to the NRC for inspection. |
| §72.214 | (Selected) List of approved spent fuel storage casks.<br><br>The following casks are approved for storage of spent fuel under the conditions specified in their Certificates of Compliance.<br><br>Certificate Number: 1014<br><br>SAR Submitted by: Holtec International<br><br>SAR Title: Final Safety Analysis Report for the HI-STORM 100 Cask System<br><br>Docket Number: 72-1014<br><br>Certificate Expiration Date: June 1, 2020<br><br>Model Number: HI-STORM 100 | The Holtec HI-STORM 100 System is approved for storage of spent fuel under the conditions specified in the HI-STORM 100 System CoC. The HI-STORM 100 System is being utilized at ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY as described in Appendices B, C, D, E, F, and G. |
| §72.216 | Reserved | |
| §72.218 | Termination of licenses. | No action required at this time. |
| §72.220 | Violations.<br><br>This general license is subject to the provisions of §72.84 for violation of the regulations under this part. | No action required. |

| **Subpart L<br>Approval of Spent Fuel Storage Casks** | |
|---|---|
| §72.230 Procedures for spent fuel storage cask submittals. | Not required for a general license per §72.13(c). |
| §72.232 Inspection and tests. | Not required for a general license per §72.13(c). |
| §72.234 Conditions of approval. | Not required for a general license per §72.13(c). |
| §72.236 Specific requirements for spent fuel storage cask approval. | Not required for a general license per §72.13(c). |
| §72.238 Issuance of an NRC Certificate of Compliance. | Not required for a general license per §72.13(c). |
| §72.240(a) The certificate holder, a licensee using a spent fuel storage casks, or the representative of a licensee using a spent fuel storage cask shall apply for re-approval of the design of spent fuel storage cask. | No action required at this time. Entergy will discuss CoC re-approval with Holtec when the CoC expiration date is approaching in sufficient time to decide to request re-approval of the CoC directly if Holtec does not. |
| §72.240(b) and (c) Conditions of spent fuel cask reapproval. | Not required for a general license per §72.13(c). |
| §72.242 Recordkeeping and reports. | Not required for a general license per §72.13(c). |
| §72.244 Application for amendment of a certificate of compliance. | Not required for a general license per §72.13(c). |
| §72.246 Issuance of amendment to a certificate of compliance. | Not required for a general license per §72.13(c). |
| §72.248 Safety analysis report updating. | Not required for a general license per §72.13(c). |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

10 CFR 72.212(b)(2)(i)(A) requires general licensees to perform written evaluations, prior to use, that establish that conditions set forth in the Certificate of Compliance (CoC) have been met.

The HI-STORM 100 CoC is comprised of three sections:

1. CoC Conditions

2. Appendix A – Technical Specifications (TS)

3. Appendix B – Approved Contents and Design Features

The following table identifies the requirements specified in each section of the HI-STORM 100 CoC and documents how Entergy complies with these requirements. Entergy, with the exception of IPEC-1, has loaded spent fuel casks at ISFSIs under Amendments 1 and 2 to the CoC as reflected in Revisions 0 through 6 of this document. IPEC-1 loaded to CoC Amendment 4 (issued specifically for IPEC-1) and is addressed uniquely in Appendix E to this report.

At ANO, casks loaded under CoC Amendment 1 are now operated under CoC Amendment 2, as reflected in Revisions 3-6 of this report. However, due to on-going rulemaking concerning proposed NRC Regulatory Issue Summary 2007-26, *Implementation of Certificate of Compliance Amendments to Previously Loaded Spent Fuel Storage Casks,* issued by the NRC on January 14, 2008, Amendment 5 will not be implemented for casks loaded under previous CoC amendments. The applicable CoC information from Amendments 2 is being retained unless the requirements became obsolete after the casks were loaded (e.g., loading operations requirements). This is determined on a CoC requirement-specific basis in the following tables.

The convention used in this report is as follows: Requirements from CoC Amendment 2 that have been revised in Amendment 5 and became obsolete after the casks were loaded have been replaced with the Amendment 5 CoC text. Requirements that remain in effect from CoC Amendment 2 for previously loaded casks and have been modified in CoC Amendment 5 are retained, and the revised text added. In both scenarios, the CoC requirements that are identical between the respective revisions are shown in regular text. Changes from Amendment 2 are shown in bold text to distinguish them.

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

| CoC CONDITIONS (Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
|---|---|
| **1. CASK** | No action required. |
| a. Model No.: HI-STORM 100 Cask System<br><br>The HI-STORM 100 Cask System (the cask) consists of the following components:<br><br>(1) interchangeable multi-purpose canisters (MPCs), which contain the fuel;<br><br>(2) a storage overpack (HI-STORM), which contains the MPC during storage; and<br><br>(3) a transfer cask (HI-TRAC), which contains the MPC during loading, unloading, and transfer operations.<br><br>The cask stores up to 32 pressurized water reactor (PWR) fuel assemblies or 68 boiling water reactor (BWR) fuel assemblies. | |
| b. Description<br><br>The HI-STORM 100 Cask System is certified as described in the Final Safety Analysis Report (FSAR) and in the U. S. Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance. The cask comprises three discrete components: the MPCs, the HI-TRAC transfer cask, and the HI-STORM overpack.<br><br>The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. **All MPC components that may come into contact with spent fuel pool water or the ambient environment are** made entirely of stainless steel except for the neutron absorbers**, aluminum seals on vent and drain port caps,** and aluminum heat conduction elements (AHCEs), which are installed in some early-vintage MPCs. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. **All confinement boundary components are made entirely of stainless steel.** The honeycombed basket, which is equipped with neutron absorbers, provides | |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

| CoC CONDITIONS (Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
|---|---|
| criticality control.<br><br>There are eight types of MPCs: the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, and MPC-68FF. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. All eight MPCs have the same external diameter.<br><br>The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a **neutron shield** jacket attached to the exterior. Two sizes of HI-TRAC transfer casks are available: the 125 ton HI-TRAC and the 100 ton HI-TRAC. The weight designation is the maximum weight of a loaded transfer cask during any loading, unloading, or transfer operation. Both transfer cask sizes have identical cavity diameters. The 125 ton HI-TRAC transfer cask has thicker shielding and larger outer dimensions than the 100 ton HI-TRAC transfer cask.<br><br>The HI-STORM 100 or 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S is a variation of the HI-STORM 100 overpack design that includes a modified lid which incorporates the air outlet ducts into the lid, allowing the overpack body to be shortened. The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its sidewall consists of plain (un-reinforced) concrete that is enclosed between inner and outer carbon steel shells. The overpack has four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has **supports** attached to its interior surface to guide the MPC during insertion and removal, provide a medium to absorb impact loads, and allow cooling air to circulate through the overpack. A loaded MPC is stored within the HI-STORM 100 or 100S storage overpack in a vertical orientation.The HI-STORM 100A **and 100SA are** variants of the HI-STORM family outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the concrete pad in high seismic applications | |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

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| **2. OPERATING PROCEDURES** | |
| Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the FSAR. | Operating procedures governing cask handling, loading, movement, surveillance, and maintenance have been or will be developed in accordance with Chapter 8 of the CFSAR. These procedures are/will be listed in Appendices A, B.2, C.2, D.2, E.2, F.2, and G.2 of this report, as completed. |
| **3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM** | |
| Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the FSAR. | Cask acceptance tests and maintenance program have been or will be developed in accordance with Chapter 9 of the CFSAR. These procedures are/will be listed in Appendices A, B.2, C.2, D.2, E.2, F.2, and G.2 of this report, as completed. |
| **4. QUALITY ASSURANCE** | |
| Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system. | Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with the Entergy QA Program, which meets the requirements of Part 72, Subpart G. |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

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| **5. HEAVY LOADS REQUIREMENTS** | |
| Each lift of an MPC, a HI-TRAC transfer cask, or any HI-STORM overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific regulatory review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.5 of Appendix A and/or Sections 3.4.6 and Section 3.5 of Appendix B to this certificate, as applicable. | Heavy-load lifts are controlled in accordance with a corporate procedure and each site's heavy load procedures. These procedures are:<br><br>**Corporate** - EN-MA-119, *Material Handling Program*<br><br>**ANO** – OP-1005.002, *Control of Heavy Loads*<br><br>**GGNS** – Standard #GGNS-CS-20, *Standard for Heavy Loads and Special Lifting Devices (NUREG-0612)*<br><br>**RBS** – MLP-7500, *Operation of the Spent Fuel Cask Crane*, GMP-0014, *Control of Load Lifting Equipment*, and DFS-0015, *Vertical Cask Transporter Operation*<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2&3** – To Be Determined<br><br>**VY** - To Be Determined<br><br>These procedures have been or will be reviewed to ensure lifts pertaining to cask activities comply with existing plant-specific heavy load requirements. The results of these reviews are/will be documented in site-specific appendices (applicable Section X.4.1.1) of this report, as completed. |

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| **6. APPROVED CONTENTS** | |
| Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate. | Each site-specific fuel selection procedure contains or will contain requirements to ensure spent fuel to be stored in the HI-STORM 100 cask meets the specifications contains in Appendix B of the CoC.<br>**ANO** - OP-1302.028, *Fuel Selection Criteria for Dry Storage*<br>**GGNS** – 17-S-02-111<br>**RBS** - REP-0061, *Fuel Selection for Dry Storage*<br>**IPEC-1** – To Be Determined<br>**IPEC-2** – 2-DCS-031-GEN, *Fuel Selection for Dry Cask Storage*<br>**IPEC-3** - To Be Determined<br>**VY** - To Be Determined |
| **7. DESIGN FEATURES** | No action required. |
| Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B to this certificate. | Entergy has reviewed each site, cask, and ancillary equipment against the requirement of CoC Appendix B. These reviews are or will be documented in site-specific appendices (applicable Section X.4.3) of this report, as completed. |
| **8. CHANGES TO THE CERTIFICATE OF COMPLIANCE** | |
| The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate. | No action required. Only the CoC holder may request the NRC to review and approve a change to the CoC. |

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| **9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE** | |
| **The air mass flow rate through the cask system will be determined by direct measurements of air velocity in the overpack cooling passages for the first HI-STORM Cask Systems placed into service by any user with a heat load equal to or greater than 20 kW. The velocity will be measured in the annulus formed between the MPC shell and the overpack inner shell. An analysis shall be performed that demonstrates the measurements validate the analytic methods and thermal performance predicted by the licensing-basis thermal models in Chapter 4 of the FSAR.**<br><br>Each first time user of a HI-STORM 100 Cask System Supplemental Cooling System (SCS) that uses components or a system that is not essentially identical to components or a system that has been previously tested, shall measure and record coolant temperatures for the inlet and outlet of cooling provided to the annulus between the HI-TRAC and MPC and the coolant flow rate. The user shall also record the MPC operating pressure and decay heat. An analysis shall be performed, using this information that validates the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling necessary.<br><br>Letter reports summarizing the results of each thermal validation test and SCS validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users. | If required, each Entergy site has or will monitor, record, and validate the heat transfer characteristics of the first HI-STORM cask placed into service under CoC Amendment 5 that meets the heat load threshold described in this CoC condition. Use of an SCS system not essentially identical to a previously tested SCS will be subjected to the test requirements specified in this CoC condition. Letter reports will be submitted to the NRC as required.<br><br>Description of these actions and results, if required, is or will be described in site-specific appendices (applicable Section X.4.1.2) of this report. |
| **10. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE** | |
| A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from | Each site has or will perform the required actions and describe the results in site-specific appendices (applicable Section X.4.1.3) of this report, as completed. |

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| CoC CONDITIONS (Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
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| the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:<br>a. Moving the MPC and the transfer cask into the spent fuel pool.<br>b. Preparation of the HI-STORM 100 Cask System for fuel loading.<br>c. Selection and verification of specific fuel assemblies to ensure type conformance.<br>d. Loading specific assemblies and placing assemblies into the MPC (using dummy fuel assembly), including appropriate independent verification.<br>e. Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool.<br>f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)<br>g. Operation of Supplemental Cooling System**, if applicable**<br>h. Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.<br>i. Transfer of the MPC from the transfer cask to the overpack.<br>j. Placement of the HI-STORM 100 Cask System at the ISFSI.<br>k. HI-STORM 100 Cask System unloading, including flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.) | |

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| **11.** When the Supplemental Cooling System is in operation to provide for decay heat removal in accordance with Section 3.1.4 of Appendix A the licensee is exempt from the requirements of 10 CR 72.236(f). | No action required |
| **12. AUTHORIZATION** | No action required. |
| The HI-STORM 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B. The HI-STORM 100 Cask System may be fabricated and used in accordance with any approved amendment to CoC No. 1014, listed in 10 CFR 72.214. Each of the licensed HI-STORM 100 System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility. | The Holtec HI-STORM 100 System, as approved for storage of spent fuel under the conditions specified in the Holtec CoC will be utilized at ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY as described in the site-specific appendices of this report. |

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| CoC APPENDIX A<br>TECHNICAL SPECIFICATIONS<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
|---|---|
| **1.0 USE AND APPLICATION** | No actions required. |
| **2.0 Section intentionally left blank** | No actions required. |
| **3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY AND SURVEILLANCE REQUIREMENT (SR) APPLICABILITY** | Specific compliance is documented in site-specific appendices (applicable Section X.4.2.1) of this report, as completed. |
| **4.0 Section intentionally left blank** | No actions required. |
| **5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS** | Specific compliance may be documented in site-specific appendices (applicable Section X.4.2.1) of this report, as completed. |
| 5.1 Deleted in Amendment 1 | |
| 5.2 Deleted in Amendment 1 | |
| 5.3 Deleted in Amendment 1 | |
| 5.4 <u>Radioactive Effluent Control Program</u><br><br>This program implements the requirements of 10 CFR 72.44(d). | Each facility has established a Radioactive Effluent Controls Program in accordance with its operating license TS. The specific TS references are:<br><br>**ANO-1** – TS 5.5.4<br>**ANO-2** – TS 6.8.4<br>**GGNS** – TS 5.5.4<br>**RBS** – TS 5.5.4<br>**IPEC-1** – TS 4.1<br>**IPEC-2** – TS 5.5.3<br>**IPEC-3** – TS 5.5.4<br>**VY** – TS 6.7.D<br><br>Each site's program has been or will be reviewed to ensure it meets the requirements of 72.44(d). These reviews are or will be documented in site-specific appendices (applicable Section X.4.2.2) of this report, as completed. |

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| **CoC APPENDIX A TECHNICAL SPECIFICATIONS (Amendment 2 with Amendment 5 changes in bold)** | **ASSESSMENT OF COMPLIANCE** |
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| a. The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC. | No action required. |
| b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations. | Entergy has or will review each facility's radiological environmental monitoring program (REMP) and has incorporated spent fuel storage cask activities into them, as appropriate. These reviews are/will be documented in site-specific appendices (applicable Section X.4.2.2) of this report, as completed. |
| c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3). | Entergy will submit an annual report to NRC in accordance with facility procedures. These procedures are:<br>**ANO** – OP-1052.022, *Radiological Effluents and Environmental Monitoring Program*<br>**GGNS** – 08-S-08-5, *Environmental Reporting*<br>**RBS** – RSP-0008, *Offsite Dose Calculation Manual*<br>**IPEC-1** – *Offsite Dose Calculation Manual*<br>**IPEC-2&3** – *Offsite Dose Calculation Manual*<br>**VY** – To Be Determined |

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| CoC APPENDIX A<br>TECHNICAL SPECIFICATIONS<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
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| 5.5 <u>Cask Transport Evaluation Program</u><br><br>This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc...) **or is being handled by a device designed in accordance with the increased safety factors of ANSI N14.6 and/or having redundant drop protection**.<br><br>Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport route conditions.<br><br>a. For free-standing OVERPACKS and the TRANSFER CASK, the following requirements apply:<br><br>1. The lift height above the transport route surface(s) shall not exceed the limits in Table 5-1 except as provided for in Specification 5.5.a.2. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than either Set A or Set B in HI-STORM FSAR Table 2.2.9.<br><br>2. For site-specific transport route surfaces that are not bounded by either the Set A or Set B parameters of FSAR Table 2.2.9, the program may determine lift heights by analysis based on the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g's at the top of the MPC fuel basket. These alternative analyses shall be commensurate with the drop analyses described in the Final Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled. | Each site has developed or will develop a Cask Transport Evaluation Program (if required) that evaluates site-specific transport route conditions. These programs are or will be documented in site-specific appendices (applicable Section X.4.2.3) of this report, as completed.<br><br>Note that the new "and/or" statement in the Amendment 5 text in the first paragraph of this TS section is inconsistent with TS Section 5.5.a.3 below it, which requires design in accordance with ANSI N14.6 <u>and</u> redundant drop protection features. It is recommended that the "and/or" statement not be implemented until this discrepancy is addressed by Holtec with a CoC amendment. |

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| 3. The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during transportation between the FUEL BUILDING and the CTF and/or the ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.<br><br>4. The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.<br><br>b. For the transport of OVERPACKS to be anchored to the ISFSI pad, the following requirements apply:<br><br>1. Except as provided in 5.5.b.2, user shall determine allowable OVERPACK lift height limit(s) above the transport route surface based on site-specific transport route conditions. The lift heights shall be determined by evaluation or analysis, based on limiting the design basis cask deceleration during a postulated drop event to $\leq$ 45 g's at the top of the MPC fuel basket. Evaluations and/or analyses shall be performed using methodologies consistent with those in the HI-STORM 100 FSAR.<br><br>2. The OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during transportation between the FUEL BUILDING and the CTF and/or ISFSI pad provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features. | |
| 5.6 Deleted in CoC Amendment 2. | |

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| 5.7 Radiation Protection Program | |
| 5.7.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below. | Radiation protection procedures governing cask handling, loading, movement, surveillance, and maintenance have been or will be developed in accordance with the CFSAR. These procedures are:<br><br>**ANO** – OP-1601.305, *Radiation Monitoring Requirements for Loading and Storage of the HI-STORM*, and OP-1601.306, *Radiation Monitoring Requirements for Un-Loading of the HI-STORM*<br><br>**GGNS** – 20-S-01-006<br><br>**RBS** – DFS-0006, *Radiological Monitoring Requirements for the HI-STORM 100 Dry Fuel Storage System*<br><br>**IPEC-1** – 0-RP-RWP-420, *Radiological Controls for Dry Cask Storage*<br><br>**IPEC-2&3** - 0-RP-RWP-420, *Radiological Controls for Dry Cask Storage*<br><br>**VY** - To Be Determined<br><br>See detailed information in site-specific appendices (applicable Section X.3.4) of this report, as completed. |
| 5.7.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents. | Entergy has or will review each facility's radiological environmental monitoring program and has or will incorporate spent fuel storage cask activities into them, as appropriate. These reviews are/will be documented in site-specific appendices (applicable Section X.4.2.2) of this report, as completed. |

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|---|---|
| 5.7.3 Based on the analysis performed pursuant to Section 5.7.2, the licensee shall establish individual cask surface dose rate limits for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:<br>a. The top of the TRANSFER CASK and the OVERPACK.<br>b. The side of the TRANSFER CASK and OVERPACK<br>c. The inlet and outlet ducts on the OVERPACK | Calculations have been or will be performed to establish design surface dose rates for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be utilized in the site radiological procedures. These design values are also utilized in calculations that describe the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents. These reviews are/will be described in site-specific appendices (applicable Section X.4.2.4) of this report, as completed. |
| 5.7.4 Notwithstanding the limits established in Section 5.7.3, the measured dose rates on a loaded OVERPACK shall not exceed the following values:<br>a. **30** mrem/hr (gamma + neutron) on the top of the OVERPACK<br>b. **300** mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts<br>5.7.5 The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in Section 5.7.8 for comparison against the limits established in Section 5.7.3 or Section 5.7.4, whichever are lower. | Procedures to control these activities are/will be listed in site-specific appendices (applicable Section X.2) of this report, as completed. |

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| 5.7.6 If the measured surface dose rates exceed the lower of the two limits established in Section 5.7.3 or Section 5.7.4, the licensee shall:<br><br>a. Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.<br><br>b. Perform a written evaluation to verify whether placement of the as-loaded OVERPACK at the ISFSI will cause the dose limits of 10 CFR 72.104 to be exceeded.<br><br>c. Perform a written evaluation within 30 days to determine why the surface dose rate limits were exceeded.<br><br>5.7.7 If the evaluation performed pursuant to Section 5.7.6 shows that the dose limits of 10 CFR 72.104 will be exceeded, the **MPC** shall not be placed into storage until appropriate corrective action is taken to ensure the dose limits are not exceeded.<br><br>5.7.8 TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:<br><br>a. A minimum of four (4) dose rate measurements shall be taken on the side of the TRANSFER CASK approximately at the cask mid-height plane. The measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.<br><br>b. A minimum of four (4) TRANSFER CASK top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid.<br><br>c. A minimum of twelve (12) dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask | |

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|---|---|
| mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.<br><br>d. A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top concrete shield, 90 degrees apart around the circumference of the lid.<br><br>e. A dose rate measurement shall be taken on contact at the surface of each inlet and outlet vent duct screen **of the OVERPACK**. | |

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| CoC APPENDIX B<br>APPROVED CONTENTS & DESIGN FEATURES<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
|---|---|
| **1.0 DEFINITIONS** | No action required. |
| **2.0 APPROVED CONTENTS** | No action required. |
| 2.1 Fuel Specifications and Loading Conditions | |
| 2.1.1 <u>Fuel To Be Stored In The HI-STORM 100 SFSC System</u><br><br>a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM 100 SFSC System.<br><br>b. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the decay heat generation limit for the stainless steel clad fuel assemblies.<br><br>**c.** For MPCs partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining ZR clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies.<br><br>**d.** All BWR fuel assemblies may be stored with or without ZR channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without ZR or stainless steel channels. | Fuel to be stored in casks is governed by individual site selection procedures ;<br><br>**ANO** – OP-1302.028, *Fuel Selection Criteria For Dry Storage*<br><br>**GGNS** – 17-S-02-111<br><br>**RBS** - REP-0061, *Fuel Selection for Dry Storage*<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2** – 2-DCS-031-GEN, *Fuel Selection for Dry Cask Storage*<br><br>**IPEC-3** - To Be Determined<br><br>**VY** - To Be Determined |

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| CoC APPENDIX B<br>APPROVED CONTENTS & DESIGN FEATURES<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
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| 2.1.2 <u>Uniform Fuel Loading</u><br><br>Any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions related to DAMAGED FUEL, FUEL DEBRIS, AND NON-FUEL HARDWARE specified in the CoC. | Fuel loading is governed by individual site selection procedures.:<br><br>**ANO** – OP-1302.028, *Fuel Selection Criteria For Dry Storage*<br><br>**GGNS** - 17-S-02-111<br><br>**RBS** - REP-0061, *Fuel Selection for Dry Storage*<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2** – 2-DCS-031-GEN, *Fuel Selection for Dry Cask Storage*<br><br>**IPEC-3** - To Be Determined<br><br>**VY** - To Be Determined |
| 2.1.3 <u>Regionalized Fuel Loading</u><br><br>Users may choose to store fuel using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Regionalized loading is limited to those fuel assemblies with ZR cladding. Figures 2.1-1 through 2.1-4 define the regions for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, and MPC-68FF models, respectively. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Section 2.4.2. Fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3. | Regionalized fuel loading is governed by individual site selection procedures:<br><br>**ANO** – OP-1302.028, *Fuel Selection Criteria For Dry Storage*<br><br>**GGNS** - 17-S-02-111<br><br>**RBS** - REP-0061, *Fuel Selection for Dry Storage*<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2** – 2-DCS-031-GEN, *Fuel Selection for Dry Cask Storage*<br><br>**IPEC-3** - To Be Determined<br><br>**VY** - To Be Determined |

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| CoC APPENDIX B<br>APPROVED CONTENTS & DESIGN FEATURES<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
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| 2.2 Violations<br><br>If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:<br><br>2.2.1 The affected fuel assemblies shall be placed in a safe condition.<br><br>2.2.2 Within 24 hours, notify the NRC Operations Center.<br><br>2.2.3 Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence. | Violations are reported in accordance with NMM Procedure EN-LI-108, *Event Notification and Reporting*. |
| 2.3 Not used at this time. | |
| 2.4 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel<br><br>This section provides the limits on ZR-clad fuel assembly decay heat, burnup, and cooling time for storage in the HI-STORM 100 System. **The method** to calculate the limits and verify compliance, including examples, is provided in Chapter 12 of the HI-STORM 100 FSAR.<br><br>2.4.1 Uniform Fuel Loading Decay Heat Limits for ZR-Clad Fuel<br><br>Table 2.4-1 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model. | Fuel to be stored in casks is governed by individual site selection procedures ;<br><br>**ANO** – OP-1302.028, *Fuel Selection Criteria For Dry Storage*<br><br>**GGNS** - 17-S-02-111<br><br>**RBS** - REP-0061, *Fuel Selection for Dry Storage*<br><br>**IPEC-1** – To Be Determined<br><br>**IPEC-2** – 2-DCS-031-GEN, *Fuel Selection for Dry Cask Storage*<br><br>**IPEC-3** - To Be Determined<br><br>**VY** - To Be Determined |

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| CoC APPENDIX B<br>APPROVED CONTENTS & DESIGN FEATURES<br>(Amendment 2 with Amendment 5 changes in bold) | ASSESSMENT OF COMPLIANCE |
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| 2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel<br><br>The maximum allowable decay heat per fuel storage location for fuel in regionalized loading **is determined using the following equations:**<br><br>$\mathbf{Q(X) = 2 \times Q_o / (1 + X^y)}$<br><br>$y = 0.23 / X^{0.1}$<br><br>$\mathbf{q_2 = Q(X) / (n_1 \times X + n_2)}$<br><br>$\mathbf{q_1 = q_2 \times X}$<br><br>**Where:**<br><br>**$Q_o$ = Maximum uniform storage MPC decay heat (34 kW)**<br><br>**X = Inner region to outer region assembly decay heat ratio ($0.5 \leq X \leq 3$)**<br><br>**$n_1$ = Number of storage locations in inner region from Table 2.4-2**<br><br>**$n_2$ = Number of storage locations in outer region from Table 2.4-2** | |

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| 2.4.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel<br><br>The maximum allowable fuel assembly average burnup varies with the following parameters:<br><br>• Minimum fuel assemble cooling time<br>• Maximum fuel assembly decay heat<br>• Minimum fuel assembly average enrichment<br><br>The maximum allowable ZR-clad fuel assembly average burnup for a given MINIMUM ENRICHMENT is calculated as described below for minimum cooling times between 3 and 20 years using the maximum permissible decay heat determined in Section 2.4.1 or 2.4.2.<br><br>Different fuel assembly average burnup limits may be calculated for different minimum enrichments (by individual fuel assembly) for use in choosing the fuel assemblies to be loaded into a given MPC.<br><br>2.4.3.1 Choose a fuel assembly minimum enrichment, $E_{235}$.<br><br>2.4.3.2 Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below.<br><br>$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$<br><br>Where:<br><br>Bu = Maximum allowable average burnup per fuel assembly (MWD/MTU)<br><br>q = Maximum allowable decay heat per fuel storage location determined in Section 2.4.1 or 2.4.2 (kW) | |

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| $E_{235}$ = Minimum fuel assembly average enrichment (wt. % $^{235}U$)<br><br>A through G = Coefficients from Tables 2.4-3 and 2.4-4 for the applicable fuel assembly array/class and minimum cooling time<br><br>2.4.3.3 Calculated burnup limits shall be rounded down to the nearest integer.<br><br>2.4.3.4 Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR [fuel] must be reduced to be equal to these values.<br><br>2.4.3.5 Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a cooling time of 4.5 years may be interpolated between those burnups calculated for 4 years and 5 years. | |
| 2.4.4 When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat for both the fuel assembly and any NON-FUEL HARDWARE, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit. | |

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| **3.0 DESIGN FEATURES** | |
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| 3.1 Site<br><br>3.1.1 <u>Site Location</u><br><br>The HI-STORM 100 Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K. | No action required. |
| 3.2 Design Features Important for Criticality Control<br><br>3.2.1 <u>MPC-24</u><br>1. Flux trap size: ≥ 1.09 in.<br>2. $^{10}B$ loading in the neutron absorbers: ≥ 0.0267 g/cm$^2$ (Boral) and ≥ 0.0223 g/cm$^2$ (METAMIC)<br><br>3.2.2 <u>MPC-68 and MPC-68FF</u><br>1. Fuel cell pitch: ≥ 6.43 in.<br>2. $^{10}B$ loading in the neutron absorbers: ≥ 0.0372 g/cm$^2$ (Boral) and ≥ 0.0310 g/cm$^2$ (METAMIC)<br><br>3.2.3 <u>MPC-68F</u><br>1. Fuel cell pitch: ≥ 6.43 in.<br>2. $^{10}B$ loading in the Boral neutron absorbers: ≥ 0.01 g/cm$^2$<br><br>3.2.4 <u>MPC-24E and MPC-24EF</u><br>1. Flux trap size:<br>i. Cells 3, 6, 19, and 22: ≥ 0.776 inch<br>ii. All Other Cells: ≥ 1.076 inches<br>2. $^{10}B$ loading in the neutron absorbers: ≥ 0.0372 g/cm$^2$ (Boral) and ≥ 0.0310 g/cm$^2$ (METAMIC) | Each MPC is manufactured and certified that it meets design requirements in accordance with the HI-STORM 100 CoC. This documentation is controlled by NMM Procedure ENS-DC-160, *Dry Fuel Storage Document Control. (A fleet procedure pertaining to this issue will be developed in the future.)* Each site will document canister type compliance as necessary including determination of fuel spacer requirements. Each site has or will describe the results in site-specific appendices (applicable Section X.4.3.1) of this report as applicable and as completed. |

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| 3.2.5 MPC-32 and MPC-32F<br>1. Fuel cell pitch: ≥ 9.158 in.<br>2. $^{10}B$ loading in the neutron absorbers: ≥ 0.0372 g/cm$^2$ (Boral) and ≥ 0.0310 g/cm$^2$ (METAMIC)<br>3.2.6 Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.<br>3.2.7 The $B_4C$ content in METAMIC shall be ≤ 33.0 wt.%<br>3.2.8 Neutron Absorber Tests<br>Section 9.1.5.3 of the HI-STORM 100 FSAR is hereby incorporated by reference into the HI-STORM 100 CoC. The minimum 10B for the neutron absorber shall meet the minimum requirements for each MPC model specified in Sections 3.2.1 through 3.2.5 above. | |
| 3.3 Codes and Standards<br>The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing Code for the HI-STORM 100 System **MPCs, OVERPACKS, and TRANSFER CASKs**, as clarified in Specification 3.3.1 below, except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections, provided a written reconciliation of the later edition against the 1995 edition, including addenda, is performed by the certificate holder. American Concrete Institute (ACI) 349-85 is the governing Code for plain concrete as clarified in Appendix 1.D of the Final Safety Analysis Report for the HI-STORM 100 Cask System. | No action required |

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| 3.3.1 <u>Alternatives to Codes, Standards, and Criteria</u><br><br>Table 3-1 lists approved alternatives to the ASME Code for the design of the **MPCs, OVERPACKS, and TRANSFER CASKs of the** HI-STORM 100 Cask System. | No action required. |
| 3.3.2 <u>Construction/Fabrication Alternatives to Codes, Standards, and Criteria</u><br><br>Proposed alternatives to the ASME Code, Section III, 1995 Edition with Addenda through 1997 including modifications to the alternatives allowed by Specification 3.3.1 may be used on a case-specific basis when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that:<br><br>1. The proposed alternatives would provide an acceptable level of quality and safety, or<br><br>2. Compliance with the specified requirements of the ASME Code, Section III, 1995 Edition with Addenda through 1997, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.<br><br>Requests for alternatives shall be submitted in accordance with 10 CFR 72.4 | No action required. Only the CoC holder can request revised or new Code alternatives. |
| 3.4 Site-Specific Parameters and Analyses<br><br>Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows: | |
| 1. The temperature of 80°F is the maximum average yearly temperature. | The maximum average yearly temperatures have or will be evaluated and described in site-specific appendices (applicable Section X.4.3.2.1) of this report, as completed. |

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| 2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40°F and less than 125°F. | The temperature extremes for ANO, GGNS, and RBS have been or will be evaluated and described in site-specific appendices (applicable Section X.4.3.2.2) of this report as completed. |
| 3. a. **For storage in freestanding OVERPACKS, t**he resultant horizontal acceleration [vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site], $G_H$, and vertical ZPA, $G_V$, on the top surface of the ISFSI pad, expressed as fractions of "g", shall satisfy the following inequality:<br><br>$G_H + \mu G_V \leq \mu$<br><br>where μ is either the Coulomb friction coefficient for the cask/ISFSI pad interface or the ratio r/h, where "r" is the radius of the cask and "h" is the height of the cask center-of gravity above the ISFSI pad surface. The above inequality must be met for both definitions of μ, but only applies to ISFSIs where the casks are deployed in a free standing configuration. Unless demonstrated by appropriate testing that a higher coefficient of friction value is appropriate for a specific ISFSI, the value used shall be 0.53. If acceleration time-histories on the ISFSI pad surface are available, $G_H$ and $G_V$ may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality shall be evaluated at each time step in the acceleration time history over the total duration of the seismic event.<br><br>If this static equilibrium based inequality cannot be met, a dynamic analysis of the cask/ISFSI pad assemblage with appropriate recognition of soil/structure interaction effects shall be performed to ensure that the casks will not tip over or undergo excessive sliding under the site's Design Basis Earthquake. | As documented in site-specific appendices (applicable Section X.4.3.2.3) of this report when completed, $G_H$ and $G_V$ satisfy the inequality for ANO, GGNS, RBS, IPEC, and VY for use of free-standing casks. |

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| b. For free-standing casks, under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will not result in cask tipover or cause a cask to fall off the pad. In addition, impact between casks should be precluded, or should be considered an accident for which the maximum g-load experienced by the stored fuel shall be limited to 45 g's. | Calculations have been or will be performed to establish design interface friction due to environmental conditions such as icing for the HI-STORM OVERPACK to be utilized in demonstrate that the earthquake will not result in cask tipover or cause a cask to fall off the pad. These design values will also be utilized in calculations to show that the maximum g-load experienced by the stored fuel will be limited to 45 g's or that there will be no contact between casks with the site's ISFSI configuration, considering the planned number of casks to be deployed and the casks array. These reviews are/will be described in site-specific appendices (applicable Section X.4.3.2) of this report, as completed. |
| c. For those ISFSI sites with design basis seismic acceleration values *that may overturn or cause excessive sliding of* free-standing casks, the HI-STORM 100 System shall be anchored to the ISFSI pad. The site seismic characteristics and the anchorage system shall meet the following requirements:<br><br>i. The site acceleration response spectra at the top of the ISFSI pad shall have ZPAs that meet the following inequalities:<br><br>$G_H \leq 2.12$<br><br>AND<br><br>$G_V \leq 1.5$<br><br>where:<br><br>$G_H$ is the vectorial sum of the two horizontal ZPAs at a three-dimensional seismic site (or the horizontal ZPA at a two-dimensional site) and $G_V$ is the vertical ZPA. | As documented in site-specific appendices (applicable Section X.4.3.2.3) of this report when completed, $G_H$ and $G_V$ satisfy the inequality for ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY for use of free-standing casks. This section is not applicable to Entergy sites. |

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| ii. Each HI-STORM 100 dry storage cask shall be anchored with twenty-eight (28), 2-inch diameter studs and compatible nuts of material suitable for the expected ISFSI environment. The studs shall meet the following requirements:<br><br>Yield Strength at Ambient Temperature: ≥ 80 ksi<br><br>Ultimate Strength at Ambient Temperature: ≥ 125 ksi<br><br>Initial Tensile Pre-Stress: ≥ 55 ksi AND ≤ 65 ksi<br><br>NOTE: The above anchorage specifications are required for the seismic spectra defined in item 3.4.3.b.i. Users may use fewer studs or those of different diameter to account for site-specific seismic spectra less severe than those specified above. The embedment design shall comply with Appendix B of ACI-349-97. A later edition of this Code may be used, provided a written reconciliation is performed.<br><br>iii. Embedment concrete Compressive Strength: ≥ 4,000 psi at 28 days | |
| 4. The analyzed flood condition of 15-fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded. | The analyzed flood conditions for ANO, GGNS, RBS, IPEC-1, IPEC-2&3, and VY meet the criteria as documented or will be documented in site-specific appendices (applicable Section X.4.3.2.4) of this report. |
| 5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK. | Each Entergy site has evaluated or will evaluate the impact of the ISFSI on each facility's fire protection plan. These reviews will be documented in site-specific appendices (applicable Section X.4.3.2.5) of this report. |

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| 6. a. For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to ≤ 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features. | Each facility's ISFSI cask storage pad has been or will be designed to adequately support the static load of the storage casks. The storage pad was or will be constructed as described in site-specific appendices (applicable Section X.4.3.2.6) of this report, as completed. Engineering evaluations summarized in each appendix will document the structural adequacy of each facility's train bay (cask transfer area) and roadway to support the static load of the casks. |
| b. For anchored casks, the ISFSI pad shall be designed to meet the embedment requirements of the anchorage design. A cask tip-over event for an anchored cask is not credible. The ISFSI pad shall be verified by analysis to limit cask deceleration during a design basis drop event to ≤ 45 g's at the top of the MPC fuel basket, except as provided for in this paragraph below. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features. | Not applicable to Entergy sites. |
| 7. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category. | Engineered features to meet the requirements of §72.104(a) are not utilized at Entergy sites. |

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| 8. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures ≥ 0°F. | Each site's loading, unloading, and transport operations activities are or will be conducted in accordance with site procedures. Such activities are prohibited when the ambient temperature of the working area is < 0°F. A list of site procedures are or will be in site-specific appendices (applicable Section X.2) of this report, as completed.<br><br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.2.8) of this report, as completed. |
| 9. For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e., longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal in available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective. | Site-specific design basis for blockage consideration is or will be discussed in site-specific appendices (applicable Section X.4.3.2.9) of this report, as completed. |

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| 3.5 Cask Transfer Facility (CTF) | |
| 3.5.1 <u>TRANSFER CASK and MPC Lifters</u><br><br>Lifting of a loaded TRANSFER CASK and MPC using devices that are not integral to structures governed by 10 CFR Part 50 shall be performed with a CTF that is designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and the below clarifications. The CTF structure requirements below do not apply to heavy loads bounded by the regulations of 10 CFR Part 50, **to the loading of an OVERPACK in a belowground restraint system which permits MPC transfer near grade level and does not require an aboveground CTF**. | Lifting of a loaded TRANSFER CASK and MPC outside of structures governed by 10 CFR Part 50 will not be performed at any Entergy site. None of the sites will utilize a CTF with the Holtec HI-STORM 100 Cask System. In accordance with NUREG-0612 requirements, the RBS MPC transfer will occur utilizing the Fuel Building Crane that travels outside but is still part of the 10 CFR 50 structure. This is not considered a "Cask Transfer Facility" as defined in the CoC. |
| 3.5.2 <u>CTF Structure Requirements</u><br><br>3.5.2.1 <u>Cask Transfer Station and Stationary Lifting Devices</u><br><br>1. The metal weldment structure of the CTF structure shall be designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. The applicable loads, load combinations, and associated service condition definitions are provided in Table 3-***3***. All compression-loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF.<br><br>2. If a portion of the CTF structure is constructed of reinforced concrete, then the factored load combinations set forth in ACI-318 (89) for the loads defined in Table 3-***2*** shall apply. | Lifting of a loaded TRANSFER CASK and MPC outside of structures governed by 10 CFR Part 50 will not be performed at any Entergy site. None of the sites will utilize a CTF with the loaded Holtec HI-STORM 100 Cask System components. |

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| 3. The TRANSFER CASK and MPC lifting device used with the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1.<br><br>4. The CTF shall be designed, constructed, and evaluated to ensure that if the MPC is dropped during inter-cask transfer operations, its confinement boundary would not be breached. This requirement applies to CTFs with either stationary or mobile lifting devices. | |
| 3.5.2.2 <u>Mobile Lift Devices</u><br><br>If a mobile lifting device is used as the lifting device, in lieu of a stationary lifting device, is shall meet the guidelines of NUREG-0612, Section 5.1, with the following clarifications:<br><br>1. Mobile lifting devices shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6(1)(a) and shall be capable of stopping and holding the load during a Design Basis Earthquake (DBE) event.<br><br>2. Mobile lifting devices shall conform to meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes." | Lifting of a loaded TRANSFER CASK and MPC outside of structures governed by 10 CFR Part 50 will not be performed at any Entergy site. None of the sites will utilize a mobile crane with loaded Holtec HI-STORM 100 Cask System components. Lifting loaded HI-STORMs will occur at ANO, GGNS, IPEC-1, IPEC-2&3, RBS, and VY as described in site-specific appendices (applicable Section X.4.2.3) of this report, as completed. |

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| 3. Mobile cranes are not required to meet the requirements of NUREG-0612, Section 5.1.6(2) for new cranes.<br>4. Horizontal movements of the TRANSFER CASK and MPC using a mobile crane are prohibited. | |
| 3.6 Forced Helium Dehydration System<br>3.6.1 System Description<br>Use of the Forced Helium Dehydration (FHD) system, (a closed-loop system) is an alternative to vacuum drying the MPC for moderate burnup fuel ($\leq$ 45,000 MWD/MTU) and mandatory for drying MPCs containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.6.2.<br>3.6.2 Design Criteria<br>3.6.2.1 The temperature of the helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.<br>3.6.2.2 The pressure in the MPC cavity space shall be $\leq$ 60.3 psig (75 psia).<br>3.6.2.3 The hourly recirculation rate of helium shall be $\geq$ 10 times the nominal helium mass backfilled into the MPC for fuel storage operations. | Each site's loading and unloading operations activities including use of a forced helium dehydration (FHD) system are or will be conducted in accordance with site procedures. A list of site procedures are or will be in site-specific appendices (applicable Section X.2) of this report, as completed.<br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.3) of this report, as completed. |

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| 3.6.2.4 The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. The limit is met if the gas temperature at the demoisturer outlet is verified by measurement to remain $\leq$ 21°F for a period of 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain $\leq$ 22.9°F for $\geq$ 30 minutes.<br>3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point $\leq$ 120°F.<br>3.6.2.6 The demoisturizing module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in FSAR Appendix 2.B) has been completed.<br>3.6.2.7 The helium circulator shall be sized to affect the minimum flow rate of circulation required by these design criteria.<br>3.6.2.8 The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MOC meets these design criteria. | |

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| 3.6.3 Fuel Cladding Temperature<br><br>A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage. | In accordance with the HI-STORM 100 CFSAR, an analysis (Holtec document HI-2022966, *Forced Helium Dehydrator Sourcebook*) was performed demonstrating compliance with all design criteria in CFSAR Section 2.B.2. The FHD system was shown to satisfy all design criteria requirements therefore the FHD design is in compliance with the CoC requirements.<br><br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.3) of this report, as completed. |
| 3.6.4 Pressure Monitoring During FHD Malfunction<br><br>During an FHD malfunction event, described in HI-STORM 100 FSAR Section 11.1 as a loss of helium circulation, the system pressure must be monitored to ensure that the conditions listed therein are met. | Each site's loading operations activities include or will include pressure monitoring of the forced helium dehydration (FHD) system operation conducted in accordance with site procedures. A list of site procedures are or will be in site-specific appendices (applicable Section X.2) of this report, as completed.<br><br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.3) of this report, as completed. |

# VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

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| 3.7 Supplemental Cooling System<br><br>3.7.1 System Description<br><br>The SCS is a water circulation system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. Use of the Supplemental Cooling System (SCS) is required for post-backfill HI-TRAC operations of an MPC containing one or more high-burnup (> 45,000 MWD/MTU) fuel assemblies. The SCS shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.7.2. | Each site's loading and unloading operations activities include or will include as appropriate use of the Supplemental Cooling system in accordance with site procedures. A list of site procedures are or will be in site-specific appendices (applicable Section X.2) of this report, as completed.<br><br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.4) of this report, as completed. |
| 3.7.2 Design Criteria<br><br>3.7.2.1 **Not used.**<br><br>3.7.2.2 **If water is used as the coolant, the system** shall be sized to limit the coolant temperature **to below 180°F under steady-state conditions** for the design basis heat load at an ambient air temperature of 100°F. **Any electric motors** shall **have** a backup power supply for uninterrupted operation.<br><br>3.7.2.3 The system shall utilize a contamination-free fluid medium in contact with the external surfaces of the MPC and inside surfaces of the HI –TRAC transfer cask to minimize corrosion.<br><br>3.7.2.4 All passive components such as tubular heat exchangers, manually operated valves and fittings shall be designed to applicable standards (TEMA, ANSI). | NOTE: The new provisions in Sections 3.7.2.2 and 3.7.2.6 suggesting a non-water (i.e., air) cooling system may be used are in conflict with Section 3.7.1, which still defines the SCS as "a water circulation system." Non-water systems should be avoided until Holtec resolves this discrepancy with a CoC amendment. |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

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| 3.7.2.5 The heat dissipation capacity of the SCS shall be equal to or greater than the minimum necessary to ensure that the peak cladding temperature is below 400°C (752°F). All heat transfer surfaces in heat exchangers shall be assumed to be fouled to the maximum limits specified in a widely used heat exchanger equipment standard such as the Standards of Tubular Exchanger Manufacturers Association.<br>3.7.2.6 The coolant utilized to extract heat from the MPC shall be high purity water **or air**. Antifreeze may be used to prevent water from freezing if warranted by operating conditions.<br>3.7.2.7 All pressure boundaries (as defined in the ASME Boiler and Pressure Vessel Code, Section VIII Division 1) shall have pressure ratings that are greater than the maximum system operating pressure by at least 15 psi.<br>3.7.2.8 All ASME Code components shall comply with Section VIII Division 1 of the ASME Boiler and Pressure Vessel Code.<br>3.7.2.9 All gasketed and packed joints shall have a minimum design pressure rating of the pump shut-off pressure plus 15 psi. | |

## VI. HI-STORM 100 CERTIFICATE OF COMPLIANCE REQUIREMENTS

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| 3.8. Combustible Gas Monitoring During MPC Lid Welding<br><br>During MPC lid-to-shell welding operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present in the welding area. | Each site's loading and unloading operations activities include or will include monitoring of combustible gas during welding activities in accordance with site procedures. A list of site procedures are or will be in site-specific appendices (applicable Section X.2) of this report, as completed.<br><br>Additional detail may be found in site-specific appendices (applicable Section X.4.3.5) of this report, as completed |

# APPENDIX A

# LIST OF CORPORATE POLICIES & PROCEDURES

# APPENDIX A

## LIST OF CORPORATE POLICIES & PROCEDURES

The corporate procedures listed below support spent fuel cask storage activities at the Entergy ISFSIs.

1. CEP-NDE-0110, *Program Section for Certification of NDE Personnel*
2. CEP-NDE-0112, Certification of Visual Examination Personnel
3. CEP-NDE-0640, Non-Section XI *Liquid Penetrant Examination (PT)*
4. CEP-NDE-1070, *Helium Leak Detection for Dry Fuel Storage*
5. EN-AD-103, *Document Control and Records Management Activities*
6. EN-EC-100, *Guidelines for Implementation of the Employee Concerns Program*
7. EN-EV-115, *Environmental Reviews and Evaluations*
8. EN-LI-100, *Process Applicability Determination*
9. EN-LI-101, *10 CFR 50.59 Review Program*
10. EN-LI-102, *Corrective Action Process*
11. EN-LI-106, *NRC Correspondence*
12. EN-LI-108, *Event Notification and Reporting*
13. EN-LI-112, *10 CFR 72.48 Review Program*
14. EN-LI-113, *License Basis Document Change Process*
15. EN-LI-115, *HI-STORM 100 Independent Spent Fuel Storage Installation Licensing Document Preparation and Control*
16. EN-MA-101, *Conduct of Maintenance*
17. EN-MA-102, *Inspection Program*
18. EN-MA-119, *Material Handling Program*
19. EN-NF-101, *Nuclear Fuel Program*
20. EN-NF-104, *Special Nuclear Materials Program*
21. EN-NF-200, *Special Nuclear Material Control*

22. EN-NF-201, *Special Nuclear Materials Reporting*
23. EN-NS-102, *Fitness for Duty Program*
24. EN-NS-112, *Medical Program*
25. EN-QV-104, *Quality Assurance Program Manual Control*
26. EN-QV-111, *Training and Certification of Inspection/Verification and Examination Personnel*
29. ENN-NDE-2.10, *Certification of NDE Personnel*
30. ENN-NDE-2.12, *Certification of Visual Testing (VT) Personnel*
31. ENN-NDE-9.40, *Liquid Penetrant Examination (PT)*
32. ENN-NDE-10.07, *Visual Inspection Procedure for the HI-STORM 100 Dry Cask Fuel Storage System*
33. ENS-DC-160, *Dry Fuel Storage Document Control*
34. ENS-HR-135, *Disciplinary Action*

# APPENDIX E

# IPEC-SPECIFIC INFORMATION
# UNIT 1

**(To be provided prior to cask loading activities at IPEC Unit 1.)**

# APPENDIX F

# IPEC-SPECIFIC INFORMATION UNITS 2 and 3

**(To be provided prior to cask loading activities at IPEC Unit 2 and 3.)**

# APPENDIX G

# VY-SPECIFIC INFORMATION

**(To be provided prior to cask loading activities at VY.)**

**ENTERGY NUCLEAR**
**10 CFR 72.212 EVALUATION REPORT**
**SITE SPECIFIC APPENDIX E**
**REVISION 1**

**Indian Point Energy Center Unit 1**
**Safety Related: ________ Yes**

**__X__ No**

Prepared by: John D. McCann / [signature] / Proj Date 6/23/2008
Print Name/Signature/Department

Reviewed by: John Jawicki / [signature] / MTC Date 6/23/08
Print Name/Signature/Department

Approved by: D. MAYER / [signature] / admin Date: 6.24.08
Print Name/Signature/Department

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# SECTION E.1
# GENERAL INFORMATION

## E.1.1 Introduction

This document is prepared pursuant to 10 CFR 72.212(b) to facilitate the review and approvals required to utilize the Holtec HI-STORM 100 dry fuel storage system at the Indian Point Energy Center (IPEC). The storage system components described in this document are unique to Indian Point Unit One, although the fuel will be stored at the Independent Spent Fuel Storage Installation (ISFSI) which is common to all three of the IPEC Units.

IPEC Unit 2 placed its first dry cask storage cask on the IPEC ISFSI on January 11, 2008. The details of the ISFSI pad design, a portion of the cask haul route, and the preparation and implementation of 10 CFR 72 and other licensing requirements applicable to supporting departments, i.e. Emergency Planning, Operations, Radiological Protection, etc. are documented in detail in the Unit 2 10CFR72.212 report, Appendix F.

The IP-1 spent fuel, prior to 2007, has been stored in the West Spent Fuel Pool in the IP-1 Fuel Handling Building (FHB). Continued on-site storage of spent fuel will be necessary until the Department of Energy (DOE) begins to accept fuel for permanent disposal as required by the Nuclear Waste Policy Act of 1982 and the contract signed by Entergy for IPEC.

Unit 1 has been shutdown since 1974 and has been maintained in a "SAFSTOR" mode since then. During operation from 1962, 120 spent fuel elements from "core A" and 124 fuel elements from "core B" were transported to the West Valley, NY. An additional 160 fuel elements from subsequent "core B" operation remain stored underwater in the West Fuel Pool. To preclude any leakage from the Unit 1 spent fuel pool, IPEC decided to place the remaining 160 fuel assemblies into dry storage and to drain the fuel pool of its water inventory.

Because of the nature of the IP-1 fuel design and the limitations of the physical facilities in the FHB, a specially designed Holtec HI-STORM-100 (HI-STORM 100S-185) system was selected for use at IP-1 and is capable of containing up to 32 IP-1 fuel assemblies. This design also accommodates up to 32 damaged fuel cans (DFC) and the loaded Multipurpose Canister and HI-TRAC transfer cask does not exceed 150,000 lbs (75 tons) in weight. These design peculiarities make the HI-STORM 100S-185 unique to Indian Point 1.

The general construction and assembly of the HI-STORM 100S-185 system is similar to other Holtec designs in that the MPC consists of a stainless steel canister with a welded base plate and lid, which is placed inside a coated carbon steel and concrete overpack that is placed on a concrete pad for storage. The IPEC Independent Spent Fuel Storage Installation (ISFSI) is designed to accommodate the five overpacks from Unit 1 as well as additional overpacks from both Units 2 and 3.

The Holtec HI-STORM 100S-185 system is designed, licensed, fabricated, and deployed on site under the regulations in 10 CFR 72 Sub Part K. The 10 CFR 72 regulations grant a general license for spent fuel storage in an NRC-certified dry cask storage system to any holder of a 10 CFR 50 license. The HI-Storm System was originally certified by the NRC in accordance with 10CFR72 Subpart L in May 2000. The HI-STORM 100 Certificate of Compliance (CoC) 72-1014 has been amended several times since that time. Amendment 4 to the Certificate, dated January 8, 2008, and the corresponding Revision 6 to the FSAR, issued February 7, 2008, describes the HI-STORM 100S-185 System which is unique to IP-1.

A system wide 10 CFR 72.212 evaluation report is maintained for Entergy Plants that utilize the HI-STORM 100 cask system. The specific conditions for the use of the HI-STORM 100 system at Indian Point 1 are addressed in this Appendix E. Since the first HI-STORM Systems to be loaded on site are from the Unit 2 fuel pool, the Site ISFSI parameters are addressed in Appendix F, which are the specifics for the Indian Point Units 2 and 3 fuel-loading campaigns.

The IP-1 site implementation review can be broken into three main components: loading and unloading, onsite transportation to the ISFSI, and off normal and accident conditions. Each of these areas is discussed in this appendix to provide a background for the required reviews. Other 10 CFR 72.212 requirements are documented as appropriate in clearly identified appendix sections

**E.1.2System Components**

The Holtec International HI-STORM 100 System for dry spent fuel storage consists of these major components or groups of components: The following listing is not all inclusive.

1) A multi-purpose canister (MPC) that contains the fuel.

2) A transfer cask (HI-TRAC), and HI-TRAC lift yoke that is used to move the HI-TRAC/MPC assemblage containing fuel from the cask loading pool to the preparation area and ultimately to the steel and concrete overpack (HI-STORM).

3) The steel and concrete overpack (HI-STORM) that provides the natural ventilation heat removal, radiation shielding, and structural protection for the MPC during storage operations.

4) A mating device containing a slide assembly used to mate the HI-STORM and the HI-TRAC, remove the HI-TRAC pool lid, and allow the transfer of the loaded MPC from the HI-TRAC to the HI-STORM.

5) A vertical cask transporter (VCT) used to transfer the loaded HI-TRAC to the Unit 2 Fuel Handling Building, and then after the MPC is loaded into the HI-STORM overpack which will be done using the Unit 2 FSB Gantry crane, to transfer the HI-STORM from the Unit 2 FHB to the ISFSI pad.

6) Ancillaries consisting of : 1) a forced helium dehydration (FHD) system including a chiller and various pumps, valves, pressure indicators and hoses mounted on a skid to facilitate preparing the MPC for storage operations, and 2) a mass spectrometer helium leak detection system for field leak testing the MPC vent and drain port cover plates.

7) The ISFSI concrete storage pad on which the loaded overpacks (HI-STORMs from Units 1, 2 and 3) are placed for long term storage operations.

8) Impact limiters in the Cask Loading Pool to reduce the stresses on the fuel for a postulated HI-TRAC drop.

9) Damaged Fuel Cans (DFC) which will be loaded into the MPCs. All Unit 1 fuel will be loaded into these Damaged Fuel Cans.

Existing major Unit 1 plant equipment used for the dry fuel storage includes the non-single failure proof fuel handling crane in the FHB and the heavy haul transport route between the Unit 1 FHB and the Unit 2 Fuel Storage Building (FSB).

The Unit 2 single-failure proof Gantry Crane and the heavy haul transport route between the Unit 2 FSB and the ISFSI Pad are addressed in Appendix F for Units 2 and 3.

### E.1.3 Unit 1 Cask Loading Sequence

The general sequence of events for moving the Unit 1 spent fuel from the West Spent Fuel Pool to the ISFSI begins with typical preparation of the components for loading activities, including inspection, cleaning, and fit-up. The general loading activities are described below. This list is intended to provide a basic understanding of the cask loading sequence, but is not a complete list of required steps or the exact sequence of steps. Detailed procedures are used to implement cask loading activities. The procedures address the required steps and the sequence of steps, and include appropriate limits and precautions.

1. The empty MPC is pre-loaded with DFC's into the HI-TRAC transfer cask prior to movement into the Unit 1 FHB. The HI-TRAC is moved into the FHB using the VCT and an air pad load transporter. Initial MPC preparation activities are completed in the dry environment of the 71'-6" elevation FHB entry area. Prior to moving the MPC to the cask loading pool, it is filled with water.

2. The HI-TRAC/MPC assemblage is lifted vertically from the 70'-6" FHB entry area using the Fuel Handling Crane and lift yoke and placed on the 30' elevation floor of the Cask Loading Pool onto the impact limiter.

3. Using the Fuel Handling Crane, 32 Unit 1 fuel assemblies meeting the limits in the HI-STORM 100 CoC are transferred from the West Fuel Pool, through the disassembly pool, into the Cask Loading Pool and then lowered into DFC's in the MPC.

4. The Fuel Handling Crane with the lift yoke installed on the Fuel Handling Crane hook is used to move the MPC lid, with its drain line installed, and lower the lid into place in the MPC.

5. After engaging the trunnions with the lift yoke, the HI-TRAC/MPC is then lifted vertically from the floor of the Cask Loading Pool to a point where the bottom of the HI-TRAC is just above the lip of the Cask Loading Pool at approximately 71' elevation.

6. The HI-TRAC is moved to the east to the work platform and is set down on the 70'-6" floor of the FHB entry area.

7. The cask is decontaminated if necessary

8. Preparations are made to support MPC lid welding, draining, drying, and backfilling.

9. A small amount of water is drained from the MPC to create a space between the water and the underside of the MPC lid. A purge source is connected to the MPC vent port connection and left in operation to provide for combustible gas control and moisture removal during welding operations.

10. The MPC lid is welded into place, and dye penetrant non destructive examinations (NDE) and MPC pressure testing are performed in accordance with the HI-STORM 100 System CoC and CFSAR and site Dry Fuel Storage procedures.

11. The MPC is drained, dried, pressure tested and helium filled in accordance with the HI-STORM 100 CoC and CFSAR, and site procedures.

12. The vent and drain port covers are welded in place and helium leak tested. The MPC closure ring is welded into place and a NDE on these welds is performed.

13. The MPC lift cleats and the HI-TRAC top lid are installed. The HI-TRAC is surveyed by Radiological Control Technicians to establish the level of loose surface contamination and dose rate, if any, and the appropriate control procedures to be implemented if any. The Fuel Handling Crane, with the lift yoke installed, is used to lift the HI-TRAC/MPC straight up until the bottom of the HI-TRAC is at approximately 71' elevation to insert the air pad load transporter under the HI-TRAC. The HI-TRAC is lowered and the Yoke disengaged.

14. The HI-TRAC/MPC is then moved out of the Fuel Handling Building on the Air Pad Casters. The Vertical Cask Transporter is moved into place, engages the HI-TRAC trunnions and transports the HI-TRAC/MPC to the entrance of the Unit 2 Fuel Building. The HI-TRAC/MPC is lowered onto the low profile transporter (LPT) and moved into the Unit 2 Fuel Storage Building.

15. Using the single-failure-proof Unit 2 Gantry crane, the HI-TRAC/MPC will be lifted and then lowered on the HI-STORM overpack and mating device which is staged on the LPT in the Unit 2 FSB.

16. The Gantry crane yoke device engages the MPC cleats, allowing the MPC to be lifted slightly.

17. Bolting that attaches the pool lid to the HI-TRAC is removed and the pool lid is lowered into the mating device drawer. The hydraulic slide on the mating device is actuated, allowing the pool lid of the HI-TRAC to be withdrawn, creating a transfer path for the MPC into the overpack.

18. The MPC is lowered into the overpack using the single-failure-proof Gantry Crane.

19. The HI-STORM/MPC, is moved out of the Unit 2 Fuel Storage Building on the Low Profile Transporter to a pre-designated location with sufficient overhead clearance to use the VCT. The HI-STORM overpack lid is placed and secured for long term storage operations.

20. The HI-STORM lift brackets are installed on the VCT. The VCT and connecting lift brackets are moved into position over the HI-STORM and the lift brackets are connected to the HI-STORM.

21. The HI-STORM is transported by the VCT on the designated transport route to the ISFSI pad and placed in its storage location. The VCT lift brackets are disconnected from the HI-STORM and moved away by the VCT.

### E.1.4 Unit 1 Cask Unloading Operations

There are no credible events related to on-site ISFSI operations that would require cask unloading, or that would damage the overpack, transfer cask, or an MPC such that the fuel cannot be recovered if cask unloading is necessary. After the 160 Unit 1 fuel assemblies are transferred to the ISFSI , Entergy intends to drain and decontaminate the Unit 1 fuel pool. If cask unloading of Unit 1 fuel is necessary after the draining of the Unit 1 pool, provisions can be made to re-flood the Unit 1 pool or to perform the unloading operation using the Unit 2 facilities.

Recovery of the loaded MPC up to the point of removing the MPC lid is the reverse of the loading sequence with certain additional considerations. The following additional considerations and steps would be implemented:

1. Once it is decided that an MPC needs to be unloaded, a thermal evaluation is performed to determine the temperature of the helium gas inside the MPC. In the case of Unit 1 fuel, all of which is over 30 years old and most of which has a low burn up history, heat removal is not considered a major concern.

2. The overpack is moved back to the Unit 2 Fuel Storage Building and the mating device and the HI-TRAC are installed. The MPC is transferred to the HI-TRAC using the Unit 2 single-failure-proof Gantry crane.

3. The vent and drain port covers are removed

4. Gas samples are taken to determine if there is any failed fuel in the MPC

5. Water can be reintroduced into the MPC up to a level appropriate to allow lid weld cutting to proceed. Upon completion of the lid weld cutting, the HI-TRAC/MPC is moved to the re-flooded Unit 1 cask load pool, or the Unit 2 cask load pit, and can be unloaded.

## E.1.5 Unit 1 Site Off-Normal Events

The events that could affect the plant due to fuel movements, fuel containment, on-site cask transport, or cask storage that are not addressed in the HI-STORM 100 system FSAR and require a site specific evaluation are discussed below:

**Cask Loading and Handling Operations**

The Indian Point Unit 1 Fuel Building Cask Handling Crane is not a single failure proof crane based on a comparison between the crane design features and the guidance of NUREG-0554. The Unit 1 technical specification 2.2.5 specifically states that:

"If a spent fuel pool contains spent fuel, the spent fuel cask shall not be moved over that pool or within a distance of that such that the cask could strike the pool if it fell or tipped."

In addition, Indian Point Unit 1, in correspondence to the AEC committed to further restrictions on the movement and load paths of spent fuel shipping casks in the Fuel Handling Building. (Reference Letter: W. Cahill, Jr. to AEC; "Responses to Directorate of Licensing Questions in Letter of March 6, 1974"; dated May 15, 1974.)

Subsequent to AEC review of the response, Indian Point Unit 1 loaded and shipped 264 fuel elements using the 30 ton IF-200 shipping cask. These casks were loaded using the existing 75 ton Fuel Handling Crane.

Based on a review of the crane's required heavy load handling operations in support of HI-STORM 100 system cask loading, certain load drops of heavy loads have been postulated and evaluated. Indian Point Unit 1 10 CFR 50 License Amendment Request for Amendment 53 (Reference: NL-07-033, February 22, 2007) requested NRC approval to use the Unit 1 Fuel Handling crane for dry fuel storage cask related operations involving heavy loads with spent fuel at Indian Point Unit 1. Included in that Amendment Request are the bases for postulating certain bounding load drops not previously evaluated by the NRC and the descriptions and evaluations of those drops. The NRC issued a Safety Evaluation Report and approved Amendment 53 on May 9, 2008 (Reference: Letter RA-08-072).

The load drops evaluated in the Amendment include the following:

1. An MPC lid drop on an MPC

2. An MPC lid drop on a HI-TRAC transfer cask flange

3. A loaded transfer cask drop straight down into the cask loading pool.

4. A loaded transfer cask drop on the corner of the cask bottom on the 30'elevation of the cask loading pool.

5. A loaded transfer cask drop on the 71' elevation edge of the cask loading pool with a subsequent tip over striking the West wall of the cask loading pool.

6. A vertical drop on the 70'-6" elevation of the floor of the Fuel Handling Building entry area.

**A MPC lid drop on a fuel filled MPC**

The MPC lid weighs approximately 10,000 lbs and meets the criterion for being a designated heavy load. This drop event involves a crane failure-induced drop of the MPC lid onto the open MPC after being loaded with spent fuel. The HI-Track/MPC is located on the impact limiter on the bottom of the cask loading pool for this event.

**A MPC lid drop on a HI-TRAC transfer cask flange**

This postulated drop of a MPC lid differs from the previous scenario in that the angle of intrusion of the edge of the lid into the Transfer Cask/MPC is more severe.

**A loaded transfer cask drop straight down into the cask loading pool.**

A crane failure-induced drop of a loaded Transfer Cask/MPC from elevation 70'-6" into the bottom of the cask loading pool at elevation 30'-0" is postulated. An impact limiter is installed at the 30' floor elevation of the cask loading pool. This drop examines the impact of the drop on the loaded fuel. The bottom of the cask loading pool is concrete founded on bed rock, and no safety related equipment or other spent fuel is located near or in the cask loading pools.

**A loaded transfer cask drop on the corner of the cask bottom on the 30' elevation of the cask handling pool floor.**

Similar to the previous scenario, this drop hypothesizes the cask lands on an edge of the cask rather than flat on the bottom. The mitigating effect of the installed impact limiter at the pool bottom is also credited in this event.

**A loaded transfer cask drop on the 71' elevation edge of the cask handling pool.**

This postulated drop configuration pivots the falling transfer cask onto the top of the west wall of the cask loading pool which separates the cask loading pool from the cask disassembly pool. The impact of the cask onto the wall is similar in configuration to the horizontal drop analysis presented in the Holtec HI-STORM FSAR but may be more severe due to the impact target properties and shape.

**A vertical drop on the 70'-6" elevation of the floor of the Fuel Handling Building entry area.**

This scenario is analyzed to determine the allowable carrying height of the loaded transfer cask above the 70'-6" Fuel Handling Building entry area. After the Transfer cask is lifted clear of the cask loading pool, it must traverse about 20 feet to the east to the welding area and must again be lifted a number of inches onto the air pad transporter or low profile transporter prior to movement to Unit 2.

During the horizontal traverse, the cask travels a number of inches above the 70'-6" concrete floor which is layered with several inches of leveling grout. The floor is founded on bedrock or engineered fill. Since there is no safety related equipment anywhere within the Fuel Handling Building, the acceptance criteria for a drop is the deceleration g loads on the contained spent fuel

**Conclusion:**

The acceptance criteria for g loads on the fuel are discussed in NUREG -1864. IPEC Unit 1 meets these criteria for the drop events analyzed as discussed in the NRC Safety Evaluation Report for Amendment 53.

## Sealing and Cleanup

### High Radiological Dose During MPC Preparation

Draining of the MPC involves the connection of hoses to the remote valve operating assemblies that are connected to the vent and drain ports in the top of the MPC lid. This evolution is performed after the HI-TRAC/MPC is placed on the 70'-6" floor of the FHB entry. Prior to welding the lid, a small amount of water is pumped from the MPC to ensure the water level is below the lid weld area. NDE is performed on the lid root weld and periodically thereafter until the weld is completed, as required by the HI-STORM 100 CoC and CFSAR, and site procedures. The MPC is pressure tested, the remaining water is removed from the MPC and the Forced Helium Dehydration (FHD) system is connected to the MPC. The MPC fuel cavity is dried until the acceptance criteria in the HI-STORM 100 CoC are met. The MPC is the backfilled with helium to the CoC- required pressure. Final sealing is accomplished by welding the vent and drain port covers plates to the lid and closure ring to the MPC lid and shell.

The primary concerns associated with these activities are the dose rates to the individuals, working around the casks, discharge of water from the cask, the release of gases from the cask, and the control of any residual cask surface contamination from the cask loading pool. MPC preparation activities are performed under the same administrative control procedures used at the Indian Point Plants for other work in a radiation controlled area to comply with 10 CFR 20. Radiation surveys will be taken as appropriate and Radiation Work Permits will identify the appropriate dose and dose rate limits, dosimetry, and protective clothing for all the activities.

Because of the height of the HI-TRAC/MPC transfer cask, a movable elevated work platform is used to access the top of the MPC. This platform design incorporates access features which also reflect the goal to maintain personnel exposure ALARA. The design also is reflective of IPEC industrial safety standards and OSHA regulations.

### Hydrogen Gas Ignition During MPC Lid Welding and Cutting

Hydrogen gas production that may occur due to oxidation of the aluminum in the neutron absorber in the MPC fuel basket, or other phenomenon, will start with the introduction of spent fuel pit water and fuel into the MPC interior. Upon MPC lid installation, any gas generated potentially could be trapped under the lid. The next evolution in the loading sequence drains water from the cask interior via the drain line through the MPC lid. Draining water creates a gas space below the weld area underneath the lid to prevent quenching of the weld. Actions are taken to prevent the concentration of hydrogen potentially reaching flammability limits in the gas space. In accordance with the HI-STORM FSAR, prior to and during lid welding operations, the gas space under the lid is purged with an inert gas.

Sampling of the MPC lid-to-shell weld gap is performed and the exhaust is monitored for combustible gas during welding operations. Purging and monitoring of the exhaust gas is continued until welding is complete to prevent possible ignition of combustible gasses from the heat of welding of passes subsequent to the root pass.

If the MPC is required to be unloaded, the fuel cavity will be flooded with water. The gas space under the MPC will be purged or exhausted prior to and during weld cutting operations in accordance with the HI-STORM FSAR. This will prevent any potential hydrogen ignition caused by the weld cutting operation. Also refer to the discussion in Section F.4.3.5 of this report.

**Transfer of the MPC**

Potential events involving the drop of the HI-TRAC transfer cask or a loaded MPC inside the FHB have been discussed previously in this section. Since the VCT is designed in accordance with ANSI N14.6 and has redundant drop protection features as required by the HI-STORM CoC, Appendix A, Section 5.5.a.3, a drop of the HI-STORM (or the HI-TRAC) is not considered a credible event except for the brief period during lifting and lowering when the redundant locking pins are not installed. In accordance with the provisions of HI-STORM CoC, Appendix A, Technical Specification Section 5.5.a.2, a site specific analysis has been performed demonstrating a drop of the HI-TRAC/HI-STORM will not induce g loads in excess of 45 g's as specified in the Technical Specification (Reference: Holtec Calculation; "Postulated Mechanical Drop Accidents at IP-1"; Report No. HI-2073755; dated March 31, 2008.) During the lift of the HI-TRAC prior to engaging the redundant locking features, an impact limited will be inserted under the HI-TRAC. After the MPC and HI-TRAC have been moved to the Unit 2 FSB by the VCT and the MPC has been transferred to the HI-STORM overpack in the Unit 2 Fuel Storage Building, using a combination of the LPT and the VCT , the HI-STORM will be transported to the ISFSI. The movement of the Unit 1 HI-STORM will be performed essentially the same as the transfer of the Unit 2 HI-STORMs from the Unit 2 Fuel Storage Building and will use the same LPT and VCT. Since Technical Specification 5.5.a limits the lift height for the Unit 1 HI-STORM to 8" (versus 11" for Unit 2), an impact limiter will be used at the ISFSI when the redundant locking pins are disengaged on the VCT.

**Storage of the Hi-STORM**

**Tornado, Flooding, and Earthquake**

The five Unit 1 HI-STORM 100S-185's will be placed on the Indian Point ISFSI which was designed and built for the storage of spent fuel from Indian Point Units 1, 2, and 3. All three Units will use the HI-STORM 100 systems with only slight variations in the Unit 1 design, primarily weight and height reductions from the Unit 2 design. The Unit 1 variations are discussed in detail in Amendment 4 to the HI-STORM CoC.

The Unit 1 design meets the Tornado, Flooding, and Earthquake design criteria delineated in the Unit 2 FSAR and as discussed in Appendix F, the Unit 2 10 CFR 72.212 (b) report. See the discussion in Section E.4.2 for the detailed discussion of site parameters.

## APPENDIX E.2A
## LIST OF SITE SPECIFIC AND UNIT 1 PROCEDURES

In addition to the corporate procedures listed in Appendix A, and IPEC Site and ISFSI procedures, IP-1 requires the following procedures in order to address the specific fuel transport and storage activities at Indian Point Unit 1. The physical arrangement of the Unit 1 FHB, the use of the existing non-single failure proof crane, the use of an IP-1 specific HI-STORM 100S-185 system, and the loading of all fuel into damaged fuel cans, dictates that Unit 2 procedures must be reviewed and revised as appropriate for use by Unit 1. Unit 2 procedures are listed below for completeness

1. MPC-32 Receipt, Handling, and Fit Up (2-DCS-001-GEN)
2. HI-TRAC100D-1P1 Receipt , Handling, and Initial Assembly (2-DCS-002-GEN)
3. HI-STORM100S-185 Receipt, Handling, and Fit Up ( 2-DCS-003-GEN)
4. RIGID Chain Drive System Operations (2-DCS-004-GEN)
5. Ancillary Pre-operational inspection and Functional Tests (2-DCS-005-GEN)
6. Vertical Cask Transporter Operation (2-DCS-006-GEN)
7. ISFSI Storm Water Pollution Prevention Inspection and Maintenance (2-DCS-007-GEN)
8. Unit 2 MPC Load and Seal (2-DCS-008-GEN)
9. MPC Transfer and HI-STORM Movement (2-DCS-009-GEN)
10. Ancillary Lay up Procedure (2-DCS-010-GEN)
11. HI-STORM Inspection ( 0-DCS-011-GEN)
12. Unit 2 MPC Unloading Procedure (2-DCS-012-GEN)

13. Transfer Table Operations (2-DCS-013-GEN)

14. Unit 1 Fuel Handling (1-DCS-014-GEN)

15. (0-DCS-015-GEN Procedure number is available for future use)

16. DCSS Special Lifting Devices Inspection (2-DCS-016-GEN)

17. (2-DCS-017-GEN Procedure number is available for future use)

18. (2-DCS-018-GEN Procedure number is available for future use)

19. (2-DCS-019-GEN Procedure number is available for future use)

20. (2-DCS-020-GEN Procedure number is available for future use)

21. (1-DCS-021-GEN Procedure number is available for future use)

22. (2-DCS-022-GEN Procedure number is available for future use)

23. FHD Operations (0-DCS-023-GEN)

24. (1-DCS-024-GEN Procedure number is available for future use)

25. Air Pad Operations (1-DCS-025-GEN)

26. Unit 2 Crane Operations (2-DCS-026-GEN)

27. FSB 110 Ton X-SAM Gantry Crane Preventative Maintenance (2-DCS-027-GEN)

28. Unit 1 MPC Load and Seal (1-DCS-028-GEN)

29. (2-DCS-029-GEN Procedure number is available for future use)

30. Fuel Selection for Dry Cask Storage-Unit 1(1-DCS-030-GEN)

31. Fuel Selection for Dry Cask Storage-Unit 2 (2-DCS-031-GEN)

32. Dry Cask Loading Readiness Guidelines (2-DCS-032-GEN )

33. Abnormal Operations (2-DCS-033-GEN)

34. HI-TRAC Annual Inspection (2-DCS-034-GEN)

35. Unit 1 MPC Unloading Procedure (1-DCS-035-GEN)

36. Radiological Controls for Dry Cask Storage (0-RP-RWP-420)

## APPENDIX E.2.B

## IPEC UNIT 1 KEY HOLTEC CASK SYSTEM DRAWINGS

| Drawing Series | Component |
|---|---|
| 4827 sheets 1-11 | HI-STORM 100S Assembly |
| 4727 sheets 1-12 | HI-TRAC 100D-IP1 Assembly |
| 4706 sheets 1-5 | MPC-32 Enclosure Vessel |
| 4706 sheets 7-8 | MPC-32 Lid Details |
| 4705 sheets 1-7 | MPC-32 Basket Assembly |
| 5097 sheets 1-6 | HI-TRAC Lifting Yoke |
| 4968 sheets1-4 | Impact Limiter |
| 4706 sheet 6 | MPC Lift Cleat |
| 4797 sheets 1-4 | Damaged Fuel Cans |
| 4706 sheet 8 | Closure Ring |
| 5187 sheets 1-2 | Unit 1 HI-TRAC Lift Links |
| | |

## SECTION E.3
## COMPLIANCE WITH 10 CFR PART 72

General compliance with the requirements specified in 10 CFR 72 is discussed in Section V of the main body of this report. Certain regulatory requirements requiring site-specific discussion for the IPEC site and the Unit 1 facility are provided below.

### E.3.1 §72.106-Controlled Area of the ISFSI

*(a) For each ISFSI or MRS site, a controlled area must be established.*

As defined by 10 CFR 72, the controlled area means that area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed. The IPEC ISFSI is located within the Unit 2 protected area at a location approximately 133 (448 ft.) meters north of the center of the Unit 2 Containment. The Hudson River shore is located approximately 168 meters (554 ft) west of the center of the ISFSI pad.

The site exclusion area boundary at the northern end of the combined IPEC Units 1, 2 and 3 site is defined as a 520 meter (1716 ft.) radius from the center of the Unit 2 Containment. That portion of the Hudson River to the east of the ISFSI but within the exclusion boundary is considered part of the controlled area should a radiological emergency necessitate any action. (Reference 10 CFR 100.3).

To the north, east, and south of the ISFSI the controlled area extends a minimum of 454 meters (1500 ft.).

The ISFSI is entirely within the boundary of the plant controlled area.

*(b)Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lenses of the eye) of 50 rem. The lens dose equivalent may not exceed 15 rem and the shallow dose equivalent to the skin or any extremity may not exceed 50 rem. The minimum distance from the spent fuel, high level radioactive waste, or reactor-related GTCC waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.*

Design basis accidents that may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. Because the damage is localized and the vast majority of the cask shielding material remains intact, the site boundary dose rate for the loaded HI-STORM overpack for accident conditions are the same as the normal condition dose rates. The dose versus distance from the HI-STORMS containing the MPC 32 is documented in Holtec Report HI-2073724, dated 7/30/2007. Therefore the accident dose limits of 10 CFR 72.106 are met. As discussed in HI-STORM FSAR Chapter 7, there is no credible leakage from the confinement boundary during accident conditions, based on Interim Staff Guidance-18 (ISG-18) for the MPC lid-to-shell weld and because the vent and drain port covers are field leak tested to a "leaktight" acceptance criterion in accordance with ANSI N14.5. Therefore there is no effluent dose contribution to the calculated normal, off-normal, or accident off site accident dose.

*(c ) The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety.*

That portion of the IPEC controlled area that extends beyond the shoreline of the Hudson River to the west of the IPEC facility is controlled as required. Agreements are in place with the cognizant Federal, State, and Local governments. Emergency and Security facilities, personnel, and equipment are in place in accordance with the existing NRC approved IPEC Emergency Planning and Security Programs.

**E.3.2 § 72.122- Overall Requirements**

*72.122(a) Quality Standards Structures, systems, and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.*

Each component or sub-component of the ISFSI and the HI-STORM 100 system is classified as Important-to-Safety (ITS), Category A, B, or C, or Not-Important-To-Safety (NITS) in accordance with the guidance in NUREG/CR-6407. These classifications are made based on the design function of the component or subcomponent.

Activities involving ITS of QAPA components or subcomponents are conducted in accordance with Holtec International's 10 CFR 72, Subpart G quality assurance program or the Entergy 10 CFR 50 Appendix B quality assurance program and documented on attachment 9.2 of engineering procedure ENN-DC-167.

*72.122(b) Protection against environmental conditions and natural phenomena.*
*(1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI or MRS and to withstand postulated accidents.*

*(2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning,*

*hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect:*

*(A)Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and*

*(B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena.*

*(ii) The ISFSI or MRS also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel, high-level radioactive waste, or reactor-related GTCC waste or onto structures, systems, and components important to safety.*

*(3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.*

*(4) If the ISFSI or MRS is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.*

The cask system being deployed at IPEC Unit 1 under the general license provisions of 10 CFR 72 is the HI-STORM 100 system, which has been certified by the NRC and is listed in 10 CFR 72.214. The cask system has been designed and analyzed to withstand environmental conditions and natural phenomena as described in the HI-STORM 100 System FSAR. The generic design criteria for the environmental conditions and natural phenomena used in the cask design were verified to be bounding for the site-specific design basis environmental phenomena applicable to the IPEC site as identified in the Unit 2 UFSAR, Chapter 2. The balance of the ISFSI design has appropriately considered environmental conditions and natural phenomena as they apply to the particular structure, system, or component of the ISFSI. The details of these design considerations may be found in the applicable design control documentation of the ISFSI design package, FCX-00550-01, April 20, 2006.

*72.122(c ): <u>Protection against Fires and Explosions.</u> Structure, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion conditions. Noncombustible and heat-resistant materials must be used wherever practicable throughout the ISFSI or MRS, particularly in locations vital to the control of radioactive materials and to the maintenance of safety functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects* *of fire and explosions on structures, systems, and components important to safety. The design of the ISFSI or MRS must include provisions to protect against adverse effects that might result from either the operation or failure of the fire suppression system.*

The HI-STORM 100 System has been designed for fires, explosive overpressure, and missiles. The IPEC ISFSI, including the transport route from the Unit 1 FHB to the Unit 2 FSB and then to the ISFSI has been evaluated to ensure that the generic design basis for the cask bounds the actual configuration at the IPEC site with respect to fire and explosion hazards (See Section E.4.3.2.5 for details). The IPEC fire protection plan and fire hazard analysis have been reviewed to ensure they address dry spent fuel storage operations at the site. A fire suppression system is not used at the IPEC ISFSI because there are no combustible materials stored at the ISFSI. Fires due to transient combustibles will be extinguished with available portable fire suppression equipment.

*72.122(d):<u>Sharing of structures, systems, and components.</u> Structures, systems, and components important to safety must not be shared between an ISFSI or MRS and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions, including the ability to return to a safe condition in the event of an accident.*

The HI-STORM 100 system does not require electric power to perform its design functions. The cask system is a welded, canister-based system, passively cooled by a naturally ventilated overpack. There are no cask leakage monitoring systems. The inlet and outlet air ducts are visually inspected for blockage on a periodic basis and monitored with a remote temperature monitoring system. The ISFSI, including the cask system shares no structures, systems, or components important to safety with any other facility.

*72.122(e): <u>Proximity of sites:</u> An ISFSI or MRS located near other nuclear facilities must be designed and operated to ensure that the cumulative effect of their combined operations will not constitute an unreasonable risk to the health and safety of the public.*

The IPEC ISFSI is co-located within the protected area of the IPEC site which includes Units 1, 2, and 3. Fuel from all three Units will eventually be stored at the common ISFSI. The existing plan is to store the fuel from all three Units in nearly identical Holtec International HI-STORM 100 cask systems. The additional direct radiation dose to the public from ISFSI operations is negligible due to the distance between the ISFSI and the site boundary and the partial shielding provided by exiting plant structures and topography during on site cask transport. The HI-STORM 100 System is designed not to release any radioactive effluents under normal, off-normal, or accident conditions, so there is no additional effluent dose to the public from ISFSI operations. The IPEC Site Environmental Radiological Monitoring Program has been revised as appropriate to monitor the cumulative site boundary dose to identify any additional dose contribution from the ISFSI if at all detectable.

*72.122(f): <u>Testing and maintenance of systems and components.</u> Systems and components that are important to safety must be designed to permit inspection, maintenance, and testing.*

No periodic maintenance of major ITS components other than minor touch up painting on the cask is required at the ISFSI. The HI-STORM 100 system is completely passive in design, so there are no mechanical or electrical systems to maintain on the storage overpack or canister. Surveillance during storage operations is limited to visual observations of the cask inlet and outlet ducts, or if installed, monitoring by means of temperature detectors. Periodic visual inspections for deterioration of the vent screens and cask surfaces are the only inspections required. The Unit 1 fuel loading campaign, which only includes the existing 160 spent fuel assemblies in the West Fuel Pool, is a one time short duration project.

*72.122(g): <u>Emergency capability.</u> Structures, systems, and components important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire, and police departments, ambulance service, and other emergency agencies.*

The operation of the ISFSI has been evaluated for its effects on the IPEC emergency response plan. The ISFSI is located within the plant protected area and access is available through locked gates. The same onsite and offsite emergency facilities as those used for the Part 50 facility are used for events associated with ISFSI operations.

*72.122(h): <u>Confinement barriers and systems:</u> (1) The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.*

*(2)For underwater storage of spent fuel, high level radioactive waste, of other reactor-related GTCC waste in which the pool water serves as a shield and a confinement medium for radioactive materials, systems for maintaining water purity and pool water level must be designed so that any abnormal operations or failure in those systems from any cause will not cause the water level to fall below safe limits. The design must preclude installations of drains, permanently connected systems, and other features that could , by abnormal operations or failure, cause a significant loss of water. Pool water level equipment must be provided to alarm in a continuously manned location if the water level in the storage pools falls below a predetermined level.*

*(3) Ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne particulate materials during normal of off-normal conditions.*

*(4) Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.*

*(5) The high-level radioactive waste and reactor-related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.*

The HI-STORM 100 system is a dry- type storage system with a redundant welded confinement barrier. The canister fuel cavity is backfilled with helium gas to promote effective decay heat removal and inhibit corrosion of the fuel cladding. The cask design has been certified with certain limits on the characteristics of the stored fuel to ensure adequate heat removal and protection of the cladding.

The design of the IPEC Unit 1 fuel assemblies includes a "wrapper" or "shroud can" which encloses the fuel rod assembly. This can precluded a detailed visual inspection of the enclosed fuel rods to determine if any of the rods meet the definition of damaged fuel as discussed in the Spent Fuel Project Office Interim Staff Guidance Document-1 Revision 1 (ISG-1). The Holtec International HI-STORM100S-185-1 design variant (Amendment 4 of the CoC) includes the capability of loading all 32 fuel bundles contained in the canister cavity in damaged fuel cans. Since IPEC Unit 1 cannot definitively identify whether any particular assembly does or does not contain damaged rods in a cost effective manner, an economic decision was made to store all 160 assemblies in damaged fuel cans.

No monitoring of the canister confinement system is required because it is a welded system. Monitoring of the overpack is limited to periodic visual inspection of the air inlet and outlet ducts to ensure they are free of blockage and the overpack is able to transfer an adequate amount of heat from the MPC to the environs. Handling and retrievability is ensured in the cask system design, which includes a transfer cask with lead and water radiations shields to protect personnel and keep occupational exposures due to loading operations well below the limits in 10 CFR Part 20.

*72.122(i): <u>Instrumentation and Control Systems.</u> Instrumentation and control systems for wet spent fuel storage and reactor-related GTCC waste storage must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that* *must remain operational under accident conditions must be identified in the Safety Analysis Report. Instrumentation systems for dry storage casks must be provided in accordance with cask design requirements to monitor conditions that are important to safety over anticipated ranges for normal conditions and off-normal conditions. Systems that are required under accident conditions must be identified in the Safety Analysis Report.*

The HI-STORM 100 System at IPEC includes a temperature monitoring system which monitors two of the inlet and two of the outlet air temperatures on each overpack on the ISFSI pad. The existence of a temperature differential confirms that continued air flow is in accordance with the design. The temperature monitoring resistance temperature detectors are classified as Important To Safety, Class C (ITS-C). The monitoring system has no accident mitigation function and, as indicated in the Technical Specification, monitoring requirements can also be satisfied by daily visual inspections.

*72.122(j):<u>Control Room or Control Area.</u> A control room or control area, if appropriate for the ISFSI or MRS design, must be designed to permit occupancy and actions to be taken to monitor the ISFSI or MRS safely under normal conditions, and to provide safe control of the ISFSI or MRS under off-normal or accident conditions.*

The ISFSI is co-located in the protected area of the IPEC Unit 2 Part 50 facility. The main control room of the power plant provides for the occupancy and actions to be taken in the event of an off-normal or accident condition at the ISFSI. No separate ISFSI control area is provided nor required.

*72.122(k): <u>Utility or other Services.</u> (1) Each utility serve system must be designed to meet emergency conditions. The design of utility services and distribution systems that are important to safety must include redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform safety functions assuming a single failure.*

*(2) Emergency utility services must be designed to permit testing of the functional operability and capacity, including the full operational sequence, of each system for transfer between normal and emergency supply sources; and to permit the operation of associated safety systems.*

*(3) Provisions must be made so that, in the event of a loss of the primary electric power source or circuit, reliable and timely emergency power will be provided to instruments, utility service systems, the central security alarm station, and operating systems, in amounts sufficient to allow safe storage conditions to be maintained and to permit continued functioning of all systems essential to safety.*

*(4)An ISFSI or MRS which is located on the site of another facility may share common utilities and services with such a facility and be physically connected with the other facility; however, the sharing of utilities and services or the physical connection must not significantly:*

(i) *Increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety; or*

(ii) *Reduce the margin of safety as defined in the basis for any technical specification of either facility.*

The HI-STORM 100 System does not require electric power or any other utilities to perform its design functions. The cask system is a welded, canister-based system, passively cooled by a naturally ventilated overpack. There is no cask leakage monitoring system. The inlet and outlet air ducts are visually inspected for blockage on a periodic basis. The ISFSI, including the cask system share no structures, systems, or components important to safety with any other facility. The ISFSI is within the boundaries of the IPEC Unit 2 protected area. The details of the structures, systems, and components associated with the IPEC Security Program and the integration of the ISFSI into the protected area envelope are discussed in the IPEC Security Program Plan. A 10 CFR 50.59 evaluation has been performed to address the ISFSI and Dry Cask Storage Operations and has concluded the requirements of 72.122(k)(4) are satisfied.

*72.122(l); Retrievability Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing and disposal.*

The HI-STORM 100 System is designed for retrievability of the spent fuel as discussed in the FSAR for the cask system.

### E.3.3 § 72.124-Criteria for Nuclear Criticality Safety

*72.124(a): Design for criticality safety: Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in the calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions,*

The HI-STORM 100S-185 System has been analyzed for the most reactive credible conditions during spent fuel loading in the cask pool, assuming fresh fuel and unborated water and found to be safely subcritical. The criticality design of the cask is described in Chapter 6 of the HI-STORM 100 System FSAR and has been reviewed and approved by the NRC.

The IPEC Unit 1 fuel pool is normally unborated and has no Technical Specification limit for boron concentration. The HI-STORM 100S-185 System, which is a unique IPEC Unit 1 variant of the Holtec System, assumes no boron concentration in the pool water and assumes no burnup credit as well. This cask and its margin of subcriticality is discussed in Amendment 4 of the Holtec CoC.

*72.124(b): <u>Methods of Criticality Control:</u> When practicable, the design of an ISFSI or MRS must be based on a favorable geometry, permanently fixed neutron absorbing materials (poisons) or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing material cannot occur over the life of the facility.*

The HI-STORM 100 System incorporates the favorable geometry of the MPC-32 fuel basket and fixed neutron absorber material for criticality control as discussed in Chapter 6 of the HI-STORM 100 System FSAR. There are no known degradation mechanisms for the fixed neutron absorbers in a helium environment over the life of the ISFSI. Therefore, positive means for verifying continued neutron absorber efficacy are not required.

*72.124(c ): <u>Criticality Monitoring:</u> A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.*

During the time period when the special nuclear material is neither beneath water shielding nor packaged in its stored configuration (i.e. from the time the cask is removed from the cask loading pool and drained, dried, and backfilled with helium), criticality monitoring is provided by two dose rate detectors capable of providing alarm functions in the event of a criticality incident. Radiological Monitoring is addressed in site procedure number 0-RP-RWP-420, Radiological Controls for Dry Cask Storage.

## E.3.4 § 72.126-Criteria for Radiological Protection

*72.126(a): <u>Exposure Control</u> Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to:*

*(1) Prevent the accumulation of radioactive material in those systems requiring access.*

*(2) Decontaminate those systems to which access is required.*

*(3) Control access to areas of potential contamination or high radiation within the ISFSI or MRS.*

*(4) Measure and control contamination of areas requiring access:*

*(5) Minimize the time required to perform work in the vicinity of radioactive components; for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement; and*

*(6) Shield personnel from radiation exposure.*

The HI-STORM system was designed to minimize exposure during the loading, unloading, and onsite transport and storage of the cask. Design aspects such as the use of a thick steel MPC lid for canister closure; the use of a shielded transfer cask for movement of the fuel-bearing canister; and the thick concrete overpack for storage are a few of the design methods employed.

Activities pertinent to the HI-TRAC and HI-STORM operation including survey and decontamination are controlled by site procedures such as 0-RP-RWP-420, as well as the cask loading/unloading specific procedures 1-DCS-028-GEN/1-DCS-035-GEN . The IPEC radiation protection program and application of these procedures also supports IPEC commitment to compliance with the requirements of 10 CFR 20 for occupational dose considerations. See Section E.2.A for a detailed list of the site specific procedures.

*72.126(b): <u>Radiological Alarm Systems.</u> Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.*

The HI-STORM 100 System emits no radioactive effluents once the MPC is prepared for storage. Existing alarm systems for radiological monitoring of fuel loading operations in the Fuel Handling Building have been judged sufficient to warn personnel of inappropriate airborne or direct radiation.

*72.126(c ): Effluent and direct radiation monitoring.*

*(1)As appropriate for the handling and storage system, effluent systems must be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided.*

*(2)Areas containing radioactive materials must be provided with systems for measuring direct radiation levels in and around these areas.*

The MPCs have seal-welded closures, so no gaseous radioactive material leak path to the environment is available and no routine monitoring of effluents from the HI_STORM casks is required.

The IPEC Radiological Environmental Monitoring Program (REMP) includes the ISFSI in the scope of the ongoing monitoring program for the other facilities on the IPEC site.

*72.126(d):Effluent Control. The ISFSI or MRS must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in §72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.*

The MPCs have redundant seal-welded closures, so no gaseous radioactive material leak path to the environment is available and no routine monitoring of effluents from the HI_STORM casks is required. The MPC design has been shown by analyses to maintain the confinement boundary integrity under all normal, off-normal, and accident conditions of service as discussed in Chapter 3 of the HI-STORM 100 System FSAR. The MPC lid-to-shell closure weld design meets the guidance in ISG-18 and the MPC vent and drain port cover plates are leak tested to a "leak tight" acceptance criterion as defined in ANSI N 14.5 and as discussed in Chapter 7 of the HI-STORM 100 System FSAR. Based on these two factors, leakage from the MPC confinement boundary is considered non-credible and no effluent controls or dose analysis is required.

### E.3.5 §72.44, § 72.144, § 72.190, §72.194-Training and Operator Requirements

*§ 72.44(b)(4): The licensee shall have an NRC-approved program in effect that covers the training and certification of personnel that meets the requirements of subpart I before the licensee may receive spent fuel and/or reactor-related GTCC waste for storage at an ISFSI or the receipt of spent fuel, high level radioactive waste, and/or reactor-related GTCC waste for storage at an MRS.*

*§72.44(b)(5): The licensee shall permit the operation of the equipment and controls that are important to safety of the ISFSI or the MRS only by personnel whom the licensee has certified as being adequately trained to perform such operations, or by uncertified personnel who are under the direct visual supervision of a certified individual.*

*§72.144(d): The licensee, applicant for a license, certificate holder, and applicant for a CoC shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained.*

*§72.190: Operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important to safety must also be certified in such operations.*

*§72.194: The physical condition and general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment of motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defect.*

The Holtec-designed HI-STORM 100 System training requirements are found in the HI-STORM FSAR and CoC. The Training program at IPEC has been reviewed and verified to meet these requirements and the requirements of 10 CFR 72. The Mechanical Training Program, ENTQ-119 (IP-SMM-MA-119), provides the overview process and protocols for the Dry Cask Storage training effort.

HI-STORM FSAR Section 8.0 requires training procedures in place to account for operation of the ISFSI. The IPEC Dry Fuel Storage Training Program directs the training, qualification and continuing training of DFS personnel.

HI-STORM FSAR Section 12.2.1 requires training modules developed or modified to require a comprehensive, site-specific training, assessment and qualification program for the operation and maintenance for the HI-STORM 100 System and ISFSI. The IPEC Training Program contains all the course curriculum and requirements for each training module of the DFS Training Program. This includes training and qualification requirements.

HI-STORM FSAR Section 12.2.2 and CoC Condition 10 require dry run training exercises of the loading, closure, handling, and transfer of the HI-STORM 100 System components to be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The following work plans were developed to ensure compliance with these requirements:

- MPC Welding Operations
- Transfer/Transport Operations
- MPC Loading Operations
- MPC Fluid Operations

10 CFR 72.44(b)(4) requires a training program in effect that covers the training and certification of personnel that operate or supervise the operation of equipment and controls that have been identified as important to safety in the SAR and License. The IPEC Dry Fuel Storage Training Program ensures compliance with these requirements.

10 CFR 72.44(b)(5) and 10 CFR 72.190 require that only trained and certified personnel operate equipment and controls that have been identified as important to safety in the SAR and license. The IPEC Dry Fuel Storage Training Program ensures compliance with these requirements.

10 CFR 72.144(d) requires the licensee, applicant for a license, certificate holder, and applicant for a CoC to provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained. Compliance and implementation of the following ensures satisfaction of these requirements:

- The IPEC Dry Fuel Storage Training Program
- ASNT-SNT-TC-1A
- ANSI-N45.2.6
- QAPM
- ENN-NDE 2.10, Certification of NDE Personnel
- ENN-NDE 2.12, Certification of Visual Testing Personnel
- EN-QV-111, Training and Certification of Inspection/Verification and Examination Personnel

10 CFR 72.194 requires the physical condition and general health of personnel certified for the operation of equipment and controls that are important to safety must not be such

as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment of motor coordination must be considered in the selection of personnel for activities that are important to safety. IPEC procedures ensure compliance with these requirements.

### E.3.6 §72.212(a)(3)-License Extension

*The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance terminates 20 years after the date that the particular cask is first used by the general licensee to store spent fuel, unless the cask's Certificate of Compliance is renewed, in which case the general license terminates 20 years after the cask's Certificate of Compliance renewal date. In the event that a cask vendor does not apply for a cask model re-approval under § 72.240, any cask user or user's representative may apply for a cask design re-approval. If a Certificate of Compliance expires, casks of that design must be removed from service after a storage period not to exceed 20 years.*

The beginning of the twenty year IPEC 10 CFR 72 general license is when the first bundle is loaded into a HI-STORM cask. IPEC Unit 2 fuel was first loaded in November 2007. The Certificate of Compliance for the HI-STORM 100 System expires June 1, 2020. On or about 2015 IPEC plans to negotiate the status of a renewal application for the HI-STORM 100 System with the Certificate Holder to determine the best course of action.

### E.3.7 §72.212(b)(2)(i)(A)-Review of the CoC

*72.212(b)(2)(i)(A): Perform written evaluations, prior to use, that establish that conditions set forth in the Certificate of Compliance have been met.*

Section VI of the main body of this report addresses the CoC conditions that have generic responses applicable to all of the Entergy plants currently using the HI-STORM 100 System. See Section E.4 for the detailed IPEC Unit 1 specific discussion of the CoC, including applicable Amendments 1, 2, and 4.

### E.3.8 §72.212(b)(2)(i)(B)-ISFSI Design

*72.212(b)(2)(i)(B): Perform written evaluations, prior to use, that establish that cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes,* *through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion.*

## STORAGE PAD

The IPEC ISFSI design and construction details are described in the IPEC Unit 2 10 CFR 72.212 report which is identified as Site Specific Appendix F. Refer to Section F.3.8 for design information.

<u>Conclusion</u>

The results of the analyses are that all HI-STORM CoC and FSAR analysis requirements are met for (normal) storage, seismic, tip- over, and sliding considerations.

In accordance with Technical Specification 5.5.a.2, a site specific analysis has been performed demonstrating the postulated drops from the lift heights for the HI-TRAC and the HI-STORM will not result in g loads on the MPC in excess of 45 g's. (Ref: Holtec Report HI-2073755, Postulated Mechanical Drop Accidents at IP-1.) Impact limiters will be used when lifting the loaded HI-TRAC with the VCT before the redundant locking feature is engaged. Similarly impact limiters will be used at the ISFSI pad when the redundant locking features are disengaged while lowering the HI-STORM onto the ISFSI pad. Other than industry codes such as NUREG-1536 and ACI, the FSAR does not specify the analysis methodology means of performing the pad analysis to meet specific limits.

### E.3.9 § 72.212 (b)(2)(i)(C)-Dose Limitations per § 72.104

*72.212(b)(2)(i)(C): Perform written evaluations, prior to use, that establish that the requirements of §72.104 have been met. A copy of this record must be retained until spent fuel is no longer stored under the general license issued under §72.210.*

10 CFR 72.104 requires that for normal operation and anticipated occurrences, the annual dose to any real individual beyond the controlled area must not exceed 25 mrem whole body and 75 mrem to the thyroid, or 25 mrem to any other organ from any discharge or direct radiation from the ISFSI and any other uranium fuel cycle facilities in the region, i.e., IPEC Units 1, 2, and 3.

The Unit 1 fuel latest discharge date was October 31, 1974 and has decayed in excess of 30 years. It has a maximum burnup of less than 27,100 MWD/MTU. The design basis fuel applicable to IP-1 assumed by Holtec for loading into the MPC-32 for site boundary doses and Tech Spec 5.7 limits has a burnup of 30,000 MWD/MTU and an assumed cooling time of 30 years for Unit 1. (Reference: Holtec Reports HI-2073724, Dose versus Distance From a HI-STORM 100S Version B Containing the MPC-32, dated July 30, 2007 and HI-2073736, HI-STORM CoC Radiation Protection Program Dose Rate Limits for IP-1 and IP-2, dated September 7, 2007)

Exclusive of the ISFSI, the normal operating doses from the IPEC site are documented in the Annual Radiological Environmental Operating Report. Direct radiation is measured at 18 locations within 2 miles of the plant site. The dose contribution from the operating plant facilities to a member of the public at the site boundary has been recorded at less than 1 mrem/year in the Report. Calculated or design dose contributions from other site sources, e.g. Temporary Low Level Storage Building, Steam Generator Mausoleums, totals approximately 7.5 mr/year at the site boundary. The calculated dose for the five Unit 1 HI-STORMS and the six Unit 2 HI-STORMS at the nearest real resident, co-incident at the eastern site boundary at approximately 1970 feet is slightly less than 1 mrem/year. The calculated dose from the ISFSI combined with the recorded and calculated design dose from the remaining facilities (Units 1, 2, and 3) is less than the 40 CFR 190 limit of 25 mrem/yr. As more casks are loaded into the ISFSI from Units 2 and 3, dose at the site boundary and the nearest residence will be re-evaluated.

**E.3.10 § 72.212(b)(3)- Review of the Cask FSAR and SER**

*72.212(b)(3): Review of the Safety Analysis Report (FSAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report, prior to the use of the general license, to determine whether or not the reactor site parameters, including analyses of the earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The result of this review must be documented in the evaluation made in paragraph (b) (2) of this section.*

The HI-STORM 100 System FSAR and the NRC's Safety Evaluation Report have been reviewed and a determination made that the reactor site parameters at the Indian Point Energy Center are bounded by the assumptions made in the generic cask FSAR and the NRC's safety basis in the SER. The details of this review as they pertain to issues such as earthquake intensity and tornado missiles may be found elsewhere in this appendix. Section E.5 of this appendix provides a chapter-by-chapter assessment of deviations between the requirements in the cask FSAR and implementation of dry cask storage at the IPEC site. All cask FSAR requirements of the license that are not listed in Section E.5 of this appendix are met. Deviations from cask FSAR requirements have been evaluated under the IPEC 10 CFR 72.48 program as applicable.

**E.3.11 § 72,212(b)(4)- Changes to 10 CFR Part 50 Technical Specifications**

*72.212(b)(4): Prior to use of the general license, determine whether activities related to storage of spent fuel under this general license involve a change in the facility Technical Specifications or require a license amendment for the facility pursuant to § 50.59 (c )(2) of this chapter. Results of this determination must be documented in the evaluation made in paragraph (b)(2) of this section.*

Several design modifications, each having their own 10 CFR 50.59 evaluation, have been implemented in support of dry fuel storage at IPEC. None of these modifications required a change to the Unit 1 operating license or technical specifications. However, a Part 50 license amendment that did not involve a change to the IPEC Unit 1 Part 50 technical specifications was required to obtain NRC approval for the use of the Fuel Handling Building crane for loading operations. The NRC had previously reviewed the use of a 40 ton IF-200 cask at Unit 1 but had not reviewed the use of a 75 ton HI-TRAC variant for use at IP-1. In addition, the original NRC evaluation of Fuel Cask handling in the Fuel Handling Building did not evaluate the drop of a lid on a loaded canister or the impact on loaded fuel in a canister assuming the drop of the transfer cask. IPEC Unit 1 submitted a License Amendment Request for Amendment 53, dated February 22, 2007, Letter No. NL-07-033, to the NRC for its review and approval.

The NRC granted approval for these issues through Amendment No. 53 to the Unit 1 license on May 9, 2008 (Reference RA-08-072).

### E.3.12 § 72.212(b)(5)-Protection Against Radiological Sabotage

*72.212(b)(5): Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to §73.55 of this chapter with the following additional conditions and exceptions:*

*(i) The physical security organization and program for the facility must be modified as necessary to assure that the activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with § 73.55 of this chapter.*

*(ii) Storage of spent fuel must be within a protected area, in accordance with § 73.55 of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with the general license.*

*(iii) For purposes of this general license, searches required by § 73.55(d)(1) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosive detection equipment.*

*(iv) The observational capability required by § 73.55(h)(6) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of closed circuit television.*

(v) *For the purposes of this general license, the licensee is exempt from § § 73.55(h)(4)(iii)(A) and 73.55(h)(5) of this chapter.*

The ISFSI is located in an expanded portion of the existing IPEC Unit 2 protected area. During transport from the Unit 1 FHB to the Unit 2 FHB and then to the ISFSI the spent fuel will be inside the protected area. The IPEC Security Plan has been updated to include the ISFSI in accordance with the requirements of 10 CFR 50.54(p). A review of the plan concludes the inclusion of the ISFSI in the IPEC security plan does not reduce the effectiveness of the plan with regard to the protection of existing facility vital equipment.

**E.3.13 §72.212(b)(6)- Review of the Emergency Plan, Quality Assurance Program, Training Program, and Radiation Protection Program**

*72.212(b)(6): Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.*

Emergency Plan

The IPEC Emergency Plan is maintained to meet the regulations in 10 CFR 50.47, 10 CFR 50.54, 10 CFR 50.72, and 10 CFR50 Appendix E. The plan follows the guidelines established in NUREG-0654/FEMA REP-1. 10 CFR 50.47(b) lists the sixteen planning standards that must be met in the Emergency Plan. The IPEC Emergency Plan was reviewed to determine if changes required to support implementation of the Holtec HI-STORM 100 System at the IPEC ISFSI will decrease the effectiveness of the Emergency Plan as required by 10 CFR 50.54 (q).

IPEC Unit 2 placed its first dry cask storage cask on the IPEC ISFSI on January 11, 2008. Since the Emergency Plan is applicable to the Site, rather than a particular Unit, the Plan was modified and the necessary revisions implemented to support the Unit 2 effort. The details are further documented in the Unit 2 10 CFR 72.212 Report, Appendix F, Revision 2.

Quality Assurance Program

The Entergy Nuclear (EN) Quality Assurance Program and Quality Assurance Program Manual (QAPM) were reviewed to assure compliance to the requirements of 10 CFR 72 for the handling, transporting, and storage of dry fuel storage canisters.

The Holtec HI-STORM 100 System QA requirements are found (by reference to the Holtec QA program Manual) in the HI-STORM FSAR and CoC, which imposes the requirements of 10 CFR 72 on both licensees and certificate holders. The EN QAPM applies to all activities associated with structures, systems, and components which are safety related or controlled by 10 CFR 72. The methods of implementation of the requirements of the QAPM are commensurate with the item's or activity's importance to safety. The QAPM implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G.

Holtec uses a graded quality approach to various subcomponents associated with the HI-STORM overpack, the HI-TRAC transfer cask, the MPC, and the ancillary components used to facilitate cask loading and onsite transport. The approach is covered by EN and/or site specific procedures that implement the QAPM.

A review of the QAPM concludes that the implementation of the ISFSI related quality assurance activities does not reduce the effectiveness of the Quality Assurance Program. Additionally there are no reductions in commitments in the QA Program as a result of the ISFSI as required by 10 CFR 50.54(a)(3).

IPEC Unit 2 placed its first dry cask storage cask on the IPEC ISFSI on January 11, 2008. Since the Entergy Quality Assurance Program is applicable to the entire fleet, rather than a particular Unit, the Plan was modified and the necessary revisions implemented to support the Dry Cask Storage effort at all the sites. The details are further documented in the Unit 2 10 CFR 72.212 Report, Appendix F, Revision 2.

Training Program

The HI-STORM 100 storage system training program requirements are found in the HI-STORM FASR and CoC, which invoke the requirements of 10 CFR 72 and require cask design-specific topics for personnel training. The IPEC dry fuel storage training program uses the Systematic Approach to Training (SAT) and is based on 10 CFR 50.120 and INPO guidelines. The training program addresses all training requirements specifically listed in the HI-STORM FSAR and CoC as well as training requirements in other parts of the 10 CFR 72 regulations. See Section E.3.5 for a more detailed discussion of the training program.

In accordance with the requirements of 10 CFR 50.120 (b)(2) the training program is periodically reviewed by management to insure continued effectiveness. This review has concluded that the addition of the dry cask storage activities has had no negative impact on the effectiveness of the program.

IPEC Unit 2 placed its first dry cask storage cask on the IPEC ISFSI on January 11, 2008. Since the Training Program is applicable to the Site, rather than a particular Unit, the Program was modified and the necessary revisions implemented to support the Unit 2 effort. The details are further documented in the Unit 2 10 CFR 72.212 Report, Appendix F, Revision 2.

Personnel qualifications, Training modules, and documentation were reviewed in conjunction with the dry runs conducted in preparation for the Unit 2 Cask Loading effort. Items that are unique to Unit 1, such as crane evolutions, fuel handling, and some ancillary equipment operation are documented specifically for Unit 1.

Radiation Protection Program

The HI-STORM 100 System radiological protection requirements are found in the HI-STORM FSAR and CoC, which invoke the requirements of 10 CFR 72 and 10 CFR 20 and provide cask specific requirements. The IPEC Radiological Protection Program has been reviewed and modified as necessary to address the loading and unloading activities as well as continued storage operations.

IPEC Unit 2 placed its first dry cask storage cask on the IPEC ISFSI on January 11, 2008. Since the Radiological Protection Program is applicable to the Site, rather than a particular Unit, the Plan was modified and the necessary revisions implemented to support the Unit 2 effort. The details are further documented in the Unit 2 10 CFR 72.212 Report, Appendix F, Revision 2.

In addition, the requirements in HI-STORM Co Appendix A, Section 5.7 "Radiation Protection Program," have been addressed for the Unit 1 fuel load and HI-TRAC design. Refer to Section E.4.2.4 of this appendix for more information.

A review of the program concluded that the activities related to ISFSI activities has not reduced the effectiveness of the IPEC Radiological Program.

# SECTION E.4
# COMPLIANCE WITH HI-STORM 100 CASK SYSTEM CERTIFICATE OF COMPLIANCE

## E.4.1 Certificate of Compliance Conditions

Compliance with the Holtec HI-STORM 100 System Certificate of Compliance (CoC) is discussed on an Entergy system-wide basis in Section VI of the main body of this report. Conditions 5, 9, and 10 of the CoC, where compliance requires a unique discussion for the IPEC Unit 1 facility, are addressed below. The Holtec CoC discussed is the NRC approved Certificate through and including Amendments 1, 2, and 4. (Amendment 3 is not applicable to IPEC Unit 1)

### E.4.1.1 Condition 5-Heavy Loads Evaluation

*Each lift of an MPC, a HI-TRAC transfer Cask, or any HI-STORM overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant specific regulatory review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable)is required to show compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.5 of Appendix A and/or Sections 3.4.6 and 3.5 of Appendix B to this certificate, as applicable.*

At IPEC Unit 1, only the MPC, MPC Lid, and HI-TRAC transfer cask will be lifted with cranes and other lift equipment falling under the site's Part 50 heavy load evaluations. All lifts of the MPC and HI-TRAC in the Unit 1 Fuel Handling Building, using the Unit 1 Fuel Handling Crane, will be in accordance with the site's heavy load protocol as defined by NUREG-0612 and implemented with the site's cask loading procedures.

The Unit 1 licensing basis does include the handling of the GE-IF-200 shipping cask system, but that device is lighter and smaller than the HI-STORM 100-IP1 components. Unit 1 has loaded and shipped 244 fuel assemblies to an offsite facility using GE-IF-200 casks. The IF-200 is a two element cask and the campaign required the handling, loading, and transport of at least 120 individual shipments using the existing Unit 1 crane and facilities arrangement.

The Unit 1 Fuel Handling Building crane is not single-failure-proof . A comprehensive NUREG-0612-based review of the heavy load handling tasks required to be performed by the crane in support of cask loading operations was performed. This review identified and evaluated all heavy load lifts involving the MPC, MPC lid, and transfer cask to be performed in the UNIT 1 Fuel Handling Building by the crane's main hoist and compares these lifts to regulatory guidance (e.g. NUREG-0612), generic NRC Communications (e.g. NRC Bulletin 96-02 and RIS-2005-25) and the current Unit 1 licensing basis and commitments.

This review concluded that certain cask and cask component lifts were not bounded by the existing Part 50 licensing basis and an amendment to the IPEC Unit 1 license was required pursuant to 10 CFR 50.59. For operations using the Unit 1 Fuel Handling Building Crane, a License Amendment Request (LAR) for Amendment 53 was submitted by Entergy on February 22, 2007 requesting NRC approval for the use of the Unit 1 FHBC for HI-STORM 100 System handling activities that are outside the existing licensing basis. NRC approval was received on May 9, 2008.

Included in the Unit 1 LAR were descriptions of all the lifts and load paths of heavy loads in the Unit 1 FHB, postulated load drops, and the analyses and evaluations of particular load drop cases that bounded all postulated load drop scenarios. See Section E.1.5 for a summary of the load drops evaluated in the LAR and Section E.3.11 for the Unit 1 license amendment number.

Crane lifts required to insert the loaded MPC into the HI-STORM overpack (stack up) will be performed using the IPEC Unit 2 Fuel Storage Building Gantry Crane. This crane is single failure proof and has been used to perform the identical lifts during the loading of Unit 2 dry cask storage MPC's and Transfer Casks. The Unit 1 MPC's and HI-TRAC are of nearly identical design to the MPC-32 and HI-TRAC used for Unit 2 except that the Unit 1 components and slightly shorter in height and lower in overall weight than the Unit 2 components.

The loaded Unit 1 MPCs in the HI-TRAC transfer casks will be transported from the Unit 1 Fuel Handling Building to the Unit 2 Fuel Storage Building for stack up and loading, using an air pad transporter and the Vertical Cask Transporter (VCT).

The loaded Unit 1 HI-STORM overpacks will be transported from the Unit 2 Fuel Storage Building to the ISFSI using as Low Profile Transporter (LPT) and the Vertical Cask Transporter (VCT).

The VCT is a commercial grade item designed with substantial safety factors. The VCT is designed in accordance with ANSI N14.6 and has redundant drop protection features as required by the HI-STORM CoC, Appendix A, Section 5.5.a.3. A drop of the HI-STORM (or the HI-TRAC) is not considered a credible event during transport of the overpack to the ISFSI except while actually raising or lowering the load.

To evaluate the short period of time when the HI-TRAC and the HI-STORM is being raised or lowered by the VCT and the redundant drop protection feature is not engaged, drop analyses have been performed in accordance with Holtec Technical Specification 5.5.a.2. These analyses have determined the acceptable lift heights for the HI-TRAC for the lifting of the HI-TRAC off the air pads outside the Unit 1 FHB and the lowering of the HI-TRAC onto the Unit 2 LPT. These analyses also address the lifting of the HI-STORM over pack off the LPT and the lowering of the over pack onto the ISFSI pad. (Reference: Holtec Calculation HI-2073755, "Postulated Mechanical Drop Accidents at IP-1)

**E.4.1.2 Condition 9-Special Requirement for First System In Place**

*The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM cask systems (for each unique MPC basket design-MPC-24/24E/24EF, MPC-32/32F, and MPC-68/68F/68FF) placed into service, by any user, with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR.*

*Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to of greater than 16 kW.*

*Each first time user of a HI-STOTM 100 Cask System Supplemental Cooling System (SCS) that uses components or a system that is not essentially identical to components or a system the has been previously tested, shall measure and record coolant temperatures for the inlet and outlet of the cooling provided to the annulus between the HI-TRAC and MPC and the coolant flow rate. The user shall also record the MPC operating pressure and decay heat. An analysis shall be performed, using this information, that validates the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling necessary.*

*Letter reports summarizing the results of each thermal validation test and SCS validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.*

The five HI-STORM 100S-185 System MPC's are a variant of the MPC-32 discussed in Chapter 4 of the FSAR. The Indian Point Unit 1 MPC's are designed to load fuel into damaged fuel containers in each of the 32 basket locations (as opposed to a maximum of eight damaged fuel containers in the standard MPC-32). The IP-1 MPC is also shorter than the other MPC-32 variants.

These variations, unique to IP-1 are discussed in Amendment 4 to the Holtec CoC.

Although the IP-1 MPC's, HI-TRAC transfer casks, and HI-STORM overpacks are a unique variant to the MPC-32 design, the age of the Indian Point Unit 1 fuel, all with over 32 years of cooling time, and the low burnup history, all less than 27,100 MWD/MTU and many less than 4,000 MWD/MTU, the heat load for all the IP-1 casks is less than 10 kW. Consequently that portion of this CoC requirement pertaining to cask heat load is not applicable to IP-1. No temperature data is required to be taken and no reports need to be submitted.

The portion of this CoC requirement pertaining to the SCS applies only to licensees who load high burnup fuel (burnup > 45,000 MWD/MTU). IP-1 as previously discussed does not have high burnup fuel to be loaded. Therefore, this requirement of the CoC is also not applicable.

**E.4.1.3 Condition 10-Pre-Operational Testing and Training Exercise**

*A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be preformed in an alternate step sequence from the actual procedures, but all the steps must be performed. The dry run shall include, but is not limited to the following:*

a. *Moving the MPC and transfer cask into the spent fuel pool or cask load pool.*

b. *Preparation of the HI-STORM 100 cask system for fuel loading.*

c. *Selection and verification of specific fuel assemblies to ensure type conformance.*

d. *Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.*

e. *Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool Or cask load pool.*

f. *MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable),* *and helium backfilling. A demonstration of the welding and helium leak test of the MPC vent and drain port covers is also included in this item.(A mockup may be used for this dry-run exercise.)*

g. *Operation of the Supplemental Cooling System if applicable.*

h. *Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling equipment.*

i. *Transfer of the MPC from the transfer cask to the overpack.*

j. *Placement of the HI-STORM 100 Cask System at the ISFSI.*

k. *HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds. ( A mockup may be used for this dry run exercise.)*

Indian Point Unit 2 has loaded MPC-32's within a few months of the Unit 1 campaign, IPEC has determined which of the items of the Dry Run Exercise need to be repeated at the IPEC site to fulfill the Unit 1 license conditions and which have already been satisfied.

Certain items are unique to Unit 1 in that they use Unit 1 unique equipment, use Unit 1 specific procedures, or are performed in a physical arrangement peculiar to Unit 1. These include moving the MPC and transfer cask to the pool, selection of the fuel assemblies, loading assemblies, installation of lids and removal of cask from the pool, and using the air pads and VCT.
Specifically these items 10a, 10b, 10c, 10d, and 10e will be included in the dry runs performed to meet Unit 1 CoC requirements.

Other items are similar or identical to the process and/or equipment which were used by the Unit 2 fuel loading campaign. These include preparation of cask system, except for loading of Damaged Fuel Cans, welding, NDE, Forced Helium Drying, stack up of the Transfer cask in the Unit 2 FSB, placement on ISFSI pad, and unloading. The evolutions use the same equipment, similar procedures and are performed by the same cask loading team. Specifically these are items 10f, 10i, 10j and 10k. These activities, documented in NRC Inspection Report, 05000247/2007009 and 07200051/2007001, dated April 10, 2008, AA No. ML081020398, will not be repeated during the "NRC" dry run for Unit 1.

Other items are not applicable to Unit 1 activities as determined by the Technical Specifications. These include operation of SCS, and upending and downending of MPC/Transfer Cask. Specifically these are items 10g and 10 h.

### E.4.2 COC Appendix A-Technical Specifications

Compliance with the Holtec HI-STORM 100 System CoC, Appendix A, "Technical specifications," is discussed on an Entergy system-wide basis in Section VI of the main body of this report. Technical Specifications where compliance requires a unique discussion for the IPEC Unit 1 facility are discussed below:

#### E.4.2.1 Limiting Conditions for Operation (LCO's) and Surveillance Requirements (SRs).

#### E.4.2.1.1 LCO and SR 3.0 Series- LCO and SR Applicability.

The LCO and SR 3.0 series of technical specifications establish general requirements for use and implementation of the specific LCOs and SRs that follow. No specific actions or implementation procedures are required.

#### E.4.2.1.2 LCO 3.1.1-Multi-Purpose Canister

This LCO establishes the MPC fuel cavity drying and helium backfill acceptance criteria for establishing the required heat transfer and corrosion-resistant environment for the stored fuel. With regard to the MPC cavity drying, IPEC Unit 1 is choosing to use the Force Helium Dehydration (FHD) system. Therefore the drying acceptance criteria for the FHD system in Table 3-1 of the HI-STORM CoC Appendix A apply. The helium backfill pressure range for MPC-32/32F in Table 3-2 of HI-STORM CoC Appendix A is used as the acceptance criterion rather than the "gram-moles/liter" acceptance criterion. These requirements are implemented via IPEC Procedures Nos. 0-DCS-23-GEN, FHD Operations and 1-DCS-028-GEN, Unit 1 MPC Load and Seal.

#### E.4.2.1.3 LCO 3.1.2-SFSC Heat Removal System

This LCO establishes operability and surveillance requirements for the HI-STORM overpack natural ventilation heat removal system. For the loaded HI-STORM overpacks stored on the ISFSI pad, daily surveillance of the inlet and outlet air ducts for blockage is performed by IPEC Operations during daily operator rounds and the use of an installed Resistance Temperature Detector (RTD) system.

**E.4.2.1.4 LCO-3.1.3- Fuel Cooldown**

This LCO establishes requirements for ensuring that the bulk temperature of the MPC fuel cavity gas is less than or equal to $200^{0}$ F before reflooding the cavity with water in the event an MPC needs to be unloaded. This LCO is implemented via IPEC Procedure 1-DCS-035-GEN, "Unit 1 MPC unloading Procedure". Refer to Section E.1.4 for a summary of the cask unloading operational sequence where this LCO would apply.

**E.4.2.1.5 LCO 3.1.4-Supplemental Cooling System(SCS)**

This LCO establishes requirements for operation of the SCS required to be used if one or more high burnup fuel assemblies (burnup > 45,000MWD/MTU) is loaded into the MPC. Unit 1 fuel has a maximum burnup of <27,100 MWD/MTU, therefore this LCO is not applicable to Unit 1 fuel loading.

**E.4.2.1.6 LCO 3.2.2-Transfer Cask Surface Contamination**

This LCO establishes removable alpha, beta, and gamma radiation contamination limits for the transfer cask surfaces and accessible portions of the MPC during onsite transport operations. IPEC 0-RP-RWP-420 includes steps to ensure that the transfer cask and accessible portions of the MPC are decontaminated to levels meeting the limits specified in the LCO prior to entering the TRANSPORT OPERATIONS mode.

In Section 1.1 of the Technical Specifications-Definitions, " TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER cask loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER cask is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS includes the transfer of the MPC between the OVERPACK and the TRANSFER CASK."

IPEC Unit 1 intends to transfer the loaded MPC and TRANSFER CASK from the Unit 1 FHB to the adjacent Unit 2 FSB for stack up and transfer to the OVERPACK using the single failure proof crane in the Unit 2 FSB. During this movement between Units 1 and 2 the MPC and TRANSFER CASK will continuously remain within the protected area of the plant.

The IPEC loose surface contamination control procedure (0-RP-RWP-420) will be implemented to control any loose surface contamination on external surfaces of the TRANSFER CASK during this movement between fuel buildings.

The applicability note in the Technical Specification "This LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING." implements this LCO after the loaded OVERPACK is loaded on the Low Profile

Transporter for TRANSPORT OPERATIONS inside the Unit 2 FSB for transport to the ISFSI.

**E.4.2.1.7 LCO 3.3.1 – Boron Concentration**

This LCO establishes minimum soluble boron concentration requirements in the MPC water during fuel loading in certain MPC designs at pressurized water reactor plants. The unique-to-Indian Point Unit 1 HI-STORM 100S-185 System is designed with no boron limit and no burnup credit as discussed in LAR and Amendment 4 of the Holtec CoC. This LCO is not applicable to the loading of IP-1 fuel.

Current 10 CFR 50 Unit 1 Technical Specifications do not specify a boron limit in the Spent Fuel Pool of the Cask Loading Pool.

**E.4.2.2Section 5.4-Radioactive Effluent Control Program**

*This program implements the requirements of 10 CFR 72.44(d).*

a. *The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.*

b. *This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR 50 operations.*

c. *An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3)*

An annual report is submitted pursuant to 10 CFR 72.44(d)(3) requirements in accordance with the "Offsite Dose Calculation Manual."

**E.4.2.3Section 5.5-Cask Transport Evaluation Program**

**[Note: This technical specification, contained in Amendment 4 of the Holtec CoC, is unique to Unit 1 and is not applicable to Unit 2 or 3 Cask Loading operations.]**

*This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFR CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or OVERPACK is in the*

*FUEL BUILDING or is being handled by a device providing support from underneath (i.e. on a rail car, heavy haul trailer, air pads, etc.).*

a. *For free standing OVERPACKS and the TRANSFER CASK, the following requirements apply:*

1. *The lift height above the transport route surface(s) shall not exceed the limits in [CoC] Table 5-1 except as provided for in Specification 5.5.a.2. Also, if applying the limits in Table 5-1, the program shall ensure that the transportation route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than either Set A or Set B in HI-STORM FSAR Table 2.2.9.*

2. *The program may determine lift heights by analysis based on the site specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g's at the top of the MPC fuel basket. These alternative analyses shall be commensurate with the drop analyses described in the Final Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.*

3. *The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during TRANSPORT OPERATIONS provided the lifting device is designed in accordance with ANSI N 14.6 and has redundant drop protection features.*

4. *The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specific in Section 3.5 of Appendix B to Certificate of Compliance 1014, as applicable.*

b. *For the transport of OVERPACKS to be anchored to the ISFSI pad the following requirements apply......*

For the lifting of the loaded HI-TRAC, lifts which are integral to either the Unit 1 FHB or the UNIT 2 FSB are governed by 10 CFR 50 regulations. The transport of the HI-TRAC between Unit 1 and Unit 2 is performed using air pads and the Vertical Cask Transporter (VCT). These lifts are defined by site-specific calculations and procedures per Technical Specification 5.5.a.2.

The Unit 2 Low Profile Transporter and the VCT is used to convey the loaded OVERPACK from the Unit 2 FSB to the ISFSI pad.

The Vertical Cask Transporter meets the requirements of Technical Specification 5.5.a.3. and a drop of the HI-STORM or the HI-TRAC during TRANSPORT operations is not considered a credible event when the redundant drop protection features are engaged. (Reference: Letter; Joe Reiss, Holtec to P. Peloquin, IPEC; "IPEC Vertical Cask Transporter Compliance with the HI-STORM CoC"; Document ID: 1535005; October 24, 2007.)

However due to the physical requirements of the LPT and the VCT, a lift height for the Overpacks of about 11" is required. This is in excess of the "Approved" lift height of 8" for Unit 1 in Table 5-1 of the Technical Specifications. In accordance with Technical Specification 5.5.a.2, a site specific analysis has been performed demonstrating the postulated drops will not result in g loads on the MPC in excess of 45 g's. (Ref: Holtec Reports HI-2073755, dated September 26, 2007 for Unit 1 and HI-2073720, August 2, 2007 for Unit 2.)

The IPEC ISFSI design utilizes free standing OVERPACKS. Section 5.5.b of this Specification does not apply to IPEC

**E.4.2.4Section 5.7 Radiation Protection Program**

1. *Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK and TRANSFER CASK outside the facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposure AS LOW AS Reasonably Achievable (ALARA). The action and criteria to be included in the program are provided below.*

2. *As part of its evaluation pursuant to 10 CFR 72.212(b)(i)(C ), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.*

3. *Based on the analysis performed pursuant to [item 2], the licensee shall establish individual cask surface dose rate limits for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:*

   a. *The top of the TRANSFER CASK and the OVERPACK*

   b. *The side of the TRANSFER CASK and OVERPACK*

   c. *The inlet and outlet ducts on the OVERPACK*

4. *Not withstanding the limits established in [item3], the measure dose rates on a loaded OVERPACK shall not exceed the following values:*

   a. *20 mrem/hr (gamma + neutron) on the top of the OVERPACK*

   b. *110 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts.*

5. *The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in [item 8] for comparison against the limits established in [item 3] of [item4], whichever are lower.*

6. *If the measure surface dose rates exceed the lower of the two limits established in [item 3] or [item 4], the licensee shall:*

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

   b. *Perform a written evaluation to verify whether placement of the as-loaded OVERPACK at the ISFSI will cause the dose limits of 10 CFR 72.104 to be exceeded, [and]*

   c. *Perform a written evaluation within 30 days to determine why the surface dose rates were exceeded.*

7. *If the evaluation performed pursuant to [item 6] shows that the dose limits of 10 CFR 72.104 will be exceeded, the OVERPACK shall not be placed into storage until the appropriate corrective action is taken to ensure the dose limits are not exceeded.*

8. *TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:*

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

   b. *A minimum of four (4) TRANSFER CASK top lid dose rates shall be measured at locations approximately halfway between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid.*

   c. *A minimum of twelve (12) dose rate measurements shall be taken on the side of the OVEROPACK in three sets of four measurements. One measurement set shall be taken at approximately the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.*

   d. *A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the egde of the top concrete shield, 90 degrees apart around the circumference of the lid.*

   e. *A dose rate measurement shall be taken on contact at the surface of each inlet and outlet vent duct screen.*

A site specific analysis was performed in accordance with the HI-STORM CoC Appendix A, Sections 5.7.2 and 5.7.3. (Holtec Calculation HI-2073736) The analysis for the five canisters to be loaded assumed 30,000 MWD/MTU and 30 Years Cooling.

The calculated dose rates on the top and on the sides of the OVERPACK are lower than the CoC Appendix A, Section 5.7.4 limits of 20 mrem/hr and 110 mrem/hour, respectively. Therefore, the calculated total surface dose rates are the appropriate limits to apply for comparison to measured values.

HI-TRAC Transfer Cask dose values assume the MPC is drained and water free.

IPEC Procedure 0-RP-RWP-420 implements the dose rate measurement requirements of the CoC Appendix A, Section 5.7.8 for the HI-TRAC transfer cask and the HI-STORM overpack. The measured dose rates are compared to the limits established in the tables above in accordance with CoC Appendix A, Section 5.7.5. If the measure dose rates exceed the established limits, the actions required by CoC Appendix A, Sections 5.7.6 and 5.7.7 (if required) will be implemented.

### E.4.3 CoC Appendix B-Approved Contents and Design Features

Section 2.0 of Appendix B to the HI-STORM CoC ,"Approved Contents" is discussed in the main body of this report, Section VI. An IPEC procedure is used to select fuel assemblies for storage in the HI-STORM 100 system that meets all the applicable requirements of HI-STORM CoC Amendment 5, Appendix B, Sections 2.1, and 2.4 (Sections 2.2 and 2.3 do not exist in the Appendix).

Section 3.0 of Appendix B to the HI-STORM CoC, "Design Features," is discussed in different sections of this report. Section 3.1 is addressed in Section VI of the main body of this report. Sections 3.2 , 3.3, 3.4, 3.6, 3.7, and 3.8 are addressed site-specifically for the IPEC ISFSI in the following paragraphs. Section 3.5 of Appendix B to the CoC pertains to a Cask Transfer Facility (CTF) which is not used for cask loading operations at IPEC and is not applicable and not discussed.

#### E.4.3.1 Section 3.2-Design Features Important for Criticality Control.

This section of the CoC addresses certain design features important for criticality control for all HI-STORM 100 System MPC models certified under 10 CFR 72. Sections 3.2.1, 3.2.2, 3.2.3, and 3.2.4 pertain to the MPC-24, MPC-68/FF, MPC-68F and MPC-24E/EF. None of these MPC's are licensed for use by Indian Point Unit 1 and are not discussed in this report. The Unit 1 MPC is a variant of MPC-32/32F and Section 3.2.5 as amended by Amendment 4 of the CoC is applicable to IP-1.

**E.4.3.1.1 Section 3.2.5-MPC-32/32F**

1. *Fuel Cell* pitch:≥9.158 *inches*

2. *$^{10}B$ loading in the neutron absorbers:≥ 0.0372 g/cm$^2$ (Boral) and ≥ 0.0310 g/cm$^2$(Metamic)*

The fuel cell pitch and the Boron loading of the neutron absorbers in the MPC are verified as part of MPC fabrication. Certification that each MPC meets these technical specification limits is provided by Holtec in the Component Completion Record (CCR) for each serial number MPC. The design of each MPC is checked to ensure that it meets the specific design features for criticality. Each MPC is then manufactured and certified that it meets the design requirements.

**E.4.3.1.2 Section 3.2.6-Fuel Spacers**

*Fuel spacers shall be sized to ensure that the active fuel region of the intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.*

The IPEC Unit 1 fuel design is a unique design designated Indian Point 1 "Core B" manufactured by Westinghouse which is 137. 2 inches long with an active fuel length of 100.3 inches with the exception of 32 assemblies (of a total of160) which have an active fuel length of 90 inches. Fuel spacers are not used in the IP-1 design MPC.

**E.4.3.1.3 Section 3.2.7-METAMIC $B_4$ C Content**

*The $B_4$ C content in METAMIC shall be ≤ 33.0 wt.%*

The limit was verified to be met by Holtec International during the MPC fabrication process as documented in the Component Completion Records for the MPC's.

**E.4.3.1.4 Section 3.2.8-Neutron Absorber Tests**

*Section 9.1.5.3 of the HI-STORM 100 FSAR is incorporated by reference into the HI-STORM 100 CoC and cannot be amended without prior approval from the USNRC. The minimum $^{10}B$ for the neutron absorber material shall meet the minimum requirements for each MPC model specified in Sections 3.2.1 through 3.2.5 above.*

This CoC requirement was verified to be met by Holtec International during the neutron absorber fabrication process as documented in the Component Completion Records for the MPC's.

**E. 4.3.2 Section 3.4-Site Specific Parameters and Analysis**

The Unit 1 FSAR does not discuss site hydrology, meteorology, geology, or natural phenomena. Consequently the information contained in chapter 2 of the Unit 2 UFSAR was used as a source for site specific parameters.

**E.4.3.2.1 Section 3.4.1-Maximum Normal Ambient Temperature**

*The temperature of 80$^0$F is the maximum average yearly temperature.*

For New York City, NY, located approximately 30 miles south of the Indian Point site and at the same approximate elevation, the highest and lowest average annual temperatures between the years 1909 and 2002 are 52.7$^0$ F (1998) and 46.8$^0$F (1958), respectively. (Reference: National Climatic Date Center-National Oceanic and Atmospheric Administration.). Therefore, it is concluded that the maximum average yearly temperature limit will not be exceeded.

**E.4.3.2.2 Section 3.4.2-Ambient Temperature Extremes**

*The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40$^0$F and less than 125$^0$F.*

The Temperature extremes for New York State are as follows:

| | | | |
|---|---|---|---|
| Highest: | Troy, New York | 108$^0$F | July 22, 1926 |
| Lowest | Old Forge, New York | -52$^0$F | Feb.18, 1979 |

The maximum temperature extreme does not approach the 125$^0$F limit and it can be concluded with reasonable assurance that the 3-day average maximum temperature limit will not be exceeded.

As for the minimum temperature extreme, the four lowest recorded temperatures by month in New York State are:

| | | | |
|---|---|---|---|
| January: | Paul Smiths, | -46$^0$ F | (1904) |
| February: | Old Forge, | -52$^0$F | (1979) |
| March: | Chazy, | -41$^0$F | (1938) |
| December: | Philadelphia NY | -47$^0$F | (1933) |

Each of these low temperatures records for their respective months were absolute lows and not averaged over a 3-day period. All four locations are in upstate New York.

On February 18, 1979, the date of the lowest recorded temperature for New York State, the minimum temperature recorded in New York City was -1$^0$F. The 3-day minimum temperatures in New York City for February 17 through February 19, 1979 were 4$^0$F, -1$^0$F, and 17$^0$F. Since New York City is only 25 miles south of Indian Point and at approximately the same elevation, the temperatures at Indian Point do not approach the low temperatures recorded in upstate New York. (Reference: National Climatic Date Center-National Oceanic and Atmospheric Administration.).

Therefore it can be concluded, with reasonable assurance, that the -40$^0$ F 3-day minimum temperature limit will not be exceeded.

**E.4.3.2.3 Section 3.4.3-Seismic Criteria**

a. *The resultant horizontal acceleration (vectorial sum of two horizontal Zero period Accelerations (ZPA's) at a three-dimensional seismic site), $G_H$, and vertical ZPA, $G_V$ on the top surface of the ISFSI pad, expressed as fractions of 'g', shall satisfy the following inequality:*

$$G_H + \mu\ G_V \leq \mu$$

*where $\mu$ is either the Coulomb friction coefficient for the cask/ISFSI interface or the ratio r/h, where 'r' is the radius of the cask and 'h' is the height of the cask center-of-gravity above the ISFSI pad surface. The above inequality must be met for both definitions of $\mu$, but only applies to ISFSIs where the casks are deployed in a freestanding configuration. Unless demonstrated by appropriate testing that a higher coefficient of friction value is appropriate for a specific ISFSI, the value used shall be 0.53. If acceleration time-histories on the ISFSI pad surface are available, $G_H$ and $G_V$ may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality shall be evaluated at each step in the acceleration time history over the total duration of the seismic event.*

*If this static equilibrium based on inequality cannot be met, a dynamic analysis of the cask/ISFSI pad assemblage with appropriate recognition for soil/structure interaction effects shall be performed to ensure that the casks will not tip over or undergo excessive sliding under the site's Design Basis Earthquake.*

b. *For free-standing casks, under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will* *not result in cask tipover or a cask to fall off the pad. In addition, impacts between casks should be precluded, or should be considered an accident for which the maximum g-load experienced by the stored fuel shall be limited to 45 g's.*

c. *(This section applies to ISFSI site requiring anchored Overpacks and is not applicable to Indian Point)*

Section 1.2.2.11 of the Indian Point 2 FSAR identifies that ground motion for a safe shutdown earthquake (SSE) corresponds to a maximum horizontal acceleration of 0.15g. The orthogonal sum of two horizontal ZPAs is determined by the square root of the sum of the squares of each horizontal ZPA. For a horizontal ZPA of 0.15g the vectorial sum is 0.212g. Vertical ground motion corresponds to a maximum vertical acceleration of 0.10g. These values of 0.212g and 0.10g, respectively, are within the allowable values for horizontal and vertical seismic accelerations to ensure no sliding or tipping. The inequality is satisfied, as follows .

$$GH + \mu\, GV \leq \mu$$
$$0.212. + 0.53\ (0.10) \leq 0.53$$
$$0.265 \leq 0.53$$

Cask Sliding Evaluation

An analysis was performed to determine the maximum displacement of any HI-STORM 100 Cask during and earthquake considering icy conditions on the ISFSI pad. This analysis demonstrated the maximum displacement is smaller than the free space between the adjacent casks and between the outer casks and the edges of the ISFSI pad.

#### E.4.3.2.4 Section 3.4.4-Flood

*The analyzed flood condition of 15-fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.*

The HI-STORM overpack is analyzed for flood effects as shown in HI-STORM FSAR table 2.2.8. The analyzed submergence depth for the MPC is 125 feet, consistent with its dual-purpose function as part of a Part 71 –certified transport system. The analyzed flood velocity for the HI-STORM 100 System is 15 ft/s.

The elevation of the ISFSI pad is approximately 90 feet above Mean Sea Level (MSL).

The Indian Point 2 FSAR, Section 2.5, discusses various flood scenarios including hurricane, maximum probable precipitation and upstream dam failure. The referenced reports identify that the combination of probable maximum hurricane, spring high tide, and wave run-up will cause water level at Indian Point to reach 14.5 ft above MSL.

The potential for flooding at the ISFSI is nonexistent. In addition the construction of the ISFSI and the topography of the surrounding land preclude the possibility of ponding in the immediate area of the ISFSI pad.

**E.4.3.2.5 Section 3.4.5-Fire and Explosion**

*The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.*

The HI-STORM FSAR postulated Fire Event for the Overpack was performed using the following key inputs, as described in the HI-STORM FSAR Section 11.2.4.2.1:

1) A diesel fuel volume of 50 gallons maximum.
2) The HI-STORM overpack engulfed in flame for 3.622 minutes, and
3) A flame temperature of 1475$^0$F

The generic overpack fire analysis shows that the fuel cladding temperature, MPC internal pressure, and overpack outer shell steel temperature all remain below their respective short term temperature limits.

The HI-STORM FSAR postulated Fire Event for the HI-TRAC transfer cask was performed using the following key inputs, as described in the HI-STORM FSAR Section 11.2.4.2.2:

1) A diesel fuel volume of 50 gallons maximum.
2) The HI-TRAC transfer cask engulfed in flame for 4.775 minutes, and
3) flame temperature of 1475$^0$F

The generic HI-TRAC fire analysis shows that the fuel cladding temperature, MPC internal pressure, and overpack outer shell steel temperature all remain below their respective short term temperature limits.

The fire protection requirements for the design of the ISFSI are contained in 10 CFR 72.122(c ), Subpart F, General Design Criteria, *Protection against Fire and Explosions.* A review of the IPEC Fire Protection Program reveals that structures, Systems, and components important to safety are designed and located such that they can continue to perform their safety function effectively under credible fire exposure conditions.

There is no fixed fire suppression system at the ISFSI site, however the facility is located within the protected area of the plant such that the fire brigade can respond to any fire emergency. The ISFSI is included in the IPEC fire protection program and Fire Hazard Analysis. Vehicles and equipment powered by internal combustion engines are limited to a maximum of 50 gallons of fuel is they are brought within the boundaries of the ISFSI facility.

A Hazards Evaluation has been performed on the haul route between the Unit 2 FSB and the ISFSI. The Unit 1 fuel canisters are loaded into the HI-STORM in the Unit 2 FSB. As part of the evaluation fire sources and explosion hazards were examined. The complete report is contained in Appendix F, Unit 2 72.212 report.

A separate Hazards Evaluation has been performed for the short haul route between the Unit 1 FHB and the Unit 2 FSB (Reference: ENERCON Letter: R. Evers to P. Peloquin, RE-N06-037, August 4, 2006).

**Conclusion:**

The results of the evaluation concluded that all potential fire hazard exposures presented an acceptable risk. Some of the exposures were determined to be non-credible sources of fires during the limited time involved in cask transfer. Others were evaluated as being bounded by the design basis fire in terms of total energy content, and therefore acceptable

### E. 4.3.2.6 Section 3.4.6-Cask Drop and Tip-Over.

a. *For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to ≤ 45 g's at the top of the MPC fuel basket. Analysis shall be performed using methodologies consistent with those described in the HI-STORM FSAR.A lift height above the ISFSI pad is not required if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop features.*

b. *Section 3.4.6(b) addresses anchored cask systems and is not applicable to the IPEC ISFSI.*

The overpack will be lifted at the ISFSI by the VCT. The VCT load links, overhead beam, and load supporting members (whose failure result in a drop of the load) of the vertical frame meet the material selection and stress requirements of ANSI N14.6. The lifting towers (Jacks) comply with ASME B30.5-1994.

The design of the VCT complies with the requirements of Section 5.5.a.3 of Appendix A of the CoC during travel. For the short duration of time during the lowering and lifting of a load when the redundant drop protection features are not installed a site specific analysis has been performed as discussed in section E.4.2.3

IPEC procedure 2-DCS-006-GEN governs onsite cask and transfer cask transportation using the VCT.

**E.4.3.2.7 Section 3.4.7-Berm and Shield Walls**

*In cases where engineered features (i.e. ,berms and shield walls) are used to ensure that the requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.*

Berms or shield walls are not required or used at the IPEC ISFSI and are not credited in the shielding analysis performed to demonstrate compliance with 10 CFR 72.104(a).

**E.4.3.2.8 Section 3.4.8-Minimum Working Area Ambient Temperature**

*LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures* $\geq 0^0F$.

The following IPEC procedures restrict loading, transport and unloading operations to temperatures greater than or equal to $0^0F$.

2-DCS-006-GEN
2-DCS-008-GEN
2-DCS-009-GEN
2-DCS-012-GEN
1-DCS-028-GEN

**E.4.3.2.9 Section 3.4.9-Cask Air Duct Blockage for Extended Time**

*Or those users whose site-specific design basis includes an event or events (e.g., flood)that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e., longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel* *cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.*

There are no postulated site-specific design basis events that could potentially result in the blockage of any HI-STORM inlet or outlet air ducts for an extended period of time. (see Section E.4.3.2.4 of this appendix for more detail on flooding.)

### E.4.3.3 Section 3.6-Forced Helium Dehydration System

#### E.4.3.3.1 Sections 3.6.1 and 3.6.2-System Description and Design Criteria

*Use of the Forced Helium Dehydration System (FHD), a closed loop system, is an alternative to vacuum drying the MPC for moderate burnup fuel (≤ 45,000 MWD/MTU)and mandatory for drying MPC's containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shut down ramps) in accordance with the criteria in Section 3.6.2.*

The IPEC Unit 1 fuel to be loaded in the HI-STORM 100 System and stored at the IPEC ISFSI is all burned less than 27,100 MWD/MTU and is not high burnup fuel. The HI-STORM CoC Appendix A LCO 3.1.1 allows the use of either vacuum drying or FHD to dry MPC's containing all moderate burnup fuel. Entergy is choosing to use the FHD system to dry the Unit 1 MPC's currently planned for storage at the ISFSI. The FHD system used at Unit 1 is designed in accordance with design criteria in Section 3.6.2 of Appendix B of the HI-STORM CoC.

The acceptance criterion for the FHD system is gas temperature exiting the demoisturizer shall be ≤ $21^0$F for ≥ 30 minutes or a gas dew point exiting the MPC shall be ≤ $22.9^0$F for ≥ 30 minutes. A dew point temperature of $22.9^0$F or less is equivalent to a vapor pressure of 3.0 torr or less. Operation of the FHD system is governed by IPEC procedure 0-DCS-023-GEN.

#### E.4.3.3.2 Section 3.6.3-Fuel Cladding Temperature

*A steady state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in the HI-STORM100 FSAR section 4.4, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation is below* *the peak cladding temperature limit of normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.*

In accordance with the HI-STORM 100 CoC and FSAR, an analysis (Holtec document HI-2022966) was performed demonstrating compliance with all design criteria in FSAR Chapter 2 and the requirements of the HI-STORM CoC, sections 3.6.2 and 3.6.3. The FHD system was shown to satisfy all the design criteria requirements therefore the FHD design is in compliance with the CoC requirements.

### E.4.3.3.3 Section 3.6.4-Pressure Monitoring During FHD Malfunction

*During a FHD malfunction event, described in the HI-STORM 100 FSAR Section 11.1 as a loss of helium circulation, the system pressure must be monitored to ensure that the conditions listed therein are met.*

The FHD System is equipped with pressure gauges to ensure that this requirement is met and also with safety relief devices to prevent the MPC structural boundary pressures from exceeding the design limits. The MPC is filled with sufficient helium to maintain the fuel in an analyzed condition while actions are taken to return the FHD to service. When the FHD is operable, the MPC helium pressure may be reduced to allow the FHD to operate.

### E.4.3.4 Section 3.7-Supplemental Cooling System (SCS)

*The SCS is a water circulation system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. Use of the SCS is required for post-backfill HI-TRAC operations of a MPC containing one or more high burnup (> 45,000 MWD/MTU) fuel assemblies. The SCS shall be designed for normal operation( excluding startup and shutdown ramps) in accordance with the criteria in Section 3.7.2.*

Indian Point Unit 1 has no fuel classified as high burnup. The maximum burnup of the 14 X 14E fuel is less than 27,100 MWD/MTU. The use of the SCS is not required for the loading of the 160 assemblies.

### E.4.3.5 Section 3.8-Combustible Gas Monitoring During MPC Lid Welding

*During MPC lid welding operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present in the welding area.*

A risk of hydrogen production and a flammable atmosphere could exist inside the MPC due to oxidation of neutron absorber panels while the MPC is filled with water. Upon MPC lid installation, any gas generated would be trapped in the gas space under the lid created when the MPC water level is lowered to facilitate welding. Purging of the space under the lid is performed prior to pre-heating, welding or grinding operations per IPEC Procedure 1-DCS-028-GEN. Continuous sampling for combustible gas buildup is performed until the welding of the MPC lid-to shell weld, including NDE, is complete. Continuous sampling is also maintained during any repairs to the welds.

During unloading operations, sampling of the MPC internal atmosphere occurs prior to penetration to the cask internals in the unloading sequence per IPEC Procedure 1-DCS-035-GEN. The weld cutting process is not expected to be an ignition source due to low temperatures and lack of sparking. The cask will be vented during the refill sequence, and any gases in the cask should be expelled from the cask with the introduction of water. With helium in the cask at the beginning of cutting evolutions, a hydrogen burn cannot occur due to the lack of oxygen to initiate and sustain the burn.

## SECTION E.5
## COMPLIANCE WITH HI-STORM 100 CASK SYSTEM SER AND FINAL SAFETY ANALYSIS REPORT

*10 CFR 72.212(b)(3): Review the Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report (SER), prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(2) of this section.*

The following documents Entergy's review of the NRC SER through CoC Amendment 4 and HI-STORM FSAR Revision 4. Any divergence from the SER or FSAR descriptions, methodologies or practices is identified. All described deviations have been evaluated under the IPEC 10 CFR 72.48 process as applicable. Changes made by Holtec generically or site-specifically for IPEC are discussed in Section E.6. The cutoff date for generic changes to the FSAR as they relate to the IPEC ISFSI and this report is January 8, 2008.

### E.5.1 SER and FSAR Chapter 1, General Description

| SER FOR COC REVISION 0 | | |
|---|---|---|
| SECTION | REQUIREMENT | CHANGE DISCUSSION |
| 8.1.3 | Hydrostatic Test | The SER states that the MPC is backfilled with helium on top of the spent fuel pool water for applicable leak testing and then filled with water for the hydrostatic test. At IPEC Unit 1, the hydrostatic test will be performed, the cask drained, dried, and backfilled with helium prior to the helium leak test of the vent and drain port cover plates as described in the SAR review section 8.1.5. |
| **SER FOR COC REVISION 1** | | |
| SECTION | REQUIREMENT | CHANGE DISCUSSION |
| N/A | | No deviations or discussion required |
| **SER FOR COC REVISION 2** | | |
| SECTION | REQUIREMENT | CHANGE DISCUSSION |
| 6.3.2 | Licensee must perform test on Metamic with $B_4C$ | IPEC will not be performing tests on METAMIC. The CoC holder, Holtec, is |

| | | |
|---|---|---|
| | concentration above 15% prior to use. | required to perform these tests |
| **FSAR CHAPTER 1** | | |
| SECTION | REQUIREMENT | CHANGE DISCUSSION |
| NA | | No Deviations or Discussion Required |

### E.5.2 FSAR Chapter 2, Principle Design Criteria

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.3 FSAR Chapter 3, Structural Evaluation

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.4 FSAR Chapter 4, Thermal Evaluation

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.5 FSAR Chapter 5, Shielding Evaluation

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.6 FSAR Chapter 6, Criticality

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.7 FSAR Chapter 7, Confinement

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No Deviations or Discussion required |

### E.5.8 FSAR Chapter 8, Operating Procedures

| Section | Requirement | Change Discussion |
|---|---|---|
| 8.0 | User-developed procedures and the design and operation of any alternate equipment must be reviewed by the certificate holder prior to implementation. | Entergy-developed procedures encompassing loading, storage, and unloading operations have been reviewed by Holtec prior to implementation. Holtec review of user-generated revisions to procedures does not provide increased assurance of compliance with the CoC, safety in loading, unloading, or storing the cask or avoiding deviations from the intent of the FSAR. Therefore, Holtec review of revisions to these procedures is not considered necessary as long as the intent of the guidance in FSAR Chapter 8 and the CoC are met. |
| Table 8.0.1 | States that lifting devices are designed to ANSI N14.6 | Certain lifting items are not designed to ANSI N14.6. This standard only applies to special lifting devices. Generally speaking the intent of NUREG-0612 is met with regard to the design of lifting devices. The NUREG refers to a number of other codes and standards. Other lifting devices such as slings are designed with the codes and standards applicable to the device, e.g., ASME B30.9. Slings may be chosen to have enhanced safety factors per NUREG-0612 depending upon the nature of the load being lifted and the carry path. |

| Section | Requirement | Change Discussion |
|---|---|---|
| 8.1.1 | In this section and others, reference is made to MPC loading operations | At IPEC Unit 1, all MPC loading operations including use of the HI-TRAC 100D version IP1 are conducted in the Cask Loading Pool |

| | | |
|---|---|---|
| | conducted in the spent fuel pool. | which is separate from the West Fuel Pool. No fuel is stored in the Cask Loading Pool and assemblies are transported from the West Fuel Pool to the Cask Loading Pool through gates and an intermediate Disassembly Pool. The Cask Loading Pool, the Disassembly Pool, and a third unused Damaged Fuel Pool are sometimes collectively referred to as the "North Pools" |
| 8.1.5 | MPC Drain Down Time | IPEC requested from Holtec and received a Unit 1 HI-TRAC 100D-IP1/MPC-32 specific time to boil calculation. (Reference Holtec Document 1535-TH-1) This table provides time allowable durations based on initial water temperatures. Because of the long cooling time and low burnup of all Unit 1 fuel (over 32 years and less than 30,000 MWD/MTU), IPEC will calculate time to initiate forced water circulation using the formula $t_{max}$ = 6.07 (212-$T_i$ )/kW . The formula is based on the Holtec calculated combined thermal inertia of the loaded HI-TRAC described in Document 1535-TH-1.<br>$t_{max}$ = maximum time (Hours)<br>$T_i$ = initial water temperature ($^0$F)<br>kW = Cask thermal loading<br>6.07= Thermal Inertia factor for HI-TRAC 100D-IP1 converted to kW |
| Section | Requirement | Change Discussion |
| Table 8.1.6 | Water filled temporary shield ring | Holtec FSAR Table 8.1.6 and other references to the temporary shield ring in the FSAR describe the temporary shield ring as a water-filled tank that fits on top of the HI-TRAC water jacket around the upper forging. In lieu of a water filled tank IPEC is using a segmented poly block ring supplied by Holtec Per drawing 4348. IPEC may also use addition lead blanket shielding to reduce occupational exposure to a minimum. |

## E.5.9 FSAR Chapter 9, Acceptance Criteria and Maintenance Program

| Section | Requirement | Change Discussion |
|---|---|---|
| 9.1.1.4 | Inspection plan reviewed and approved by Holtec | The requirement for review and approval of NDE plans is aimed at welding performed in |

| | | |
|---|---|---|
| | | the shop during fabrication and is appropriate. The inspection of MPC closure welds performed at IPEC are in accordance with applicable Holtec drawings, the requirements of the FSAR including qualification of inspectors to SNT-TC-1A and the requirements identified in FSAR Table 9.1.4. Additional review by Holtec of IPEC NDE procedures is not considered necessary. |
| 9.2.1 | Perform load test on the HI-TRAC trunnions annually. | It is anticipated that IPEC Unit 1` will complete its loading campaign in less than 12 months. If a period of greater than 12 months of use should be anticipated, load testing or the optional dimensional and NDE examination shall be performed in accordance with the requirements of ANSI N14.6 |

**E.5.10 FSAR Chapter 10, Radiation Protection**

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No deviations or discussion required |

**E.5.11 FSAR Chapter 11, Accident Analysis**

| Section | Requirement | Change Discussion |
|---|---|---|
| 11.2.1.4, 11.2.2.4, 11.2.3.4 | Special handling procedures will be developed by the ISSI operator to upright the HI-TRAC/overpack after a handling or tip over event | The FSAR states that "special" procedures will be developed. "Special" meaning tailored to the as found condition after the event. IPEC Health Physics will perform a radiological assessment of the area in the event of a tip-over accident and take appropriate actions in accordance with site procedures. IPEC will then assess the damage and develop a special handling procedure to upright the HI-TRAC or overpack taking into account the radiological and environmental conditions. |

**E.5.12 FSAR Chapter 12, Operating Controls And Limits**

| Section | Requirement | Change Discussion |
|---|---|---|
| 12.2.2 | Dry Run Training on Receipt Inspection of Cask Components | This requirement does not appear in Condition 10 of the CoC. Also there are several phases of receipt inspection. A site inspection is |

| | | |
|---|---|---|
| | | performed for shipping damage. Another inspection is checking for pre-use cleaning and equipment check out.<br>All of these functions are standard practice under Entergy's 10 CFR 50 Appendix B Quality Assurance Program. |

### E.5.13 FSAR Chapter 13, Quality Assurance

| Section | Requirement | Change Discussion |
|---|---|---|
| NA | | No deviations or discussion required |

## SECTION E.6
## 72.48 REVIEWS & OUTSTANDING
## CASK LICENSING BASIS DOCUMENT CHANGES

The primary licensing documents of record used in this 72.212 evaluation report are the HI-STORM CoC Amendment 4, issued on January 8, 2008 , and Revision 6 of the HI-STORM 100 System FSAR, issued February 7, 2008.

### F.6.A HOLTEC GENERIC HARDWARE CHANGES

None affecting the design of Unit 1 equipment.

### F.6.B HOLTEC GENERIC CHANGES TO FSAR

| FSAR CHANGE INDEX FILTER RESULTS | | | | |
|---|---|---|---|---|
| **Chapter Number** | **FSAR Section** | **Holtec 72.48 No.** | **Affected Revision** | **Notes** |
| **1** | **Table 1.0.1** | **872** | **6.00** | **Modified 2/14/2008** |
| **2** | **2.1.3** | **872** | **6.00** | **Modified 2/14/2008** |
| **2** | **2.2.3.4** | **872** | **6.00** | **Modified 2/14/2008** |
| **3** | **3.4.4.3.1.9** | **872** | **6.00** | **Modified 2/14/2008** |
| **5** | **5.0 2nd Para.** | **872** | **6.00** | **Modified 2/14/2008** |
| **6** | **6.2.4** | **872** | **6.00** | **Modified 2/14/2008** |
| **6** | **6.4.2.4** | **872** | **6.00** | **Modified 2/14/2008** |
| **6** | **6.4.4** | **872** | **6.00** | **Modified 2/14/2008** |
| **7** | **7.1.5 2nd Para.** | **872** | **6.00** | **Modified 2/14/2008** |
| **8** | **8.0 8th Para.** | **872** | **6.00** | **Modified 2/14/2008** |
| | | | | |

## F.6.C SITE-SPECIFIC IPEC HARDWARE CHANGES

None applicable to Unit 1

**ENTERGY NUCLEAR**
**10 CFR 72.212 EVALUATION REPORT**
**SITE SPECIFIC APPENDIX F**
**REVISION 2**
**Indian Point Energy Center Unit 2**
**Safety Related: No**

Prepared by: John D. McCann / [signature] / DFS Project / 12/20/2007
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Reviewed by: John Janicki / [signature] / MTC / 12/20/2007
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Approved by: Michael J. Rutkoske / [signature] / Project Management 12/20/2007
Print Name/Signature/Department/Date

## Change Statement

**Revision 1 incorporates the following changes:**

1. Page F-1, Cover and Signature Page, added Revision Number and deleted acceptance signatures by department heads and plant manager. These signatures are not required per EN-LI-115 but were requested for Revision 0 by project management.

2. Page F-14, Hydrogen Gas Ignition during MPC Lid Welding and Cutting, and Page F-57, Section 3.8, changed "less than 10% of the lower explosive limit" to "less than 50%" to be consistent with the welding procedures.

3. Page F-24, 72.122(f), Testing and Maintenance of systems and components, deleted statement that no in-situ periodic inspection were required and added the statement that testing and inspection is to be in accordance with FSAR Table 9.2.1. The statement was added that annual load testing of Trunnions was not required.

4. Page 39, Condition 5, added words to indicate the site specific analysis is limited to the Low Profile Transporter and that the maximum lift height of the HI-STORM is 11" in accordance with Technical Specification 5.5.a

5. Page F-53, Section 3.4.5, Fire and Explosion, added a reference to NUREG-1864 discussing fire from an airplane crash.

6. Page F-65, FSAR Chapter 9.2.1, added a cross reference to the OSRC meeting approving the 72.48 evaluation deleting the requirement for annual load testing of the HI-TRAC trunnions.

7. Page F-68, Section F.6.C, Site Specific Hardware Changes, added discussion on the modification of the HI-TRAC top lid and grinding of the welds on the Upper Flange

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## SECTION F.1
## GENERAL INFORMATION

### F.1.1 Introduction

This document is prepared pursuant to 10 CFR 72.212(b) to facilitate the review and approvals required to utilize the Holtec HI-STORM 100 dry fuel storage system at the Indian Point Energy Center (IPEC) Unit 2 Independent Spent Fuel Storage Installation (ISFSI). Indian Point Unit 2 is owned and operated by Entergy Nuclear Northeast. Before this, all spent fuel has been stored under water in a pool in the Fuel Storage Building (FSB). Continued storage of spent fuel will be necessary until the Department of Energy (DOE) begins to accept fuel for permanent disposal as required by the Nuclear Waste Policy Act of 1982 and the contract for IPEC Unit 2 signed by the previous owner-operator, Consolidated Edison, on June 17, 1983. Under the existing provisions of the contract, the DOE would not have started accepting fuel from IPEC Unit 2 prior to the year 2005. IPEC Unit 2 lost full core offload capability in its spent fuel pit subsequent to 2R17, 2006.

As of 2006, DOE is still not accepting spent fuel. Thus, additional spent fuel storage capacity for IPEC Unit 2 is necessary. The Holtec HI-STORM 100 System was selected for use at IPEC Unit 2 and is designed to contain up to 32 PWR fuel assemblies in each storage cask. The HI-STORM 100 System consists of a stainless steel multi-purpose canister (MPC) with a welded base plate and lid, which is placed inside a coated carbon steel and concrete overpack that is placed on a concrete pad for storage. The ISFSI pad is designed to store up to 2400 IPEC spent fuel assemblies in 75 casks. Approximately 1280 of these fuel assemblies in 40 casks are expected to be from Unit 2; 960 of these assemblies in 30 casks from Unit 3; and 160 assemblies in 5 casks from Unit 1.

The Holtec HI-STORM 100 System is designed, licensed, fabricated, and deployed on site under the regulations in 10 CFR 72. The 10 CFR 72 regulations grant a general license for spent fuel storage in an NRC-certified dry storage cask system to anyone who holds a 10 CFR 50 license. The HI-STORM 100 System was originally certified by the NRC in accordance with 10 CFR 72, Subpart L in May 2000. The HI-STORM 100 Certificate of Compliance (CoC) 72-1014 has been amended three times since that time. IPEC Unit 2 will conduct its first fuel loading campaign in accordance with Amendment 2 to this CoC. (Amendment 3 was issued on May 29, 2007 but is not applicable to IPEC's 2007/2008 loading campaign.) A system-wide 10 CFR 72.212 evaluation report is maintained for Entergy[1] plants that utilize the HI-STORM 100 cask system. The specific conditions for use of the HI-STORM 100 System at IPEC Unit 2 are addressed in this appendix. Unit 3 will also be loading fuel assemblies at some future date and will share the same ISFSI facility with Unit 2. As plans for the Unit 3 loading effort are made, information specific to the Unit 3 effort will be incorporated into this appendix.

The IPEC Unit 2 site implementation review can be divided into four main components: loading and unloading, onsite transportation to the ISFSI, normal storage at the ISFSI, and off-normal and accident conditions. Each of these areas is discussed in this appendix to provide a background for the required reviews. Other 10 CFR 72.212 requirements are documented as appropriate in clearly identified appendix sections.

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[1] The HI-STORM System is used for dry cask storage at plants operated by both Entergy Operations and Entergy Nuclear Northeast. "Entergy" is therefore used throughout this report to mean Entergy Operations and/or Entergy Nuclear Northeast.

### F.1.2 System Components

The Holtec International HI-STORM 100 System for dry spent nuclear fuel storage consists of these major components or groups of components:

1. A multi-purpose canister (MPC) that contains the fuel (Important to Safety ITS-A) The PWR MPC-32 model is used at IPEC Unit 2.

2. A transfer cask (HI-TRAC) (Important to Safety ITS-B) that is used to move the HI-TRAC/MPC assemblage containing fuel from the spent fuel pit to the preparation area and ultimately to the steel and concrete overpack (HI-STORM).

3. The steel and concrete overpack (HI-STORM (Important to Safety ITS-B) that provides natural ventilation heat removal, radiation shielding, and structural protection for the MPC during storage operations.

4. A mating device (Important to Safety) containing a slide assembly used to mate the HI-STORM and HI-TRAC, remove the HI-TRAC pool lid, and allow transfer of the loaded MPC from the HI-TRAC to the HI-STORM.

5. A vertical cask transporter (VCT) (Not Important to Safety) used to transport the loaded HI-STORM overpack from outside the FSB to the ISFSI pad.

6. Ancillaries consisting of: 1) a forced helium dehydration (FHD) system (Not Important to Safety) including a chiller and various pumps, valves, pressure indicators and hoses mounted on a skid to facilitate preparing the MPC for storage operations, and 2) a mass spectrometer helium leak detection system (Not Important to Safety) for field leak testing the MPC vent and drain port cover plates.

7. The ISFSI concrete storage pad (Not Important to Safety) on which the loaded overpacks (HI-STORMs) are placed for long-term storage operations. (Note that the IPEC ISFSI pad has been constructed in accordance with Safety Related criteria)

Existing major plant equipment used for dry fuel storage includes the single-failure-proof gantry crane (a main hoist with 110-ton load capacity and an auxiliary (transfer) hoist with 45-ton load capacity), fuel handling manipulator crane (Important to Safety) and the site heavy-haul transport route between the FSB and the ISFSI pad (Not Important to Safety).

### F.1.3 IPEC Unit 2 Cask Loading Sequence

The general sequence of events for moving spent fuel from the Unit 2 spent fuel pit to the ISFSI begins with typical preparation of the components for loading activities, including inspection, cleaning, and fit-up. The general loading activities to be accomplished are described below. This list is intended to provide a basic understanding of the cask loading sequence, but should not be interpreted as the complete list of required steps or the exact sequence of steps. Detailed procedures are used to implement cask loading activities. The procedures address the required steps and the sequence of steps, and include appropriate limits and precautions.

1. The empty HI-TRAC transfer cask and MPC are moved into the FSB through the FSB doors using a low profile transporter (LPT) and a push-pull tugger. The gantry crane hoist is used to lift and insert the MPC into the empty HI-TRAC. The crane main hoist has a permanently installed transfer cask lifting yoke attached to the hook block. Initial MPC preparation activities are completed in the dry environment of the cask work area. Prior to moving the MPC to the spent fuel pit, it is partially filled with borated water to prevent annulus water pressure from lifting the MPC and to avoid splash-back of spent fuel pit water.

2. As necessary, the spent fuel pit water level is lowered sufficiently such that the HI-TRAC containing an empty MPC can be lowered to rest on the spent fuel pit floor in the cask loading area without spillage of water over the spent fuel pit curbs due to the water volume displacement caused by the cask.

3. The HI-TRAC/MPC assemblage is lifted vertically out of the cask work area using the gantry crane main hoist and placed on the spent fuel pit floor in the cask loading area.

4. The main hoist is disengaged from the HI-TRAC lifting trunnions.

5. 32 IPEC fuel assemblies meeting the limits in the HI-STORM 100 System CoC are transferred into the open MPC using the electric monorail hoist located on the spent fuel pit bridge.

6. The gantry crane transfer hoist is used to move the MPC lid, with its drain line installed, to a position over the open MPC and then lowers the lid into place on the MPC. The MPC lid provides sufficient shielding to raise the HI-TRAC/MPC out of the spent fuel pit.

7. The gantry crane main hoist hook is then engaged on the HI-TRAC lifting trunnions. The HI-TRAC/MPC is then lifted vertically from the spent fuel pit. The surfaces of the HI-TRAC are sprayed with primary water as it is removed.

8. After the HI-TRAC is clear of the spent fuel pit, it is moved laterally by the gantry crane and lowered into the cask work area.

9. The main hoist is disengaged from the HI-TRAC trunnions.

10. Scaffolding (a work platform) is installed around the HI-TRAC and the HI-TRAC surface above the top of the scaffold, the MPC lid, and the shell area above the annulus shield are decontaminated.

11. The annulus water level is lowered and the annulus seal is deflated and removed. The annulus seal area is decontaminated.

12. If used, the temporary shield ring is installed and filled with demineralized water. The annulus shield is installed.

13. All procedural requirements are completed to support MPC lid welding, draining, drying, and backfilling.

14. A small amount of water is drained from the MPC to create a small space between the water and the underside of the MPC lid. A vacuum or inert gas purge source is connected to the MPC vent port connection and left in operation to provide for combustible gas control and moisture removal during MPC lid welding. The MPC lid is welded in place, and dye penetrant non destructive examinations (NDE) and MPC pressure testing are performed in accordance with the HI-STORM 100 System CoC and FSAR, and site Dry Cask Storage (DCS) procedures.

15. The MPC is drained, dried using the FHD system, and helium filled in accordance with the HI-STORM 100 System CoC (LCO 3.1.1), HI-STORM 100 FSAR, and site DCS procedures. The vent and drain port covers are welded in place and helium leak tested. Finally, the MPC closure ring is welded in place and NDE inspections on these welds are performed. At this time the fuel is totally confined within the MPC.

16. The MPC lift cleats and the HI-TRAC top lid are installed. The gantry crane main hoist is used to lift the HI-TRAC/MPC straight up. Before the HI-TRAC/MPC is lifted completely from the cask work area, the exterior of the HI-TRAC is decontaminated. The inner shell of the HI-TRAC and most of the outer shell of the MPC should be clean due to the clean water in the HI-TRAC annulus during the loading and welding sequence. The upper annulus region receives additional verification of cleanliness by Radiation Protection personnel prior to moving the HI-TRAC from the cask work area.

17. Following the safe load path in the FSB, the gantry crane is used to move the HI-TRAC to a point above the HI-STORM location in the FSB. The HI-STORM overpack will have previously been staged in the FSB on the low profile cart with the mating device secured to the top of the overpack body by bolted connections Shims are installed between the low profile transport cart and the floor.

18. The HI-TRAC/MPC is lowered by the gantry crane and placed on top of the mating device. The HI-TRAC is secured to the mating device by clamp bolts.

19. The MPC lifting device is connected to the MPC lift cleats, allowing the MPC to be lifted slightly.

20. Bolting that attaches the pool lid to the HI-TRAC is removed and the pool lid is lowered into the mating device drawer. The hydraulic slide in the mating device is actuated, allowing the pool lid of the HI-TRAC to be withdrawn, creating a transfer path for the MPC into the overpack.

21. The MPC is slowly lowered into the overpack using the MPC lifting device which is then released. The gantry crane main hoist then used to move the empty HI-TRAC off of the mating device and place it in a storage location. The mating device remains bolted to the HI-STORM to provide shielding during MPC lift cleat removal.

22. MPC lift cleats and the mating device are removed and placed in designated storage locations in the FSB.

23. The shims between the low profile cart and the floor are removed, the FSB rollup door opened, and the tugger connected to the HI-STORM/LPT. The HI-STORM (without a top lid) is moved on the LPT approximately 100 feet to an area east of the overhead passageway for lid installation.

24. The HI-STORM overpack lid is installed using the vertical cask transporter (VCT) or a mobile crane. The VCT is moved back into position over the HI-STORM and the lift brackets are connected to the HI-STORM.

25. The HI-STORM is lifted the minimum necessary to clear the LPT (about 11 inches above grade). The redundant locking pins are installed. The LPT is removed.

26 The HI-STORM is transported by the VCT on the designated transport route to the ISFSI pad and placed in its storage location. The HI-STORM lift brackets are disconnected from the overpack and moved away by the VCT. The overpack lid is secured in place for long-term storage operations.

**F.1.4 IPEC Unit 2 Cask Unloading Sequence**

Although unlikely to be necessary, provisions are in place to transport the cask back to the FSB and unload fuel from the MPC due to some unforeseen event. There are no credible events related to onsite ISFSI operations that would require cask unloading, or that would damage the overpack, transfer cask, or an MPC such that the fuel cannot be recovered in the spent fuel pit if cask unloading is necessary. In the case of postulated drops of a loaded HI-TRAC or HI-STORM, the MPC is designed to remain intact and able to be transported back to the spent fuel pit and unloaded.

Generally speaking, recovery of the loaded MPC up to the point of removing the MPC lid is the reverse of the loading sequence with certain additional considerations. Steps that are the reverse of loading operations are not repeated here. The following additional steps would be implemented to remove the fuel and return it to the spent fuel pit storage racks:

1. Once it is decided that an MPC needs to be unloaded, the overpack is moved back to the FSB using the VCT, the LPT, and the push-pull tugger. The mating device and HI-TRAC are installed and the MPC is transferred into the HI-TRAC. The HI-TRAC bottom lid is installed while on the mating device.

2. The HI-TRAC/MPC is moved back to the cask work area.

3. The vent and drain port covers are removed so that the Remote Valve Operating Assemblies (RVOAs)s .

4. A gas sample is taken from the MPC to determine whether there is any failed fuel in the MPC.

5. Based on the gas sample results, the specific steps for unloading the cask will be determined. If samples indicate there is a significant amount of failed fuel, the cask will be vented using an appropriate radioactive waste vent path.

6. Borated water can be introduced into the MPC up to a level that is appropriate to allow lid weld cutting to proceed, in accordance with the pressure limits specified in Technical specification LCO 3.1.3. At this point, purging or exhausting of the space under the MPC lid must be commenced and the MPC lid-to-shell weld is removed. Upon completion of lid weld cutting, the HI-TRAC/MPC is moved to the spent fuel pit using the equipment and procedures from cask loading operations in reverse order.

7. Once the HI-TRAC/MPC is in place in the spent fuel pit, the MPC lid is removed and the fuel assemblies are returned to the spent fuel pit storage racks.

### F.1.5 IPEC Unit 2 Site Off-Normal Events

The events that could affect the plant due to fuel movement, fuel containment, onsite cask transport, or cask storage that are not addressed in the HI-STORM 100 System FSAR and require a site specific evaluation are discussed below.

**Cask Loading and Handling Operations**

A traveling single-failure-proof gantry crane with a design rated load capacity of 110 tons has been installed in the IPEC Unit 2 FSB. The crane was provided by Ederer LLC and is used primarily to move dry cask storage equipment into and out of the spent fuel pit. The crane design and associated handling equipment conform to the guidance in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," as applicable, for heavy load lifts over the spent fuel pit. The crane is capable of lifting the fully loaded HI-TRAC 100 transfer cask and associated components, but will not lift the Holtec HI-STORM 100 overpack or lid.

Entergy decided to install the single-failure-proof gantry crane because: 1) the existing 40-ton overhead crane did not have the load capacity to handle the HI-TRAC 100 transfer cask and 2) the existing 40-ton crane is not single-failure-proof. The 40-ton crane remains in place, however, as it is used for other load handling activities in the FSB.

Entergy submitted a license amendment request to the NRC on November 1, 2004, requesting review and approval of the Ederer single-failure-proof gantry crane for use in the Unit 2 FSB for moving spent fuel casks and related operations involving heavy loads. The NRC approved the LAR, which included supplemental information provided on April 12, July 22, and September 26, 2005, and issued license amendment No. 244 to the Unit 2 Part 50 operating license on November 21, 2005.

The FSB gantry crane has two hoists – a main hoist with a 110-ton capacity and an auxiliary (transfer) hoist with a 45-ton capacity. Both hoists meet the single-failure-proof guidance of NUREG-0554. The main hoist uses a gantry that can traverse a portion of the FSB truck bay and a cantilever girder-mounted trolley that extends to the spent fuel pit. The gantry crane by design is unable to move spent fuel casks over any area of the spent fuel pit where spent fuel is stored. Nonetheless, safe load paths have been determined, analyzed, and documented in procedures for control of heavy loads handled by the gantry crane to provide defense in depth.

The main hoist is equipped with an integral transfer cask lift yoke assembly which is permanently attached to the hook block.

Use of the single-failure-proof gantry crane to move dry cask storage equipment in the FSB eliminates the need to postulate and evaluate potential heavy load drops of dry cask storage equipment in the FSB, as such drops are not credible.

One potential drop involving spent fuel in the spent fuel pit is not eliminated by use of the gantry crane, as summarized below:

**Drop of a Fuel Assembly into an MPC**

Transfer of spent fuel assemblies from the fuel storage racks in the spent fuel pit to the MPC is accomplished by use of an electric monorail hoist located on the spent fuel pit bridge. The potential for drop of a fuel assembly into an MPC while the MPC is being loaded has been considered.

This accident is similar to the accident evaluated in Section 14.2.1.1 of the IPEC Unit 2 Part 50 UFSAR in which a fuel assembly is dropped in the spent fuel pit during refueling. The primary concerns addressed in this accident are the potential radiological consequences associated with gross damage to the dropped fuel assembly. The results of the analysis show that the radiological consequences associated with the underwater breach of all fuel rods in a single fuel assembly do not exceed acceptable limits. The conditions and configurations assumed in Section 14.2.1.1 of the IPEC Unit 2 UFSAR envelope dropping of a spent fuel assembly into the MPC, as the HI-TRAC/MPC can be viewed as a spent fuel storage rack with approximately the same fuel interface conditions.

**<u>Sealing and Cleanup</u>**

**High Radiological Dose During MPC Preparation**

Draining of the MPC involves the connection of hoses to the remote valve operating assemblies (RVOAs) that are connected to the vent and drain ports in the top of the MPC lid after the HI-TRAC/MPC is placed in the cask work area (a dry environment). Prior to welding the lid, a small amount of water is pumped from the MPC to ensure the water level is below the lid weld area. NDE is performed on the lid root weld and periodically thereafter until the weld is completed, as required by the HI-STORM 100 CoC and FSAR, and site DCS procedures. The MPC is pressure tested, the remaining water is removed from the MPC fuel cavity, and the FHD system is connected to the MPC. The MPC fuel cavity is dried until the acceptance criteria in the HI-STORM 100 CoC are met. The MPC is then backfilled with helium to the CoC-required pressure. Final sealing is accomplished by welding the vent and drain port cover plates to the lid and closure ring to the MPC lid and shell.

The primary concerns associated with these activities are dose rates to the individuals working around the cask (e.g., decontamination personnel, riggers, welders, etc.), discharge of water from the cask, and releases of gases (hydrogen or helium) from the cask. MPC preparation activities will be performed under the same administrative control procedures used at IPEC for other work in radiation controlled areas to ensure compliance with 10 CFR 20. Radiation surveys will be taken, as appropriate, and Radiation Work Permits will identify appropriate dose and dose rate limits, dosimetry, and protective clothing for the activities.

**Hydrogen Gas Ignition During MPC Lid Welding and Cutting**

Hydrogen gas production that may occur due to oxidation of the aluminum in the neutron absorber in the MPC fuel basket, or other phenomenon, will start with the introduction of spent fuel pit water and fuel into the MPC interior. Upon MPC lid installation, any gas generated potentially could be trapped under the lid. The next evolution in the loading sequence drains water from the cask interior via the drain line through the MPC lid. Draining water creates a gas space below the weld area underneath the lid to prevent quenching of the weld. Actions are taken to prevent the concentration of hydrogen potentially reaching flammability limits in the gas space. In accordance with the HI-STORM FSAR, prior to and during lid welding operations, the gas space under the lid is purged with an inert gas.

Continuous sampling of the MPC lid-to-shell weld gap is performed and the exhaust is monitored for combustible gas during welding operations until the MPC lid root weld layer, including NDE, is complete. Continuous sampling also includes all repairs to the root weld layer, if required. If welding of the root weld layer is interrupted for any reason, combustible gas concentrations are verified to be less than 50% of the lower explosive limit prior to continuing welding.

If the MPC is required to be unloaded, the fuel cavity will be flooded with borated water. The gas space under the MPC will be purged or exhausted prior to and during weld cutting operations in accordance with the HI-STORM FSAR. This will prevent any potential hydrogen ignition caused by the weld cutting operation. Also refer to the discussion in Section F.4.3.5 of this report.

**Transfer of the MPC**

After the MPC is successfully transferred from the HI-TRAC to the HI-STORM overpack, the overpack is transported with a crawler-type VCT to the ISFSI. The VCT is attached to the HI-STORM using lift brackets that are bolted into the overpack body through the lid. The cask is suspended above the ground only to those heights necessary to transport the cask to the ISFSI and place it in its designated storage location without impacting the transport route or any obstructions along the way. The VCT is equipped with locking pins which provide redundant drop protection features to the hydraulic lift mechanism. This feature precludes the possibility of a cask drop during transport to the ISFSI. Site specific analyses have been performed to determine the allowable lift height of the cask prior to and during the installation of the locking pins.

**Storage of the HI-STORM**

**Tornado, Flooding, and Earthquake**

**Tornado**

The tornado wind and missile criteria used for design of the HI-STORM 100 Cask System are provided in Tables 2.2.4 and 2.2.5 of the HI-STORM FSAR. The wind criteria are a maximum tornado wind speed of 360 mph (290 mph rotational and 70 mph translational) with a pressure drop of 3 psi. The following tornado-generated missiles were analyzed for the HI-STORM 100 System cask design, including both the HI-STORM overpack and the HI-TRAC transfer cask:

- 3960 pound (1800 kg) automobile at 126 mph
- 8-inch diameter, 275 pound (125 kg), rigid solid steel cylinder at 126 mph
- 1-inch diameter, 0.48 pound (0.22 kg), steel sphere at 126 mph

As discussed in the IPEC Unit 2 and Unit 3 UFSARs, the tornado parameters considered bounding for the IPEC site include a tornado with 300 mph tangential velocity, traverse velocity of 60 mph, and a differential pressure drop of 3 psi, and the following missiles:

- 4000 pound automobile at 50 mph
- 4" x 12" x 12' wood plank at 188 mph

Because the generically analyzed missiles for the HI-STORM 100 Cask System and the site-specific tornado missiles for IPEC are both based on Spectrum II missiles as defined in Section 3.5.1.4 of NUREG-0800, the generic HI-STORM 100 tornado missile analysis envelopes the site-specific tornado missiles.

**Flooding**

As described in Section 2.5 of the IPEC Unit 2 UFSAR, the highest recorded water elevation in the vicinity of the IPEC site was 7.4-feet above mean sea level (MSL), which occurred during an exceptionally severe hurricane in November 1950. Since the Hudson River water elevation would have to reach 15-feet 3-inches above MSL before it would seep into any of the IPEC buildings, the potential for flooding at the IPEC site is considered to be extremely remote.

Seven different hypothetical flooding conditions governing the maximum water level at the site were investigated. The most severe condition resulted from the simultaneous occurrence of a standard project flood, a failure of the Ashokan Dam, and a storm surge in New York Harbor at the mouth of the Hudson River resulting from a standard project hurricane. The water level under these conditions would reach 14-feet above MSL. Local wave action due to wind effects has been determined to add 1-foot to the river elevation producing a maximum water elevation of 15-feet above MSL at the IPEC site, still slightly below the critical elevation of 15-feet 3-inches.

Therefore, in view of the recorded hydrologic history and the most severe hypothetical flooding condition, flooding at the IPEC site that would affect plant operation or removal and transport of spent fuel to the ISFSI is considered highly unlikely.

The elevation of the ISFSI at the IPEC site is approximately 90 feet above MSL. As this is well above the 15-foot level expected for the most severe hypothetical flooding condition at the IPEC site, the potential for flooding at the ISFSI is considered to be nonexistent.

Drainage provisions and the slope of the pad at the ISFSI site also preclude the possibility of any significant ponding at or near the ISFSI.

## Earthquake

The HI-STORM 100 System CoC and FSAR include two seismic evaluation criteria. The first involves an algebraic inequality to be executed using the site's seismic design basis Zero Period Accelerations (ZPAs), along with a coefficient representing two different variables, namely 1) the Coulomb coefficient of friction between the pad and the cask, and 2) the ratio of the cask radius to the height of the center-of-gravity above the ISFSI pad surface. If the inequality is satisfied for both values of the coefficient, then the cask may be deployed in a free standing mode and the second seismic evaluation criteria (applicable to anchored casks) do not apply. If the inequality is not satisfied for both values of the coefficient using ZPAs, a time-history analysis may be performed to verify whether the cask may still meet the first seismic evaluation criteria and be deployed in a free standing configuration.

If neither the inequality nor the time-history analysis is successful in proving that the cask may be deployed in a free standing mode, the second seismic evaluation criteria must be met and the cask must be anchored to the ISFSI pad. An evaluation of the IPEC site-specific seismic design criteria has been performed and a determination made that the HI-STORM 100 System may be deployed at the IPEC ISFSI in a free standing mode. Therefore, the second seismic evaluation criteria are not applicable to the IPEC ISFSI. The comparison of the IPEC site-specific seismic design basis against the HI-STORM CoC/FSAR free standing seismic evaluation criteria is discussed in more detail in Section F.4.3.2.3 of this appendix.

## APPENDIX F.2A
## LIST OF SITE SPECIFIC PROCEDURES

In addition to the corporate procedures listed in Appendix A, and IPEC Site and ISFSI procedures, IP-2 requires the following procedures in order to address the specific fuel transport and storage activities at Indian Point Unit 2. (Some procedures are common to Unit 1 and some procedures are unique to Unit 1 to reflect the differences in facilities and equipment. Both Unit 1 and Unit 2 procedures are listed for completeness of the listing.)

1. MPC-32 Receipt, Handling, and Fit Up (2-DCS-001-GEN)
2. HI-TRAC100D-1P1 Receipt , Handling, and Initial Assembly (2-DCS-002-GEN)
3. HI-STORM100S-185 Receipt, Handling, and Fit Up ( 2-DCS-003-GEN)
4. RIGID Chain Drive System Operation (2-DCS-004-GEN )
5. Ancillary Pre-operational inspection and Functional Tests (2-DCS-005-GEN)
6. Vertical Cask Transporter Operation (2-DCS-006-GEN)
7. ISFSI Storm Water Pollution Prevention Inspection and Maintenance (2-DCS-007-GEN)
8. Unit 2 MPC Load and Seal (2-DCS-008-GEN)
9. MPC Transfer and HI-STORM Movement (2-DCS-009-GEN)
10. Ancillary Layup Procedure (2-DCS-010-GEN)
11. HI-STORM Inspection ( 0-DCS-011-GEN)
12. Unit 2 MPC Unloading Procedure (2-DCS-012-GEN)
13. Transfer Table Operations (2-DCS-013-GEN)

14. Unit 1 Fuel Handling (1-DCS-014-GEN)

15. (0-DCS-015-GEN Procedure number is available for future use)

16. DCSS Special Lifting Devices Inspection (2-DCS-016-GEN)

17. (2-DCS-017-GEN Procedure number is available for future use)

18. (2-DCS-018-GEN Procedure number is available for future use)

19. (2-DCS-019-GEN Procedure number is available for future use)

20. (2-DCS-020-GEN Procedure number is available for future use)

21. (1-DCS-021-GEN Procedure number is available for future use)

22. (2-DCS-022-GEN Procedure number is available for future use)

23. FHD Operations (0-DCS-023-GEN)

24. (1-DCS-024-GEN Procedure number is available for future use)

25. Air Pad Operations (1-DCS-025-GEN)

26. Unit 2 Crane Operations (2-DCS-026-GEN)

27. (2-DCS-027-GEN Procedure number is available for future use)

28. Unit 1 MPC Load and Seal (1-DCS-028-GEN)

29. (2-DCS-029-GEN Procedure number is available for future use)

30. Fuel Selection for Dry Cask Storage-Unit 1(1-DCS-030-GEN)

31. Fuel Selection for Dry Cask Storage-Unit 2 (2-DCS-031-GEN)

32. Dry Cask Loading Readiness Guidelines (2-DCS-032-GEN )

33. Abnormal Operations (2-DCS-033-GEN)

34. HI-TRAC Annual Inspection (2-DCS-034-GEN)

35. Unit 1 MPC Unloading Procedure (1-DCS-035-GEN)

36. Radiological Controls for Dry Cask Storage (0-RP-RWP-420)

## SECTION F.2.B
## IPEC KEY HOLTEC CASK SYSTEM DRAWINGS

| Drawing Series | Component |
|---|---|
| 4116 | HI-STORM 100S Version B Assembly |
| 3993 | MPC Enclosure Vessel |
| 3927 | MPC-32 Fuel Basket Assembly |
| 4128 | HI-TRAC 100 Transfer Cask |
| 2602 | HI-STORM Lift Bracket |
| 2507 | AWS Base Plate |
| 2511 | MPC Lift Cleat |
| 4150 | HI-STORM Mating Device |
| 4399 | Mating Device Spacer Ring for HI-STORM 100S |

## SECTION F.3
## COMPLIANCE WITH 10 CFR PART 72

General compliance with the requirements specified in 10 CFR 72 is discussed in Section V of the main body of this report. Certain regulatory requirements requiring site-specific discussion for IPEC are provided below.

### F.3.1 §72.106 - Controlled Area of the ISFSI

*(a) For each ISFSI or MRS site, a controlled area must be established.*

As defined by 10 CFR 72, the controlled area means that area immediately surrounding an ISFSI or MRS for which the licensee exercises authority over its use and within which ISFSI or MRS operations are performed. The IPEC ISFSI is located within the plant protected area at a location approximately 370 feet north of the Unit 2 Containment. The exclusion area boundary for Unit 2 is defined by a 520-meter (1706-feet) radius circle drawn about the Unit 2 reactor center. The ISFSI is within the boundary of the exclusion area which is entirely owned and controlled by Entergy.

*(b) Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent may not exceed 15 rem and the shallow dose equivalent to skin or any extremity may not exceed 50 rem. The minimum distance from the spent fuel, high-level radioactive waste, or reactor-related GTCC waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.*

Design basis accidents that may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. Because the damage is localized and the vast majority of the cask shielding material remains intact, the site boundary dose rates for the loaded HI-STORM overpack for accident conditions are the same as the normal condition dose rates. The dose versus distance from the HI-STORMS containing the MPC 32 is documented in Holtec Report HI-2073724, dated 7/30/2007. Therefore, the accident dose limits of 10 CFR 72.106 are met. As discussed in HI-STORM FSAR Chapter 7, there is no credible leakage from the confinement boundary during accident conditions based on ISG-18 (for the MPC lid-to-shell weld) and because the vent and drain port cover plate welds are field leak tested to a "leaktight" acceptance criterion in accordance with ANSI N14.5. Therefore, there is no effluent dose contribution to the calculated normal, off-normal, or accident offsite accident dose from the ISFSI.

*(c) The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety.*

The IPEC ISFSI (and associated controlled area) is completely contained within the plant exclusion area, which is entirely owned and controlled by Entergy. There are no public highways or railroads traversing the ISFSI controlled area. The Hudson River is located about 75-meters to the west of the ISFSI site, but this part of the river is well within the plant exclusion area and traffic on the river is subject to Entergy control in the event of an unexpected radiological emergency.

### F.3.2 §72.122 – Overall Requirements

*72.122(a): Quality Standards. Structures, systems, and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.*

Each component or subcomponent of the ISFSI and HI-STORM 100 System is classified as Important-to-Safety (ITS), Category A, B, or C, or Not-Important-to-Safety (NITS) in accordance with the guidance in NUREG/CR-6407. These classifications are made based on the design function of the component or subcomponent.

*72.122(b): Protection against environmental conditions and natural phenomena.*
*(1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI or MRS and to withstand postulated accidents.*

*(2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect:*

*(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect:*

*(A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and*

*(B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena.*

*(ii) The ISFSI or MRS also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel, high-level radioactive waste, or reactor-related GTCC waste or on to structures, systems, and components important to safety.*

*(3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.*

*(4) If the ISFSI or MRS is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.*

The cask system being deployed at the IPEC ISFSI under the general license provisions of 10 CFR 72 is the HI-STORM 100 System (CoC 1014 Amendment 2), which has been certified by the NRC and is listed in 10 CFR 72.214. The cask system has been designed and analyzed to withstand environmental conditions and natural phenomena as described in the HI-STORM 100 System FSAR. The generic design criteria for environmental conditions and natural phenomena used in the cask design were verified to be bounding for the site-specific design basis environmental phenomena applicable to the IPEC site. The balance of the ISFSI design has appropriately considered environmental conditions and natural phenomena as they apply to the particular structure, system, or component of the ISFSI. The details of these design considerations may be found in the applicable design control documentation for the ISFSI.

As a part of plant preparations for potential severe weather, IPEC Procedures OAP-008, "Severe Weather Preparations," and 2-DCS-008 and 9 require termination of any fuel handling or cask movement operations in progress. This action prevents movement of a loaded HI-STORM, without its permanent lid installed, out of the FSB during predicted high winds or tornado conditions, making the possibility of a tornado missile strike non-credible.

*72.122(c): <u>Protection against fires and explosions.</u> Structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI or MRS, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI or MRS must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.*

The HI-STORM 100 System has been designed for fires, explosive overpressure, and missiles. The IPEC ISFSI, including the transport route from the FSB to the ISFSI, has been evaluated to ensure that the generic design basis for the cask bounds the actual configuration at IPEC with respect to fire and explosion hazards (see Section F.4.3.2.5 for details). The IPEC fire protection plan and fire hazards analysis have been reviewed to ensure they envelope dry spent fuel storage operations at the site. A fire suppression system is not used at the IPEC ISFSI because there are no combustible materials stored at the ISFSI. Fires due to transient combustibles will be extinguished with portable fire suppression equipment.

*72.122(d): <u>Sharing of structures, systems, and components.</u> Structures, systems, and components important to safety must not be shared between an ISFSI or MRS and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions, including the ability to return to a safe condition in the event of an accident.*

The HI-STORM 100 System does not require electric power to perform its design functions. The cask system is a welded, canister-based system, passively cooled by a naturally ventilated overpack. There are no cask leakage monitoring systems. The inlet and outlet air ducts are visually inspected for blockage on a periodic basis. The ISFSI, including the cask system shares no structures, systems, or components important to safety with any other facility.

*72.122(e): <u>Proximity of sites.</u> An ISFSI or MRS located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.*

The IPEC ISFSI is co-located within the protected area of the IPEC Part 50 facilities. The cumulative effects of combined operations of the Part 50 facilities and the ISFSI will not constitute an unreasonable risk to the health and safety of the public. The additional direct radiation dose to the public from ISFSI operations is relatively small due to the distance between the ISFSI and the site boundary (see Section F.3.9). The HI-STORM 100 System is designed not to release any radioactive effluents under normal, off-normal, or accident conditions, so there is no additional effluent dose to the public from ISFSI operations.

*72.122(f): <u>Testing and maintenance of systems and components.</u> Systems and components that are important to safety must be designed to permit inspection, maintenance, and testing.*

No periodic maintenance of major ITS components, other than minor touch-up painting on the casks, is required at the ISFSI. The HI-STORM 100 System is completely passive in design, so there are no mechanical or electrical systems to maintain on the storage overpack or canister. Periodic inspections and testing requirements of the storage systems during storage operations are identified in Table 9.2.1 of the Holtec FSAR. (NOTE: Annual load testing of the HI-TRAC trunnions has been replaced by annual NDE of the trunnions. Refer to procedure 2-DCS-016-GEN) Surveillance during storage operations is limited to visual observations of the cask air inlet and outlet ducts. Periodic maintenance, inspection and testing of ITS ancillaries are performed in accordance with the Preventative Maintenance Process.

*72.122(g): <u>Emergency capability.</u> Structures, systems, and components important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.*

The operation of the ISFSI has been evaluated for its effects on the IPEC emergency response plan and modifications to the plan were made as necessary. The ISFSI is located within the plant's protected area and access is available through gates in the protected area. The same onsite and offsite emergency facilities as those used for the Part 50 facilities are used for events associated with ISFSI operations.

*72.122(h): Confinement barriers and systems. (1) The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.*

*(2) For underwater storage of spent fuel, high-level radioactive waste, or reactor-related GTCC waste in which the pool water serves as a shield and a confinement medium for radioactive materials, systems for maintaining water purity and the pool water level must be designed so that any abnormal operations or failure in those systems from any cause will not cause the water level to fall below safe limits. The design must preclude installations of drains, permanently connected systems, and other features that could, by abnormal operations or failure, cause a significant loss of water. Pool water level equipment must be provided to alarm*

*in a continuously manned location if the water level in the storage pools falls below a predetermined level.*

*(3) Ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.*

*(4) Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.*

*(5) The high-level radioactive waste and reactor-related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.*

The HI-STORM 100 System is a dry-type storage system with a redundant welded confinement barrier. The canister fuel cavity is backfilled with helium gas to promote effective decay heat removal and inhibit corrosion of the fuel cladding. The cask design has been certified with certain limits on the characteristics of the stored fuel to ensure adequate heat removal and protection of the fuel cladding. Damaged fuel is required to be stored in damaged fuel containers to ensure the damaged fuel remains retrievable and in a geometry that is bounded by the criticality analysis.

No monitoring of the canister confinement system is required because it is a welded system. Monitoring of the overpack is limited to periodic visual inspection of the air inlet and outlet ducts to ensure they are free of blockage and the overpack is able to transfer an adequate amount of heat from the MPC to the environs. Handling and retrievability is ensured in the cask system design, which includes a transfer cask with lead and water radiation shields to protect personnel and keep occupational exposures due to loading operations well below the limits in 10 CFR Part 20. Each HI-STORM overpack is equipped with Resistance Temperature Detectors on two of the inlet and two of the outlet vents to indicate continued air flow by indicating and measuring the temperature differential between the inlet and outlet vents.

*72.122(i): <u>Instrumentation and control systems.</u> Instrumentation and control systems for wet spent fuel and reactor-related GTCC waste storage must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report. Instrumentation systems for dry storage casks must be provided in accordance with cask design requirements to monitor conditions that are important to safety over anticipated ranges for normal conditions and off-normal conditions. Systems that are required under accident conditions must be identified in the Safety Analysis Report.*

The HI-STORM 100 System is a completely passive system that requires no instrumentation control systems. The RTDs installed on the vents of the HI-STORM overpack are for indication only and act as a backup to visual confirmation that blockage of the vents has not occurred.

*72.122(j): <u>Control room or control area.</u> A control room or control area, if appropriate for the ISFSI or MRS design, must be designed to permit occupancy and actions to be taken to monitor the ISFSI or MRS safely under normal conditions, and to provide safe control of the ISFSI or MRS under off-normal or accident conditions.*

The ISFSI is co-located in the protected area of the IPEC Part 50 facilities. The main control room of Unit 2 provides a location for IPEC to direct occupancy and actions to be taken in the event of an off-normal or accident condition at the ISFSI. No separate ISFSI control room or control area is provided.

*72.122(k): <u>Utility or other services.</u> (1) Each utility service system must be designed to meet emergency conditions. The design of utility services and distribution systems that are important to safety must include redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform safety functions assuming a single failure.*

*(2) Emergency utility services must be designed to permit testing of the functional operability and capacity, including the full operational sequence, of each system for transfer between normal and emergency supply sources; and to permit the operation of associated safety systems.*

*(3) Provisions must be made so that, in the event of a loss of the primary electric power source or circuit, reliable and timely emergency power will be provided to instruments, utility service systems, the central security alarm station, and operating systems, in amounts sufficient to allow safe storage conditions to be maintained and to permit continued functioning of all systems essential to safe storage.*

*(4) An ISFSI or MRS which is located on the site of another facility may share common utilities and services with such a facility and be physically connected with the other facility; however, the sharing of utilities and services or the physical connection must not significantly:*

*(i) Increase the probability or consequences of an accident or malfunction of components, structures, or systems that are important to safety; or*

*(ii) Reduce the margin of safety as defined in the basis for any technical specifications of either facility.*

The HI-STORM 100 System does not require electric power or any other utilities to perform its design functions. The cask system is a welded, canister-based system, passively cooled by a naturally ventilated overpack. There are no cask leakage monitoring systems. The inlet and outlet air ducts are visually inspected for blockage on a periodic basis. The ISFSI, including the cask system shares no structures, systems, or components important to safety with any other facility.

*72.122(l): <u>Retrievability.</u> Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal.*

The HI-STORM 100 System is designed for retrievability of the spent fuel as discussed in the FSAR for the cask system. The MPC is certified for transportation in the HI-STAR 100 System transport overpack to a disposal site without repackaging of the fuel at the IPEC site.

### F.3.3 §72.124 – Criteria for Nuclear Criticality Safety

*72.124(a): <u>Design for criticality safety.</u> Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.*

The HI-STORM 100 System has been analyzed for the most reactive credible conditions during spent fuel loading in the spent fuel pit, assuming fresh fuel and borated water and found to be safely subcritical. The criticality design of the cask is described in Chapter 6 of the HI-STORM 100 System FSAR and has been reviewed and approved by the NRC.

*72.124(b) <u>Methods of criticality control.</u> When practicable, the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.*

The HI-STORM 100 System incorporates the favorable geometry and fixed neutron absorber material of the MPC-32 fuel basket, as well as soluble boron in the spent fuel pit water for criticality control as discussed in Chapter 6 of the HI-STORM 100 System FSAR. There are no known degradation mechanisms for the fixed neutron absorbers in a helium environment over the life of the ISFSI. Therefore, positive means for verifying continued neutron absorber efficacy are not required.

*72.124(c): <u>Criticality Monitoring.</u> A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.*

During the time period when the special nuclear material is neither beneath water shielding nor packaged in its stored configuration (i.e., from the time the cask is removed from the cask pool; moved to the cask work area; and drained, dried, and backfilled with helium), criticality is considered to be not credible because the design features of the Holtec MPC-32 preclude accidental criticality when containing CoC-authorized fuel with borated water. There is no credible dilution mechanism for the borated water in the MPC and no credible MPC load drops that could lead to criticality in the MPC.

While a criticality accident is not considered credible, radiation monitors, as required by General Design Criterion (GDC) 63, are located in the fuel handling and storage areas of the FSB.

These monitors would alert personnel to excessive radiation levels associated with criticality in the MPC and allow them to initiate appropriate safety actions.

### F.3.4 §72.126 – Criteria for Radiological Protection

*72.126(a): Exposure control. Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to:*

*(1) Prevent the accumulation of radioactive material in those systems requiring access;*

*(2) Decontaminate those systems to which access is required;*

*(3) Control access to areas of potential contamination or high radiation within the ISFSI or MRS;*

*(4) Measure and control contamination of areas requiring access;*

*(5) Minimize the time required to perform work in the vicinity of radioactive components; for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement; and*

*(6) Shield personnel from radiation exposure.*

The HI-STORM system was designed to minimize exposure during the loading, unloading, and onsite transport and storage of the cask. Design aspects such as the use of a thick steel MPC lid for canister closure; the use of a shielded transfer cask for movement of the fuel-bearing canister; and the thick concrete overpack for storage are a few of the design methods employed.

Activities pertinent to the HI-TRAC and HI-STORM operation, including survey and decontamination, are controlled by site radiation protection procedures as well as the cask specific procedures for loading/unloading/transportation/storage of spent fuel. The IPEC radiation protection program and application of these procedures also supports the IPEC commitment to compliance with the requirements of 10 CFR 20 for occupational dose considerations. See Section F.2.A for a more detailed list of procedures.

*72.126(b): Radiological alarm systems. Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.*

The HI-STORM 100 System emits no radioactive effluents once the MPC is prepared for storage. Existing alarm systems for radiological monitoring fuel loading operations in the FSB have been judged sufficient to warn personnel of inappropriate airborne or direct radiation. In addition to the area radiation monitors and continuous air monitoring in the spent fuel pit area and area monitors in the FSB, monitoring is performed during onsite transportation as required by the site DCS procedures.

*72.126(c): <u>Effluent and direct radiation monitoring.</u> (1) As appropriate for the handling and storage system, effluent systems must be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided.*

*(2) Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.*

The MPCs have redundant seal-welded closures, so no gaseous radioactive material leak path to the environment is available and no routine monitoring of effluents from the HI-STORM casks is required. Other than procedural requirements necessary for normal radiological surveys, use of remote instrumentation is not anticipated.

An area environmental dose monitoring device is installed in the ISFSI area to meet the TLD and radiological environmental monitoring requirements.

*72.126(d): <u>Effluent control.</u> The ISFSI or MRS must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in § 72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.*

The MPCs have redundant seal-welded closures, so no gaseous radioactive material leak path to the environment is available and no routine monitoring of effluents from the HI-STORM casks is required. The MPC design has been shown by analysis to maintain the confinement boundary integrity under all normal, off-normal, and accident conditions of service as discussed in Chapter 3 of the HI-STORM 100 System FSAR. The MPC lid-to-shell closure weld design meets the guidance in ISG-18 and the MPC vent and drain port cover plates are leak tested to a "leak tight" acceptance criterion as defined in ANSI N14.5 and as discussed in Chapter 7 of the HI-STORM 100 System FSAR. Based on these two factors, leakage from the MPC confinement boundary is considered non-credible and no effluent controls or dose analysis is required.

### F.3.5 §72.44, §72.144, §72.190, §72.194 – Training and Operator Requirements

*§72.44(b)(4): The licensee shall have an NRC-approved program in effect that covers the training and certification of personnel that meets the requirements of subpart I before the licensee may receive spent fuel and/or reactor-related GTCC waste for storage at an ISFSI or the receipt of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste for storage at an MRS.*

*§72.44(b)(5): The license shall permit the operation of the equipment and controls that are important to safety of the ISFSI or the MRS only by personnel whom the licensee has certified*

*as being adequately trained to perform such operations, or by uncertified personnel who are under the direct visual supervision of a certified individual.*

*§72.144(d): The licensee, applicant for a license, certificate holder, and applicant for a CoC shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained.*

*§72.190: Operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important to safety must also be certified in such operations.*
*§72.194: The physical condition and the general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defect.*

The IPEC Training Program was reviewed to assure compliance to the requirements of 10 CFR 72 for the handling, transporting and storage of dry fuel storage canisters.

The Holtec-designed HI-STORM 100 System training requirements are found in the HI-STORM FSAR and CoC. The Training Program at IPEC has been reviewed and verified to meet these requirements and 10 CFR 72 requirements. Specifics of this review are listed below:

- HI-STORM FSAR Section 8.0 requires training procedures in place to account for operation of an ISFSI. The IPEC Dry Fuel Storage Training Program directs the training, qualification and continuing training of DCS personnel.

- HI-STORM FSAR Section 12.2.1 requires training modules developed or modified to require a comprehensive, site-specific training, assessment and qualification program for the operation and maintenance of the HI-STORM 100 System & ISFSI. The IPEC DCS Training Program contains all the course curriculum and requirements for each training module of the DCS Training Program. This includes training and qualification requirements.

- HI-STORM FSAR Section 12.2.2 and CoC Condition 10 require dry run training exercises of the loading, closure, handling, and transfer of the HI-STORM 100 System components to be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The procedures, listed in Section F.2A were developed to ensure compliance with these requirements.

- 10 CFR 72.44(b)(4) and 10 CFR 72.190 require a training program in effect that covers the training and certification of personnel that operate or supervise the operation of equipment and controls that have been identified as important to safety in the safety analysis report and license. The IPEC DCS Training Program ensures compliance with these requirements.

- 10 CFR 72.44(b)(5) and 10 CFR 72.190 require that only trained and certified personnel operate equipment and controls that have been identified as important to safety in the safety analysis report and license. The IPEC DCS Training Program ensures compliance with these requirements.

- 10 CFR 72.144 (d) requires the licensee, applicant for a license, certificate holder, and applicant for a CoC to provide for indoctrination and training of personnel performing activities affecting quality as necessary to ensure that suitable proficiency is achieved and maintained. Compliance & implementation of the following ensures satisfaction of these requirements:

  - The IPEC DCS Training Program
  - ASNT-SNT-TC-1A
  - ANSI-45.2.6
  - QAPM
  - NDE 2.10, Certification of NDE Personnel.
  - NDE 2.12, Certification of Visual Testing (VT) Personnel.
  - QV-111, Quality Control Certification.

- 10 CFR 72.194 requires the physical condition and the general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defects. Compliance with ASME B30.2, and procedures NS-102, Fitness for Duty, NS-112, Medical Program, and MA-119, Material Handling Program, ensures compliance with these requirements.

### F.3.6 §72.212(a)(3) – License Extension

*The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance terminates 20 years after the date that the particular cask is first used by the general licensee to store spent fuel, unless the cask's Certificate of Compliance is renewed, in which case the general license terminates 20 years after the cask's Certificate of Compliance renewal date. In the event that a cask vendor does not apply for a cask model re-approval under §72.240, any cask user or user's representative may apply for a cask design re-approval. If a Certificate of Compliance expires, casks of that design must be removed from service after a storage period not to exceed 20 years.*

The beginning of the 20-year IPEC 10 CFR 72 general license is when the first fuel assembly is loaded into a HI-STORM MPC. Current options upon approaching the end of life include; 1) renew the ISFSI license (via renewal of the cask CoC), 2) buy a different cask design and transfer the fuel, or 3) ship the fuel to another location (i.e., a permanent repository).

Section V of the main body of this report addresses IPEC-specific plans for license renewal.

### F.3.7 §72.212(b)(2)(i)(A) – Review of the CoC

*72.212(b)(2)(i)(A): Perform written evaluations, prior to use, that establish that conditions set forth in the Certificate of Compliance have been met.*

Section VI of the main body of this report addresses the CoC conditions that have generic responses applicable to all of the Entergy plants currently using the HI-STORM 100 System. See Section F.4 for detailed, IPEC site-specific discussion of the CoC, including Appendices A and B.

### F.3.8 §72.212(b)(2)(i)(B) – ISFSI Design

*72.212(b)(2)(i)(B): Perform written evaluations, prior to use, that establish that cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion.*

**STORAGE PAD**

The IPEC ISFSI is located approximately 370 feet north of the Unit 2 containment. The pad surface is at approximately 90 feet mean sea level (MSL).

The reinforced concrete ISFSI pad is designated as not important to safety. However, to provide additional assurances of proper design and installation, the ISFSI pad was designed and constructed as a Safety Related structure. The basis for this classification is that the dry fuel cask systems are stored as free standing structures with a lack of physical connection in accordance with the Holtec HI-STORM 100 System FSAR requirements (see Section F.4.3.2.3 of this appendix).

The HI-STORM 100 System design basis requires that neither a non-mechanistic tip-over nor an 11-inch vertical drop of a loaded cask onto the ISFSI pad results in cask deceleration levels greater than 45g's at the top of the fuel for the HI STORM 100S overpack. The cask deceleration is a function of the following factors associated with the ISFSI pad:

- Modulus of elasticity of the soil subgrade
- Thickness of concrete pad
- Compressive strength of concrete
- Strength of concrete reinforcement

The IPEC ISFSI pad analysis[2] uses the following primary parameters found in the HI-STORM FSAR (Rev.4), Table 2.2.9, parameter Set "A":

- Surface that provides a static coefficient of friction $\geq$ 0.53 between the concrete and the bottom of the overpack.
- Maximum compression strength of concrete ($f'_c$) at 28 days curing = 4.2 ksi.
- Maximum modulus of elasticity ($E_{sg}$) of subgrade = 28,000 ksi.
- Maximum slab thickness (t) = 36".
- Pad reinforcement (top and bottom, both directions): 60 ksi yield strength ASTM material.

Applicable Loads & Combinations

The load combinations used in the design of the reinforced concrete ISFSI pad are in compliance with the HI-STORM FSAR and NUREG-1536. The free standing casks are to be stored in a 6 x 13 array providing space for 75 casks. The ISFSI is 200 feet long by 90 feet wide.

The storage pads are reinforced concrete flush on grade with access for the Vertical Cask Transporter. Concrete material used in the construction of the pads is made of air-entrained, normal weight concrete with a twenty-eight day compressive strength between 3000 and 4200 psi.

The storage pad is designed for the loads from the HI-STORM 100S storage overpacks, transporter equipment, and live loads. The pad is a non-safety-related structure, however it is designed for seismic loads.

Calculations and drawings that document the storage pad design include the following:

- Calculation FCX-00550-00 – ISFSI Concrete Pad Design Calculation
- Calculation FCX-00568-00 – ISFSI Retaining Wall
- Calculation FCX-00540-00 – ISFSI Development of Acceleration Time Histories
- Calculation FCX-00541-00 – ISFSI Development of Strain Compatible Soil Properties
- Calculation FCX-00542-00 – ISFSI SSI Analysis of Support Pad with Casks
- Calculation Ip-RPT-05-00754 – ISFSI SSI Analysis of Support Pad with Casks with New Soil Properties.
- Drawing DMD-400557-AA – ISFSI Storage Pad, Concrete Outline and Reinforcing

The ISFSI storage pad was implemented in accordance with ENN-DC-115 – ER Response Development.

Soils

[2] Calculation FCX-00550-00, "ISFSI Concrete Pad Design Calculation"

The design of the storage pad was based on subsurface soil investigations performed by Tectonic Engineering and Surveying consultants P.C. The work included field exploration (soil borings), soil testing (field and laboratory) and a geotechnical assessment of the data. The subsurface soil investigations are documented in the following reports:

- IP-RPT-06-00007 "Geotechnical Test Results, Proposed ISFSI Sites, Indian Point energy Center, Buchanan, New York", dated June 6, 2003.
- IP-RPT-04-00885 "Geotechnical Investigation Independent Spent Fuel Storage Installation Indian Point 2 Power Plant Buchanan, New York"
- 3635.03 "Recommendations For Alternate Engineered Fill Aggregate Proposed Independent Spend Fuel Storage Installation"

Design Basis Earthquake (DBE) Spectra

Seismic loads for the design of the ISFSI reinforced pad are based on the IPEC safe shutdown earthquake (SSE) response spectra corresponding to a maximum zero period ground acceleration (ZPA) value of 0.15 g. Vertical ground motion corresponds to a maximum vertical acceleration of 0.10 g.

Conclusion

The results of the storage pad analysis show that all H-STORM CoC and FSAR analysis requirements are met for (normal) storage, seismic, tip over, and sliding considerations. It is also noted that the HI-STORM FSAR-described non-mechanistic tip over and end drop design basis analysis performed using LS-DYNA3D are enveloped by the IPEC-performed analysis. Also, the HI-STORM is moved within 11 inches of the ground by the onsite VCT and does not challenge the 11-inch maximum height limitation. In accordance with the HI-STORM FSAR, the ISFSI owner has the option of constructing the pad to comply with specific limits set forth without performing site specific cask impact analysis. IPEC has chosen to comply with "Set A" ISFSI pad design parameters. Other than industry codes such as NUREG-1536 and ACI, the FSAR does not specify the analysis methodology means of performing the pad analysis to meet the specific limits.

**F.3.9 §72.212(b)(2)(i)(C) – Dose Limitations per §72.104**

*72.212(b)(2)(i)(C): Perform written evaluations, prior to use, that establish that the requirements of §72.104 have been met. A copy of this record must be retained until spent fuel is no longer stored under the general license issued under §72.210.*

10 CFR 72.104 requires that for normal operation, the annual dose to any real individual beyond the controlled area must not exceed 25 mrem whole body and 75 mrem to the thyroid, or 25 mrem to any other organ from any discharges or direct radiation.

A site specific calculation performed by the cask vendor provides a description of the general methodology and analyses performed to estimate the annual dose for various cask placement configurations and distances from the cask storage area for six loaded casks (and 5 loaded Unit 1 casks) assuming 45,500 MWD/MTU burnup and 10 year cooling for Unit 2 fuel. (Reference: Holtec Reports HI-2073724, date July 30, 2007 and HI-2073736, dated September 7, 2007)

These calculations also address the contact dose requirements of Appendix A to the Holtec CoC, Technical Specification 5.7.3 for the HI-STORM and the HI-TRAC.

From the center of the IPEC ISFSI pad, the boundaries of the IPEC owner controlled area are approximately 472 meters (1548 feet) to the north, 592 meters (1944 feet) to the east, greater than 600 meters (>2000 feet) to the south, and 169 meters (554 feet) to the west. In the east and south directions, the eastern boundary is the nearest at 1944 feet. The dose rate from the ISFSI at the eastern boundary, coincident with the nearest resident is calculated to be approximately 0.14 mrem/year per cask for the Unit 2 Casks. Assuming 8760 hours of occupancy, the annual dose would be approximately 1 mrem/year with the contribution from the five Unit 1 casks included. The western and northern boundary of the owner controlled area is the shoreline of the Hudson River.

After the initial placement of 11 casks at the ISFSI, six Unit 2 casks and five Unit 1 casks, actual dose measurements will be documented and used to benchmark the site boundary calculations prior to the placement of additional casks with fuel from Units 2 and 3. The actual fuel burnup and cool down times for fuel in subsequent casks will be used as the source term rather than the bounding 45,000/10 year input used for the first six casks. The site boundary dose assessment will be reanalyzed to determine the maximum number of casks permissible while meeting the limits of 10 CFR 72.104 and 40 CFR 190.

Exclusive of the ISFSI, the normal operating doses from the IPEC site are documented in the Annual Radiological Environmental Operating Report (last issued April 30, 2006 for 2005 data). Direct radiation is measured at 18 locations within 2 miles of the plant site. The dose contribution from the operating plant facilities to a member of the public at the site boundary has been recorded at less than 1 mrem/year in the Report. Calculated or design dose contributions from other site sources, e.g. Temporary Low Level Storage Building, Steam Generator Mausoleums, totals approximately 7.5 mr/year at the site boundary. The calculated dose from the ISFSI combined with the recorded and calculated design dose from the remaining facilities (Units 1, 2, and 3) is less than the 40 CFR 190 limit of 25 mrem/yr, i.e. 9.5 mrem/year assuming 8760 hours/year occupancy.

**F.3.10 §72.212(b)(3) – Review of the Cask FSAR and SER**

*72.212(b)(3): Review the Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(2) of this section.*

The HI-STORM FSAR and the NRC's Safety Evaluation Report have been reviewed and a determination made that the reactor site parameters at IPEC are bounded by the assumptions made in the generic cask FSAR and the NRC's safety basis in the SER. The details of this review as they pertain to issues such as earthquake intensity and tornado missiles may be found elsewhere in this appendix. Section F.5 of this appendix provides a chapter-by-chapter assessment of deviations between requirements in the cask FSAR and implementation of dry storage at the IPEC site. All cask FSAR requirements of the licensee that are not listed in Section F.5 of this appendix are met. Deviations from cask FSAR requirements have been evaluated under the IPEC 10 CFR 72.48 program, as applicable.

**F.3.11 §72.212(b)(4) – Changes to 10 CFR Part 50 Technical Specifications**

*72.212(b)(4): Prior to use of the general license, determine whether activities related to storage of spent fuel under this general license involve a change in the facility Technical Specifications or require a license amendment for the facility pursuant to §50.59(c)(2) of this chapter. Results of this determination must be documented in the evaluation made in paragraph (b)(2) of this section.*

Several design modifications, each having their own 10 CFR 50.59 evaluation, have been implemented in support of dry fuel storage at IPEC. None of these modifications required a changed to the IPEC operating license or technical specifications. However, a Part 50 license amendment that did not involve a change to the IPEC Part 50 technical specifications was required to obtain NRC approval of the use of the FSB single-failure-proof gantry crane for cask loading operations.

IPEC operating license amendment No. 244 was granted by the NRC on November 21, 2005 providing the approval to use the FSB gantry crane for all dry fuel storage cask loading operations in the Unit 2 FSB.

**F.3.12 §72.212(b)(5) – Protection Against Radiological Sabotage**

*72.212(b)(5): Protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to §73.55 of this chapter with the following additional conditions and exceptions.*

*(i) The physical security organization and program for the facility must be modified as necessary to assure that activities conducted under this general license do not decrease the effectiveness of the protection of vital equipment in accordance with §73.55 of this chapter.*

*(ii) Storage of spent fuel must be within a protected area, in accordance with §73.55(c) of this chapter, but need not be within a separate vital area. Existing protected areas may be expanded or new protected areas added for the purpose of storage of spent fuel in accordance with this general license.*

*(iii) For purposes of this general license, searches required by §73.55(d)(1) of this chapter before admission to a new protected area may be performed by physical pat-down searches of persons in lieu of firearms and explosives detection equipment.*

*(iv) The observational capability required by §73.55(h)(6) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of closed circuit television.*

*(v) For the purpose of this general license, the licensee is exempt from §§73.55(h)(4)(iii)(A) and 73.55(h)(5) of this chapter.*

The ISFSI is located within the IPEC protected area. During transport to the ISFSI pad from the FSB, the overpack will remain in the protected area. The site Security Plan and Physical Protection Program provide the appropriate controls during this transport activity. The IPEC Security Plan was updated to include the IPEC ISFSI in accordance with the requirements of 10 CFR 50.54(p). Any future changes to the plan will also be made in accordance with §50.54(p).

### F.3.13 §72.212(b)(6) – Review of the Emergency Plan, Quality Assurance Program, Training Program, and Radiation Protection Program

*72.212(b)(6): Review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if their effectiveness is decreased and, if so, prepare the necessary changes and seek and obtain the necessary approvals.*

<u>Emergency Plan</u>

The IPEC Emergency Plan is maintained to meet the regulations in 10 CFR 50.47, 10 CFR 50.54, 10 CFR 50.72, and 10 CFR 50 Appendix E. The plan follows the guidelines established in NUREG-0654/FEMA REP-1, Revision 1. 10 CFR 50.47(b) lists the sixteen planning standards that must be met in the Emergency Plan. The IPEC Emergency Plan was reviewed to determine if changes required to support implementation of the Holtec HI-STORM 100 System at the IPEC ISFSI will decrease the effectiveness of the Emergency Plan.

The following licensing basis documents were consulted in this review:

1. Holtec FSAR, Revision 4
2. HI-STORM CoC, Amendment 2
3. 10 CFR 72.32c
4. 10 CFR 72.212(b)(6)

The Holtec HI-STORM 100 System does not decrease the effectiveness of the IPEC Emergency Plan or implementing procedures as it relates to Part 50 activities. However, the HI-STORM FSAR does contain provisions to establish procedures to:

- Address establishing emergency action levels and implementation of the emergency action program (Chapter 8)
- Written procedures to account for such things as emergency response. (Chapter 8)
- Address removal of material blocking the air inlet ducts prior to the fuel clad reaching its short term temperature limit. (Blizzard snow accumulation, though remote, is the only credible mechanism for air inlet blockage.)
- Include emergency action plan provisions for corrective actions for cask burial under debris. (This scenario is considered not credible due to the physical arrangement of the ISFSI pad in relation to the surrounding topography.)

The IPEC Emergency Plan and site classification procedures address ISFSI action levels and implementation of the emergency action program.. When an event is declared, the Emergency Director's responsibility is to make the notifications and to ensure availability of response staff. The corrective actions are implemented in accordance with plant procedures commensurate with the safety significance of the situation.

Quality Assurance Program

The Entergy Nuclear Northeast (ENN) Quality Assurance Program and Quality Assurance Program Manual (QAPM) were reviewed to assure compliance to the requirements of 10 CFR 72 for the handling, transporting and storage of dry fuel storage canisters.

The Holtec HI-STORM 100 storage system QA requirements are found in the HI-STORM FSAR and CoC, which impose the requirements of 10 CFR 72 on both licensees and certificate holders. The ENN QAPM applies to all activities associated with structures, systems, and components that are safety related or controlled by 10 CFR 72. The methods of implementation of the requirements of the QAPM are commensurate with the item's or activity's importance to safety. The applicability of the requirements of the QAPM to other items and activities is determined on a case-by-case basis. The QAPM implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G.

Holtec uses a graded quality approach on various subcomponents associated with the HI-STORM overpack, the HI-TRAC transfer cask, the MPC, and the ancillary components used to facilitate cask loading and onsite transport. This approach is covered by ENN and/or site specific procedures that implement the QAPM.

Therefore, the current ENN Quality Program and QAPM are not impacted by any requirements for the Holtec HI-STORM 100 system.

Training Program

The Holtec HI-STORM 100 storage system training program requirements are found in the HI-STORM FSAR and CoC, which invoke the requirements of 10 CFR 72 and require cask design-specific topics for personnel training. The IPEC dry fuel storage training program uses the Systematic Approach to Training and is based on 10 CFR 50 and INPO guidelines. The program addresses all training requirements specifically listed in the HI-STORM FSAR and CoC and well as requirements in other parts of the 10 CFR 72 regulations. See Section D.3.5 for a more detailed discussion of the dry fuel storage training program.

Radiation Protection Program

The Holtec HI-STORM 100 storage system radiological protection requirements are found in the HI-STORM FSAR and CoC, which invoke the requirements of 10 CFR 72 and provide cask-specific requirements. The IPEC Radiological Protection Program has been reviewed and modified as necessary to address dry spent fuel cask loading, unloading, and storage operations. In addition, the requirements in HI-STORM CoC Appendix A, Section 5.7 "Radiation Protection Program," have been addressed. See Section F.4.2.4 of this appendix for more information.

## SECTION F.4
## COMPLIANCE WITH HI-STORM 100 CASK SYSTEM
## CERTIFICATE OF COMPLIANCE

### F.4.1 Certificate of Compliance Conditions

Compliance with the Holtec HI-STORM 100 System Certificate of Compliance (CoC) is discussed on an Entergy system-wide basis in Section VI of the main body of this report. Conditions 5, 9, and 10 of the CoC, where compliance requires a unique discussion for the IPEC ISFSI, are addressed below.

#### F.4.1.1 Condition 5 – Heavy Loads Evaluation

*Each lift of an MPC, a HI-TRAC transfer cask, or any HI-STORM overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific regulatory review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.5 of Appendix A and/or Sections 3.4.6 and 3.5 of Appendix B to this certificate, as applicable.*

At IPEC Unit 2, the loaded MPC and HI-TRAC transfer cask, and associated lids and mating devices, will be lifted with the single failure proof FSB gantry crane in accordance with the guidance of NUREG-0612 and implemented by the cask loading procedures. An overview of the design of the FSB gantry crane and the license amendment authorizing use of the gantry crane to move spent fuel casks is provided in Section F.1.5.

The loaded overpack will be transported from outside the FSB to the ISFSI by the VCT. The VCT is a commercial-grade item designed in accordance with the applicable sections of ANSI N14.6 with redundant drop protection features. In accordance with CoC Appendix A, Section 5.5.a.3, a drop of the loaded overpack or HI-TRAC transfer cask is not credible when the redundant locking pins are engaged. A site specific analysis in accordance with CoC Appendix A, Section 5.5.a.2 has been performed to demonstrate the HI-STORM design loading of 45 g's is not exceeded during a drop onto the Low Profile Transporter while the HI-STORM is being raised or lowered prior to engaging the locking pins.

The maximum lift height of the HI-STORM is restricted to 11". When the HI-STORM is being lowered onto the ISFSI Pad, this lift height is within the limits specified in Technical Specification 5.5.a.1.

#### F.4.1.2 Condition 9 – Special Requirements for First Systems in Place

*The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM Cask Systems (for each unique MPC basket design - MPC-24/24E/24EF, MPC-32/32F, and MPC-68/68F/68FF) placed into service, by any user, with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR.*

*Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.*

*Each first time user of a HI-STORM 100 Cask System Supplemental Cooling System (SCS) that uses components or a system that is not essentially identical to components or a system that has been previously tested, shall measure and record coolant temperatures for the inlet and outlet of cooling provided to the annulus between the HI-TRAC and MPC and the coolant flow rate. The user shall also record the MPC operating pressure and decay heat. An analysis shall be performed, using this information, that validates the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling necessary.*

*Letter reports summarizing the results of each thermal validation test and SCS validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.*

The portion of this CoC requirement pertaining to cask heat load applies to all general licensees using the Holtec HI-STORM 100 System. IPEC has confirmed that this CoC requirement has been successfully implemented by other HI-STORM users based on the heat loads of MPC-32/32Fs to date. The HI-STORM 100 cask system and MPC-32 was first used by the Tennessee Valley Authority's Sequoyah Nuclear Power Station for storage of spent fuel with heat loads up to 22.6 kW beginning on July 13, 2004 (Docket No. 72-034). The results of measurements made of the temperature rise from the HI-STORM inlet ducts to the outlet ducts were reported to the NRC in a letter from TVA to the NRC on October 15, 2004 and show the measured temperatures were lower than the calculated temperatures. In accordance with CoC 1014, Condition 9, no additional testing of the HI-STORM 100 system is required and reference to the TVA letter for Sequoyah fulfills the requirement of Condition 9. Therefore, no temperature data is required to be taken by IPEC and no reports need to be submitted to the NRC.

The portion of this CoC requirement pertaining to the Supplemental Cooling System (SCS) applies only to general licensees using the Holtec HI-STORM 100 System who load high burnup fuel (burnup > 45,000 MWD/MTU). CoC Appendix A, LCO 3.1.4 requires the SCS to be used only if high burnup fuel is loaded into the MPC. IPEC does not plan to load high burnup fuel at this time. Therefore, this requirement of the CoC is not applicable.

### F.4.1.3 Condition 10 – Pre-Operational Testing and Training Exercise

*A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:*

a. *Moving the MPC and the transfer cask into the spent fuel pool.*

b. *Preparation of the HI-STORM 100 Cask System for fuel loading.*

c. *Selection and verification of specific fuel assemblies to ensure type conformance.*

d. *Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.*

e. *Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool.*

f. *MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)*

g. *Operation of the Supplemental Cooling System.*

h. *Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling equipment.*

i. *Transfer of the MPC from the transfer cask to the overpack.*

j. *Placement of the HI-STORM 100 Cask System at the ISFSI.*

k. *HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.)*

The dry run training exercises are made up of scenarios performed over the course of several weeks. The details of the dry runs are documented in the work orders which will be used to control the dry run exercises.

The Supplemental Cooling System (SCS) operation test/training exercise will not be performed as IPEC is not loading high burnup fuel at this time. If IPEC decides to load high burnup fuel in the future, the SCS demonstration will be completed prior to that loading campaign.

The Transfer Cask upending/downending test/training exercise will not be performed. The site's cask handling arrangement during spent fuel loading and unloading operations does not require handling of the transfer cask in the horizontal orientation. The transfer cask will be handled vertically at all times.

As a result of a public comment during rulemaking for the HI-STORM CoC Amendment 2, the NRC modified the HI-STORM CoC to re-instate a helium leakage test for the MPC vent and drain port cover plates. This CoC change was captured in a Holtec Engineering Change Order to make a conforming change to Revision 3 to the HI-STORM FSAR, which had already been issued. This requirement remains in FSAR Revision 4.

### F.4.2 CoC Appendix A – Technical Specifications

Compliance with the Holtec HI-STORM 100 System CoC, Appendix A, "Technical Specifications," is discussed on an Entergy system-wide basis in Section VI of the main body of this report. Technical Specifications where compliance requires a unique discussion for the IPEC ISFSI, are addressed below.

#### F.4.2.1 Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs)

##### F.4.2.1.1 LCO and SR 3.0 Series - LCO and SR Applicability

The LCO and SR 3.0 series of technical specifications establish general requirements for use and implementation of the specific LCOs and SRs that follow. These general requirements are very similar to those used in the IPEC Unit 2 Part 50 standard technical specifications. No specific actions or implementation procedures are required. Operators are trained on the use of the LCO and SR 3.0 series of technical specifications.

##### F.4.2.1.2 LCO 3.1.1 – Multi-Purpose Canister (MPC)

This LCO establishes the MPC fuel cavity drying and helium backfill acceptance criteria for establishing the required heat transfer and corrosion-resistant environment for the stored fuel. With regard to MPC cavity drying, IPEC is choosing to use the Forced Helium Dehydration (FHD) System. Therefore, the drying acceptance criteria for the FHD system in Table 3-1 of HI-STORM CoC Appendix A apply. The helium backfill pressure range for MPC-32/32F in Table 3-2 of HI-STORM CoC Appendix A is used as the acceptance criterion. These requirements are implemented via IPEC procedures.

##### F.4.2.1.3 LCO 3.1.2 – SFSC Heat Removal System

This LCO establishes operability and surveillance requirements for the HI-STORM overpack natural ventilation heat removal system. For loaded HI-STORM overpacks stored on the ISFSI pad, daily surveillances of the inlet and outlet air ducts for blockage are performed in accordance with the applicable daily surveillance test procedure (the procedure used depends on the operating mode of the plant). If any air duct blockage is found, a work order is initiated to return the air ducts to the operable condition within the completion time established for the condition in the LCO.

##### F.4.2.1.4 LCO 3.1.3 – Fuel Cooldown

This LCO establishes requirements for ensuring that the bulk temperature of the MPC fuel cavity gas is less than or equal to 200°F before re-flooding the cavity with water in the event an MPC needs to be unloaded. See Section F.1.4 for a summary of the cask unloading operational sequence where this LCO would apply.

##### F.4.2.1.5 LCO 3.1.4 – Supplemental Cooling System

This LCO establishes operability requirements for a supplemental cooling system required to be used if one or more high burnup fuel assemblies (burnup $\geq$ 45 GWD/MTU) are loaded into the MPC. Currently, IPEC does not plan to load any high burnup fuel assemblies. Therefore, this LCO is not applicable to dry cask storage at IPEC at this time.

##### F.4.2.1.6 LCO 3.2.2 – Transfer Cask Surface Contamination

This LCO establishes removable alpha, beta, and gamma radiation contamination limits for the transfer cask surface and accessible portions of the MPC during on site transport operations. IPEC procedure "MPC Loading and Sealing Operations," includes steps to ensure that the transfer cask and accessible portions of the MPC are decontaminated to levels meeting the limits specified in this LCO prior to entering the transport operations mode.

**F.4.2.1.7 LCO 3.3.1 – Boron Concentration**

This LCO establishes minimum soluble boron concentration requirements in the MPC water during fuel loading in MPC 32/32F. For the Unit 2 W 15x15 fuel, the minimum boron concentrations are 1,800 ppm for initial enrichments ≤4.1 wt% U-235 and 2,500 ppm for initial enrichments >4.1 wt% and ≤5,0 wt% U-235. This LCO is implemented in IPEC procedure, 2-DCS-008-GEN, " Unit 2 MPC Loading and Sealing Operations." IPEC is not loading assemblies which would be designated as "damaged fuel" so the higher boron limits for "damaged fuel" are not applicable to the 2007/2008 fuel loading campaign.

**F.4.2.2 Section 5.4 – Radioactive Effluent Control Program**

This program implements the requirements of 10 CFR 72.44(d).

a. *The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.*

b. *This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations*

c. *An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).An annual report be submitted to the Commission in accordance with Sec. 72.4, specifying the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous 12 months of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent releases. On the basis of this report and any additional information that the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate. The report must be submitted within 60 days after the end of the 12-month monitoring period.*

An annual report is submitted pursuant to 10 CFR 72.44(d)(3) requirements in accordance with the IPEC Offsite Dose Calculation Manual.

**F.4.2.3 Section 5.5 – Cask Transport Evaluation Program**

*This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable* *when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.).*

a. *For free-standing OVERPACKS and the TRANSFER CASK, the following requirements apply:*

1. *The lift height above the transport route surface(s) shall not exceed the limits in [CoC] Table 5-1 except as provided for in Specification 5.5.a.2. Also, the program shall ensure that the transportation route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than either Set A or Set B in HI-STORM FSAR Table 2.2.9.*

2. *For site-specific transport route surfaces that are not bounded by either the Set A or Set B parameters of FSAR Table 2.2.9, the program may determine lift heights by analysis based on the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g's at the top of the MPC fuel basket. These alternative analyses shall be commensurate with the drop analyses described in the Final Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.*

3. *The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during transportation between the FUEL BUILDING and the CTF and/or ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.*

4. *The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specific in Section 3.5 of Appendix B to Certificate of Compliance 1014, as applicable.*

For lifting of the loaded HI-TRAC, all lifts are made by the single-failure-proof gantry crane in the FSB. The transfer of the MPC from the HI-TRAC to the HI-STORM is also performed with the gantry crane in the FSB. The loaded HI-STORM overpack is handled by the Vertical Cask Transporter (VCT).

The Unit 2 Low Profile Transporter and the VCT is used to convey the loaded OVERPACK from the Unit 2 FHB to the ISFSI pad.

The Vertical Cask Transporter meets the requirements of Technical Specification 5.5.a.3. and a drop of the HI-STORM or the HI-TRAC during TRANSPORT operations is not considered a credible event. (Reference: Letter; Joe Reiss, Holtec to P. Peloquin, IPEC; "IPEC Vertical Cask Transporter Compliance with the HI-STORM CoC"; Document ID: 1535005; October 24, 2007.)

While the HI-STORM is being lifted from the LPT and being lowered onto the ISFSI pad by the VCT hydraulic system, the redundant locking pins are not engaged. A failure of the hydraulic system could, during this short duration evolution, result in a drop of the HI-STORM from a height of 11 inches onto the ISFSI pad. This is within the Technical Specification limits specified in TS Table 5-1 which is applicable to engineered concrete ISFSI pads meeting the parameters of FSAR table 2.2.9.

A site specific analysis has been performed for a drop of six inches onto the Low Profile Transporter (LPT). The LPT does not meet the ISFSI design assumptions and requirements of FSAR Table 2.2.9 which identifies limits for an engineered concrete pad. The LPT is a steel platform mounted on rollers and a track. The analysis demonstrates that a HI-STORM drop of 6 inches onto the LPT results in a maximum deceleration of 16.8 g's which is below the design limit of 45 g's. (Reference: IPEC Document No. IP-RPT-07-00080, dated August 15, 2007)

The IPEC ISFSI design utilizes free standing OVERPACKS. Section 5.5.b of this Specification does not apply to IPEC

### F.4.2.4 Section 5.7 – Radiation Protection Program

*1. Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK and TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposure As Low As Reasonably Achievable (ALARA). The action and criteria to be included in the program are provided below.*

*2. As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.*

*3. Based on the analysis performed pursuant to [item 2], the licensee shall establish individual cask surface dose rate limits for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:*

*a. The top of the TRANSFER CASK and the OVERPACK*

*b. The side of the TRANSFER CASK and OVERPACK*

*c. The inlet and outlet ducts on the OVERPACK*

*4. Notwithstanding the limits established in [item 3], the measured dose rates on a loaded OVERPACK shall not exceed the following values:*

*a. 20 mrem/hr (gamma + neutron) on the top of the OVERPACK*

*b. 110 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts*

5. *The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in [item 8] for comparison against the limits established in [item 3] or [item 4], whichever are lower.*

6. *If the measured surface dose rates exceed the lower of the two limits established in [item 3] or [item 4], the licensee shall:*

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

b. *Perform a written evaluation to verify whether placement of the as-loaded OVERPACK at the ISFSI will cause the dose limits of 10 CFR 72.104 to be exceeded, [and]*

c. *Perform a written evaluation within 30 days to determine why the surface dose rates were exceeded.*

7. *If the evaluation performed pursuant to [item 6] shows that the dose limits of 10 CFR 72.104 will be exceeded, the OVERPACK shall not be placed into storage until the appropriate corrective action is taken to ensure the dose limits are not exceeded.*

8. *TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:*

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

b. *A minimum of four (4) TRANSFER CASK top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid.*

c. *A minimum of twelve (12) dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken at approximately the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.*

d. *A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top concrete shield, 90 degrees apart around the circumference of the lid.*

e. *A dose rate measurement shall be taken on contact at the surface of each inlet and outlet vent duct screen.*

A site specific analysis was performed in accordance with the HI-STORM CoC Appendix A, Sections 5.7.2 and 5.7.3. (Holtec Calculation HI-2073736) The analysis for the initial six canisters to be loaded assumed 45,000 MWD/MTU and 10 Years Cooling. Based on this analysis, the overpack and transfer cask surface dose limits to be used for the site are as follows:

| CoC Appendix A Reference | | Location | Site-Specific Calculated Total Surface Dose Rate (Dry Condition mrem/hr) | Max value per Technical Specification 5.7.4 (mrem/hr) |
|---|---|---|---|---|
| 5.7.3 | a(1) | The top lid of the transfer cask | 202.6 | N/A |
| | a(2) | The top lid of the overpack | 5.4 | 20 |
| 5.7.3 | b(1) | The side of the transfer cask (mid-height) | 801.5 | N/A |
| | b(2) | The side of the overpack mid height<br>60 inches below mid height<br>60 inches above mid height | 21.1<br>20.8<br><15.1 | 110<br>110<br>110 |
| 5.7.3 | c(1) | The inlet air duct of the overpack | 38.3 | N/A |
| | c(2) | The outlet air duct of the overpack | 16.7 | N/A |

The calculated dose rates on the top and on the side of the overpack are lower than the CoC Appendix A, Section 5.7.4 limits of 20 mrem/hr and 110 mrem/hr, respectively. Therefore, the calculated total surface dose rates above are the appropriate limits to apply for comparison to measured values.

IPEC Procedure 0-RP-RWP-420 implements the dose rate measurement requirements of CoC Appendix A, Section 5.7.8 for the HI-TRAC transfer cask and HI-STORM overpack. The measured dose rates are compared to the limits established in the table above in accordance with CoC Appendix A, Section 5.7.5. If measured dose rates exceed the established limits, the actions required by CoC Appendix A, Sections 5.7.6 and 5.7.7 (if required), will be implemented.

### F.4.3 CoC Appendix B – Approved Contents and Design Features

Section 2.0 of Appendix B to the HI-STORM CoC, "Approved Contents," is discussed in the main body of this report, Section VI. An IPEC procedure is used to select fuel assemblies for storage in the HI-STORM 100 System that meets all applicable requirements of HI-STORM CoC Amendment 2, Appendix B, Sections 2.1, 2.2, and 2.4 (Section 2.3 of this CoC appendix does not exist).

Section 3.0 of Appendix B to the HI-STORM CoC, "Design Features," is discussed is different sections of this report. Sections 3.1, "Site" and 3.3, "Codes and Standards," are addressed in Section VI of the main body of this report. Sections 3.2, 3.4 and 3.6 through 3.8 are addressed site-specifically for the IPEC ISFSI below. Section 3.5 pertains to a Cask Transfer Facility (CTF) which is not used for cask loading operations at IPEC and is, therefore, not applicable and not discussed further.

#### F.4.3.1 Section 3.2 – Design Features Important for Criticality Control

This section of the CoC addresses certain design features important for criticality control for all HI-STORM 100 System MPC models certified under 10 CFR 72. Sections 3.2.1 through 3.2.4 pertain to the MPC-24, MPC-68/68FF, MPC-68F and MPC-24E/EF, respectively. The 68/68FF and MPC-68F are BWR models. The MPC-24 and MPC-24E/EF are PWR models not used at IPEC. Therefore, these CoC sections are not applicable to the IPEC ISFSI and are not discussed further in this appendix. CoC Section 3.2.5 pertains to the MPC 32/32F, which is used at IPEC, and is discussed in Section F.4.3.1.1. Use of the "24" series MPC model would require a revision to this report.

##### F.4.3.1.1 Section 3.2.2 – MPC-32/32F

1. *Fuel cell pitch $\geq$ 6.43 inches*
2. *$^{10}B$ loading in the neutron absorbers: $\geq$ 0.0372g/cm$^2$ (Boral) and $\geq$ 0.0310g/cm$^2$ (METAMIC)*

The fuel cell pitch and $^{10}B$ loading of the neutron absorbers in the MPC are verified as part of MPC fabrication. Certification that each MPC meets these technical specification limits is provided by Holtec in the Component Completion Record (CCR) for each serial number MPC. The design of each MPC-32 is checked to ensure that it meets the specific design features for criticality. Each MPC is then manufactured and certified that it meets the design requirements.

##### F.4.3.1.2 Section 3.2.6 – Fuel Spacers

*Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.*

The IPEC Unit 2 fuel design is W 15x15, which is 159.8 inches long with an active fuel length of 144 inches. In accordance with IPEC Procedure 2-DCS-031-GEN a 8- inch to 10.5-inch upper fuel spacer is required, depending on Non-Fuel Component type, and a 4-inch lower fuel spacer is required for all locations.

#### F.4.3.1.3 Section 3.2.7 – METAMIC $B_4C$ Content

*The $B_4C$ content in METAMIC shall be $\leq$ 33.0 wt.%.*

The limit was verified to be met by Holtec International during the MPC fabrication process as documented in the Component Completion Records for the MPCs.

#### F.4.3.1.4 Section 3.2.8 - Neutron Absorber Tests

*Section 9.1.5.3 of the HI-STORM 100 FSAR is hereby incorporated by reference into the HI-STORM 100 CoC. The minimum $^{10}B$ for the neutron absorber material shall meet the minimum requirements for each MPC model specified in Sections 3.2.1 through 3.2.5 above.*

This CoC requirement was verified to be met by Holtec International during the neutron absorber fabrication process as documented in the Component Completion Records for the MPCs.

### F.4.3.2 Section 3.4 – Site-Specific Parameters and Analysis

#### F.4.3.2.1 Section 3.4.1 - Maximum Normal Ambient Temperature

*The temperature of 80°F is the maximum average yearly temperature.*

Maximum temperature data representative of the IPEC site were obtained from the National Climatic Data Center of NOAA. For New York City, located approximately 25 miles south of IPEC and at the same approximate elevation, the highest and lowest average annual temperatures between the years of 1909 and 2002 are 52.7°F (1998) and 46.8 °F (1958), respectively. Therefore, this requirement is met.

#### F.4.3.2.2 Section 3.4.2 – Ambient Temperature Extremes

*The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40°F and less than 125°F.*

Data used to establish ambient temperature extremes at the IPEC site were obtained from the National Climatic Data Center of NOAA. The highest temperature recorded in the state of New York was 108°F in Troy on July 22, 1926, which does not approach the 125°F 3-day upper limit. The lowest recorded temperature in the state of New York was -52°F in Old Forge on February 18, 1979. In New York City, the minimum recorded temperatures on February 17, 18, and 19, 1979 were 4˚F, -1˚F, and 17˚F, respectively. Because of the proximity of New York City to IPEC, these temperatures should be representative of the IPEC site and are well above the -40˚F 3-day lower limit. Therefore, this requirement is met.

### F.4.3.2.3 Section 3.4.3 - Seismic Criteria

*a. The resultant horizontal acceleration (vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site), $G_H$, and vertical ZPA, $G_V$, on the top surface of the ISFSI pad, expressed as fractions of 'g', shall satisfy the following inequality:*

$$G_H + \mu G_V \leq \mu$$

*where $\mu$ is either the Coulomb friction coefficient for the cask/ISFSI interface or the ratio r/h, where 'r' is the radius of the cask and 'h' is the height of the cask center-of-gravity above the ISFSI pad surface. The above inequality must be met for both definitions of $\mu$, but only applies to ISFSIs where the casks are deployed in a freestanding configuration. Unless demonstrated by appropriate testing that a higher coefficient of friction value is appropriate for a specific ISFSI, the value used shall be 0.53. If acceleration time-histories on the ISFSI pad surface are available, $G_H$ and $G_V$ may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality shall be evaluated at each time step in the acceleration time history over the total duration of the seismic event.*

*If this static equilibrium based inequality cannot be met, a dynamic analysis of the cask/ISFSI pad assemblage with appropriate recognition of soil/structure interaction effects shall be performed to ensure that the casks will not tip over of undergo excessive sliding under the site's Design Basis Earthquake.*

*b. For free-standing casks, under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will not result in cask tipover or cask a cask to fall off the pad. In addition, impact between casks should be precluded, or should be considered an accident for which the maximum g-load experienced by the stored fuel shall be limited to 45 g's.*

<u>μ as the Coefficient of Friction</u>

Calculations IP-CALC-05-00754 and FCX-00542 provide accelerations values located at the ISFSI pad surface based on the site ground response spectra and soil structure interaction effects. Using the maximum resulting accelerations at the top pad surface, the value of the inequality $G_H + \mu G_V \leq \mu$, *with* $\mu = 0.53$, *is 0.31 <0.53.*

<u>μ as the Ratio of Cask Radius to Center-of-Gravity</u>

There are several values of overpack radius and center-of-gravity in the HI-STORM 100S and 100S Version B overpack drawings (3669 and 4116) and FSAR Table 3.2.3, respectively. The approach is to use the values of 'r' and 'h' from the various 100S overpack designs that produce the highest value of 'μ':

r (max) = 133.875"/2 = 66.9" (for the HI-STORM 100S Version B overpack per Dwg 4116)
h (min) = 111.88 inches (for the HI-STORM 100S Version B(218) with loaded MPC-32)

r/h (max) = 66.9/111.88 = $\mu$ = 0.60, thus:

$G_H + \mu (G_V) \leq \mu$

From calculation IP-CALC-05-00754, $G_X$ = 0.17 , $G_Y$ = 0.16 , $G_Z$ =0.16(Vertical)

$G_H = \sqrt{(G_X)^2 + (G_Y)^2} = \sqrt{(.17)^2 + (.16)^2} = 0.23$

0.23 + 0.60(0.16) = 0.33

0.33is less than 0.60 and is acceptable for free-standing casks

The IPEC site-specific seismic criteria are acceptable under both evaluations of 'μ' required by the HI-STORM CoC.

Cask Sliding Evaluation

The ISFSI Pad surface is level with the adjacent grade on all four sides consequently if a HI-STORM cask translates horizontally during a seismic event, a tip over event is not probable. In addition, an analysis (Calculation IP-RPT-07-00200) was performed to determine the maximum displacement of any HI-STORM 100S cask during an earthquake considering icy conditions on the ISFSI pad at IPEC. This analysis demonstrated that the maximum displacement is smaller than the free space between adjacent casks and between the outer casks and the edges of the ISFSI pad.

**F.4.3.2.4 Section 3.4.4 - Flood**

*The analyzed flood condition of 15-fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.*

The concrete ISFSI pad is situated such that no significant storm rain water accumulation on the surface of the pad can occur. The pad is designed with a slight slope, includes under-pad drainage provisions, and adjacent roadways all slope away from the pad. Local topography allows storm runoff to flow away from the pad in the south, west, and north directions. The cask air inlet duct opening used in the thermal analysis is measured from the top of the overpack base plate. The casks to be deployed at the IPEC ISFSI have a 1-inch thick base plate. No significant ponding on the ISFSI pad is expected. Therefore, rain water will not cause blockage of the cask inlet vents.

The HI-STORM overpack is analyzed for flood effects as shown in HI-STORM FSAR Table 2.2.8. The analyzed submergence depth for the MPC is 125 feet, consistent with its dual-purpose function as part of a 10 CFR 71-certified transport system. The analyzed flood velocity for the HI-STORM 100 System is 15 ft/s.

As described in Section F.1.5, the most severe hypothetical flooding condition at the IPEC results in a maximum water elevation of 15-feet above MSL. The elevation of the IPEC ISFSI pad is approximately 90 feet above MSL. Therefore, flooding of the casks during storage on the ISFSI pad is not a concern.

### F.4.3.2.5 Section 3.4.5 - Fire and Explosion

*The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.*

The HI-STORM FSAR postulated fire event for the overpack was performed using the following key inputs, as described in HI-STORM FSAR Section 11.2.4.2.1:

1) A diesel fuel volume of 50 gallons maximum,
2) The HI-STORM overpack engulfed in flame for 3.622 minutes, and
3) A flame temperature of 1475 °F

The generic overpack fire analysis shows that the fuel cladding temperature, MPC internal pressure, and overpack outer shell steel temperature all remain below their respective short term temperature limits.

The IPEC fire hazards evaluation is based on the HI-STORM overpack and not the HI-TRAC transfer cask because the MPC is transferred into the HI-STORM overpack prior to embarking on the transport route at IPEC. Flammable fuel will only be in the vicinity of the overpack during transport from the FSB to the ISFSI pad and at the pad while the cask is being placed in its designated position.

The fire protection requirements for the ISFSI are contained in 10 CFR 72.122(c), Subpart F, General Design Criteria, *Protection Against Fires and Explosions*. A review of the IPEC Fire Protection Program reveals that structures, systems and components important to safety are designed and located such that they can continue to perform their safety function effectively under credible fire exposure conditions. A fire hazards evaluation and an explosion hazards risk assessment (Calculation No. IP2-RPT-03-00015) have been performed by Enercon Services, Inc. for the transport of spent fuel in HI-STORM overpacks to the ISFSI and storage of the overpacks on the ISFSI pad. All fire hazards are evaluated in comparison to the design basis overpack fire event as described in Chapter 11 of the HI-STORM FSAR.

The methodology employed during development of the ISFSI fire hazards evaluation consisted of the following steps:

1) Identify the design basis fire as established in Holtec HI-STORM Final Safety Analysis Report ,

2) Identify the travel path for the HI-STORM overpack and the location of the ISFSI storage pad,

3) Identify all credible fire sources,

4) Evaluate the potential impact of each credible fire source on the HI-STORM overpack.

Some of the fire sources were eliminated based on established administrative controls or adequate shielding. The remaining fire sources were evaluated using either a comparison of total combustible energy content (if the total is below that of the design basis fire) or evaluated using standard heat transfer techniques to quantify potential heat addition to the HI-STORM overpack during its transport and its permanent residence at the ISFSI pad.

The short-term maximum overpack outer shell steel temperature limit of 600° F was compared to a steady-state surface temperature calculation methodology. A steady-state temperature profile was considered to be a conservative assumption. Any finite fire duration would be expected to result in a lower temperature. The HI-STORM FSAR states in Section 11.2.4.2.1 that the time constant for the overpack is 127.7 hours, or approximately five days. This implies that significant heat from a fire would not penetrate the thick concrete walls during any realistic time estimate for a site fire. The types of fires evaluated in the fire hazards analysis could be expected to be mitigated by either being extinguished, moving the cask hauler away from the fire source or otherwise shielding the casks from the fire, within one day, or one tenth of the thermal time constant. Therefore, it was conservative to assume steady-state surface temperatures and acceptable to compare them to a short term criteria.

The design basis fire is an engulfing fire around the overpack that results when the diesel fuel contents of a hypothesized VCT fuel tank are spilled around the overpack and assumed to burn in place. The combustion material is 50 gallons of diesel fuel, which is assumed to burn for 3.622 minutes at 1475°F. The ambient temperature is assumed to be 100°F. The resulting calculation shows that the HI-STORM outer shell reaches 570°F, which is below the 600°F short-term temperature limit for the outer shell steel specified in HI-STORM FSAR Table 2.2.3. Knowing that the thermal energy content of diesel fuel is 130,000 BTU/gal, the energy content of the design basis fire can be calculated. Therefore, the energy involved is 6.5 MBtu. This worst case design basis fire bounds any engulfing or non-engulfing fire involving combustible material with less than this energy content.

The results of the evaluation concluded that all potential fire hazard exposures presented an acceptable risk. Some of the exposures were determined to be non-credible sources of fires during the limited time involved in cask transfer. Others were evaluated as being bounded by the design basis fire in terms of total energy content, and therefore being acceptable. The rest were evaluated for their impact on the overpack surface temperatures using heat transfer equations and conservative assumptions.

Appendix C.2 of NUREG-1864 (Reference: "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant, NUREG-1864, A. Milliakos, et. al., March 2007) also concludes the effects of a fire more severe than the Holtec Design Basis Fire does not result in fuel or canister damage.

Since the ISFSI pad and overpack transport pathway is exterior to and sufficiently separated from IPEC plant structures, no automatic fire detection or suppression systems were incorporated into the design of the IPEC ISFSI. Several yard fire hydrants are installed along the transport pathway. The fire hydrants along with the trained Fire Brigade provide sufficient capacity and capability to minimize the adverse effects of a fire on the overpacks and all associated components. The overpacks are designed so that no adverse effects will results due to fire suppression activities.

Several potential explosion hazards have been identified near the haul path of the HI-STORM overpack as it is moved from the FSB to the IPEC ISFSI. Although the explosive content of some of the hazards near the haul path are potentially large enough to damage the overpack, the risks of an explosion during the brief periods of transport have been calculated using probabilistic risk assessment methods to be not credible (Calculation No. IP2-RPT-03-00015). In addition, some potential explosion hazards, such as the hydrogen supply trailer, will be relocated away from the haul path during transport operations. Also, potential explosion hazards are a sufficient distance from the ISFSI pad so as to not present a hazard to stored overpacks.

Therefore, risks associated with identified explosion hazards are acceptable.

**F.4.3.2.6 Section 3.4.6 - Cask Drop and Tip-Over**

*For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to ≤ 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.*

In accordance with the HI-STORM CoC, IPEC has the option of constructing the pad to comply with specific limits set forth in the cask FSAR without performing a site specific cask drop analysis. IPEC has elected this option and the IPEC ISFSI pad is designed in accordance with the "Set A' requirements in HI-STORM FSAR Table 2.2.9. The HI-STORM FSAR describes the overpack non-mechanistic tipover and end drop design basis analyses that were performed using LS-DYNA3D. These analyses are enveloped by the IPEC-performed analyses and construction of the pad.

The overpack will be lifted at the ISFSI by the VCT. The VCT load links, overhead beam, and load supporting members (whose failure result in a drop of the load) of the vehicle frame meet the material selection and stress requirements of ANSI N14.6. VCT Lifting Towers (Jacks) comply with ASME B30.5-1994. The VCT is equipped with locking pins which, together with the hydraulic lift system, provide redundancy for supporting the load during travel. In accordance with Technical Specification 5.5 (a)(3), with the locking pins installed, the TRANSFER CASK and OVERPACK may be lifted to any height necessary during transport between fuel building and/or the ISFSI. While the HI-STORM is being lifted from the LPT and being lowered onto the ISFSI pad by the VCT hydraulic system, the redundant locking pins are not engaged. A failure of the hydraulic system could, during this short duration evolution, result in a drop of the HI-STORM from a height of 11 inches. This is within the Technical Specification limits specified in TS Table 5-1. The 11 inch maximum lift height provides the clearance necessary to remove the HI-STORM overpack from the LPT.

**F.4.3.2.7 Section 3.4.7 - Berms and Shield Walls**

*In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.*

Berms or shields walls are not required or used at the IPEC ISFSI and are not credited in the shielding analysis performed to demonstrate compliance with 10 CFR 72.104(a).

**F.4.3.2.8 Section 3.4.8 - Minimum Working Area Ambient Temperature**

*LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures $\geq$ 0°F.*

IPEC procedures 2-DCS-008-GEN, "MPC Loading & Sealing Operations," 2-DCS-009-GEN, "MPC Transfer & HI-STORM Movement," and " MPC Unloading Procedure", 2-DCS -012-GEN restrict loading, transport, and unloading operations to temperatures greater than or equal to 0 °F. Also, IPEC procedure 2-DCS-006-GEN, "VCT Operation " restricts VCT operations with the HI-STORM overpack to temperatures greater than or equal to 0°F and a maximum temperature of 100°F. Working area ambient temperature requirements for operation of the FSB single-failure-proof gantry crane are provided in procedure 2-DCS-026-GEN, "Unit 2 Crane Operations"

**F.4.3.2.9 Section 3.4.9 – Cask Air Duct Blockage for Extended Period**

*For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e., longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal in available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.*

There are no postulated site-specific design basis events (e.g., floods, mud slides) that could potentially result in the blockage of any HI-STORM inlet or outlet air ducts for an extended period of time (see Section F.4.3.2.4 of this appendix for more detail on flooding). Therefore, this CoC requirement is not applicable to the IPEC ISFSI.

**F.4.3.2.10 FSAR Table 1.0.3/2.V.2(b)(3)(f)- Other Natural Phenomena**

*10CFRPart 72 identifies several other natural phenomena events (including seiches, tsunami, and hurricane) that should be addressed for spent fuel storage.*

Seiches are phenomena unique to fairly large lakes and similar bodies of water, e.g. Lake Michigan, and the IPEC site is not susceptible to Seiches. Tsunami events are unique to open ocean coastlines and the IPEC site location approximately twenty miles up river from New York harbor protects the site from the effects of Tsunami.

The effects of Hurricanes on the site are discussed in section 2.5 in the Unit 2, and also in the Unit 3, FSAR. The maximum flood height is 15 above MSL as discussed in Section F.1.5 of this report. The maximum probable wind speed at the site has been determined to be 90 miles per hour as reported in the IPEC FSARs. The flooding effects and the wind effects of a hurricane are enveloped by the other postulated floods and wind phenomena, e.g. Tornado, as discussed in Section F.1.5 of this report

### F.4.3.3 Section 3.6 – Forced Helium Dehydration System

#### F.4.3.3.1 Sections 3.6.1 and 3.6.2 – System Description and Design Criteria

*Use of the Forced Helium Dehydration (FHD) system, (a closed-loop system) is an alternative to vacuum drying the MPC for moderate burnup fuel (≤ 45,000 MWD/MTU) and mandatory for drying MPCs containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.6.2.*

The fuel to be loaded in the initial 2007/2008 loading campaign in the HI-STORM 100 System and stored at the IPEC ISFSI is all burned less than 45 GWD/MTU and is, therefore, not high burnup fuel. In accordance with the HI-STORM CoC, Appendix A, LCO 3.1.1, either vacuum drying or forced helium recirculation may be used to dry MPCs containing all moderate burnup fuel (i.e., burnup ≤ 45,000 MWD/MTU). Entergy is choosing to use the FHD system to dry the MPCs currently planned for storage at the IPEC ISFSI. The FHD system being used at IPEC is designed in accordance with design criteria in Section 3.6.2 of Appendix B to the HI-STORM CoC. The equipment has been certified that it meets these requirements and performance testing was performed to document compliance.

The acceptance criterion for the FHD system is gas temperature exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes or gas dew point exiting the MPC shall be ≤ 22.9°F for ≥ 30 minutes. This is verified by ensuring the dew point of the gas stream is equal to or below 22.9°F by measurement. A dew point temperature of 22.9°F or less is equivalent to a vapor pressure of equal to or les than 3.0 torr. Operation of the FHD system is governed by IPEC procedure 0-DCS-023-GEN.

#### F.4.3.3.2 Section 3.6.3 – Fuel Cladding Temperature

*A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.*

In accordance with the HI-STORM 100 CoC and FSAR, an analysis (Holtec document HI-2022966) was performed demonstrating compliance with all design criteria in FSAR Chapter 2 and the requirements of HI-STORM CoC Appendix B, Sections 3.6.2 and 3.6.3. The FHD system was shown to satisfy all design criteria requirements. Therefore, the FHD design is in compliance with the CoC requirements.

#### F.4.3.3.3 Section 3.6.4 – Pressure Monitoring During FHD Malfunction

*During an FHD malfunction event, described in HI-STORM 100 FSAR Section 11.1 as a loss of helium circulation, the system pressure must be monitored to ensure that the conditions listed therein are met.*

The FHD System is equipped with pressure gauges to ensure that this requirement is met and also with safety relief devices to prevent the MPC structural boundary pressures from exceeding the design limits. The MPC is filled with sufficient helium to maintain the fuel in an analyzed condition while actions are taken to return the FHD to service. When the FHD is operable, the MPC helium pressure may be decreased to allow the FHD to operate.

### F.4.3.4 Section 3.7 – Supplemental Cooling System

*The SCS is a water circulation system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. Use of the Supplemental Cooling System (SCS) is required for post-backfill HI-TRAC operations of an MPC containing one or more high burnup (> 45,000 MWD/MTU) fuel assemblies. The SCS shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.7.2.*

The HI-STORM casks loaded at IPEC will not contain any fuel assemblies burned greater than 45,000 MWD/MTU. Thus, the supplemental cooling system is not required at this time. This appendix will require revision to address the use of a supplemental cooling system prior to loading any Unit 2 or Unit 3 fuel assemblies classified as high burnup fuel.

### F.4.3.5 Section 3.8 – Combustible Gas Monitoring During MPC Lid Welding

*During MPC lid welding operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present in the welding area.*

A risk of hydrogen production and a flammable atmosphere could exist inside the MPC due to oxidation of neutron absorber panels while the MPC is filled with water. Upon MPC lid installation, any gas generated would be trapped in the gas space under the lid created when the MPC water level is lowered to facilitate lid welding. Purging of the space under the MPC lid is performed prior to pre-heating, welding or grinding operations per procedure 2-DCS-008-GEN, "MPC Load & Seal Operations." Continuous sampling for combustible gas buildup is performed until the root layer of the MPC lid-to-shell weld, including NDE, is complete. Continuous sampling is also maintained during any repairs to the root weld, if required. If completion of the root weld of the MPC is interrupted for any reason, combustible gas concentration is be verified to be < 50% of the lower explosive limit (LEL) prior to continuing welding.

During unloading operations, sampling of the MPC internal atmosphere occurs prior to penetration to the cask internals in the unloading sequence per procedure 2-DCS-012-GEN, "MPC Unloading." The weld cutting process is not expected to be an ignition source due to low temperature and no sparks, the cask will be vented during the refill sequence, and any gases in the cask should be expelled from the cask with introduction of water. Without any gases in the cask, combustion of the hydrogen will not be possible even if an ignition source were to be available. With helium in the cask initially, a hydrogen burn cannot occur due to the lack of oxygen to initiate and sustain the burn. The primary defenses at IPEC for flammable gases during unloading are first, minimization of gases by venting and purging, and second, the exclusion of ignition sources.

**SECTION F.5**
**COMPLIANCE WITH HI-STORM 100 CASK SYSTEM**
**SER AND FINAL SAFETY ANALYSIS REPORT**

*10 CFR 72.212(b)(3):Review the Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in paragraph (b)(2) of this section.*

The following documents Entergy's review of the NRC's Safety Evaluation Report through CoC Amendment 2 and HI-STORM FSAR Revision 4. Any divergence from the SER or FSAR descriptions, methodologies or practices is identified. All described deviations have been evaluated under the IPEC 10 CFR 72.48 process, as applicable. Changes made by Holtec generically or site-specifically for IPEC are discussed in Section F.6.

### F.5.1 SER and FSAR Chapter 1, General Description

| SECTION | REQUIREMENT | CHANGE DISCUSSION |
|---|---|---|
| **SER FOR CoC REVISION 0** | | |
| 8.1.3 | Hydrostatic Test | The SER states that the MPC is backfilled with helium on top of the spent fuel pool water for applicable leak testing and then filled with water for the hydrostatic test. At IPEC, the hydrostatic test will be performed, and the cask drained, dried, and backfilled with helium prior to the helium leak test of the vent and drain port cover plates as described in SAR Section 8.1.5. |
| **SER FOR CoC REVISION 1** | | |
| NA | | No deviations or discussion required. |
| **SER FOR CoC REVISION 2** | | |
| 6.3.2 | Licensee must perform tests on Metamic with $B_4C$ concentration above 15% prior to use. | IPEC will not be performing tests on METAMIC. The CoC holder, Holtec, is required to perform these tests. |
| **CHAPTER 1** | | |
| NA | | No deviations or discussion required. |

### F.5.2 FSAR Chapter 2, Principle Design Criteria

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 2.0.4.1 | Safety Classification of ISFSI pad and applicability of QA program | This HI-STORM FSAR section states that the ISFSI pad design and construction shall be performed in accordance with a Part 72, Subpart G-compliant QA program. The IPEC ISFSI pad is classified as NITS. Therefore, its design and construction are not required to be performed under the QA program. |
| 2.2.1.4 | Annual Average Soil Temperature | IPEC does not collect information on site annual average soil temperatures. However, HI-STORM FSAR Table 2.2.2 shows that the bounding annual average temperature is 77°F based on the highest reported annual value in the United States in Key West, Florida. The maximum average annual air temperature in the IPEC area between the years of 1909 and 2002 was 52.7°F in 1998 (see Section F.4.3.2.1). Clearly, this ambient air temperature correlates to an annual average soil temperature lower than the 77°F bounding temperature specified in the HI-STORM FSAR. |

### F.5.3 Chapter 3, Structural Evaluation

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 3.4.11.1 | ISFSIs located in areas subject to conditions that degrade casks | The HI-STORM FSAR notes that the licensee will evaluate site conditions that may degrade storage casks to determine inspection frequency. The IPEC site is bounded by the environmental conditions evaluated in the FSAR. HI-STORM overpack material stored on the ISFSI pad is compatible with the operating environments listed in Table 3.4.2. The IPEC site is not located in a salt water/air environment. Therefore the inspections described in Chapter 9 of the FSAR and implemented by IPEC in procedure 0-DCS-011-GEN. "HI-STORM Inspection," are sufficient. |

### F.5.4 FSAR Chapter 4, Thermal Evaluation

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 4.5.6 | HI-TRAC Pressure Relief Valves | The HI-TRAC water jacket pressure relief valves are removed and the drain and vent plugs and the pressure relief valve tap plugs are installed prior moving the HI-TRAC (containing an empty MPC) from the cask work area to the spent fuel pit. The pressure relief valves are stored in a designated storage box to prevent potential damage or mishandling. After the HI-TRAC (containing a loaded MPC) is moved from the spent fuel pit to the cask work area, the two pressure relief valve tap plugs and the two vent plugs are removed from the HI-TRAC water jacket. The HI-TRAC water jacket is filled through the vents until the water level is about one-half inch below the relief valve ports. The HI-TRAC water jacket pressure relief valves and vent plugs are then installed. The pressure relief valves will then be available to prevent over pressurization during normal heat up and accident conditions after closure activities are complete. |

### F.5.5 FSAR Chapter 5, Shielding Evaluation

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| NA | | No deviations or discussion required. |

### F.5.6 FSAR Chapter 6, Criticality Evaluation

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| NA | | No deviations or discussion required. |

### F.5.7 FSAR Chapter 7, Confinement

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| NA | | No deviations or discussion required. |

### F.5.8 FSAR Chapter 8, Operating Procedures

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 8.0 | User-developed procedures and the design and operation of any alternate equipment must be reviewed by the certificate holder prior to implementation. | Entergy-developed procedures encompassing loading, storage, and unloading operations have been reviewed by Holtec prior to implementation. Holtec review of user-generated revisions to procedures does not provide significant increased assurance of compliance with the CoC, safety in loading, unloading, or storing the cask, or avoiding deviations from the intent of the FSAR. Therefore, Holtec review of revisions to these procedures is not considered necessary as long as the intent of the guidance in FSAR Chapter 8 and the CoC are met. |
| Table 8.0.1 | States that lifting devices are designed to ANSI N14.6 | The FSB single-failure-proof gantry crane and the HI-TRAC lifting yoke are designed in accordance with NUREG-0554 and NUREG-0612. Other lift components used with the HI-STORM 100 cask system meet ANSI B30.9 requirements, including the additional guidelines of NUREG-0612, Section 5.1.6(1)(b). |
| 8.1.5 | MPC Drain Down Time Limit | The maximum allowable time duration for fuel to be submerged in water contained in an MPC in the HI-TRAC prior to the start of drying operations is described in Section 4.5.1.1.5. Based on the design cask heat load, the combined thermal inertia of the loaded HI-TRAC, starting and limiting water temperatures, and assuming an adiabatic heat-up, Table 4.5.6 is provided in the FSAR. This table provides time allowable durations based on initial water temperatures. If there is insufficient time to perform wet transfer activities in accordance with the limitations in the table, forced water circulation shall be initiated to maintain MPC cavity water to less than 150°F. The minimum flow required can be calculated using the formula Mw = $Q/(T_{MAX}-T_{IN})$ with Q being the maximum |

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| | | thermal cask loading of 28.74 kW and $T_{MAX}$ equal to 212°F. However, instead of utilizing only the maximum thermal cask loading as the basis for time duration to begin drying operations, the formula $t_{max} = 7.62(212-T_i) / kW$ can be used to determine maximum allow time to start re-circulation flow. The formula is based on the Holtec calculated combined thermal inertia of the loaded HI-TRAC described in FSAR section 4.5.1.1.5. The kW term is the cask thermal loading in kilowatts, $T_i$ is the initial pool temperature in °F, and $t_{max}$ is the maximum time allowed in hours. The value 7.62 is the thermal inertia value in Btu from FSAR Table 4.5.5 converted to kW (26032/3417=7.62). |
| 8.1.5.2<br>8.1.5.3 | Prepare for MPC lid Welding<br>Weld the MPC Lid as follows: | To support the welding activities, weld procedures are qualified in accordance with ASME Section IX and Section III code requirements. The weld procedures fully meet Holtec International's welding specifications and Entergy's program requirements. The welding is performed using the vendor proprietary welding machine designed specifically for the lid to shell welding evolution. The procedures used have been reviewed in accordance with IPEC's protocol applicable to vendor procedures to be used on site and validated on MPC mockups during the preoperational dry runs. |
| 8.1.6 | The licensee is responsible for assessing and controlling floor loading conditions... | The truck bay area of the FSB has been extensively modified, a LPT system and Gantry Crane has been installed and the design of the work space floor has been designed and reconstructed considering the loading impressed by the HI-STORM 100 System components. |

### F.5.9 FSAR Chapter 9, Acceptance Criteria and Maintenance Program

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 9.1.1-4 | Inspection plan reviewed and approved by Holtec. | Inspection of the MPC closure welds performed at IPEC are in accordance with applicable Holtec drawings, the requirements of this FSAR section including qualification of inspector to SNT-TC-1A, and to the NDE requirements identified in FSAR Table 9.1.4. As documented in the Holtec HI-STORM 100 FSAR, confinement of radioactive material inside the MPC fuel cavity is assured by a combination of the redundant welds with inspection and testing techniques that include visual and dye penetrant examination of all field welds, pressure testing of the MPC, and helium leak testing of the vent and drain port cover plate welds. Therefore, with compliance to the FSAR requirements, Holtec review and approval of the weld inspection plan for welds performed at IPEC does not provide significant increased assurance of compliance with the CoC, safety in loading or storing the cask, or avoiding deviations from the intent of the FSAR. Therefore their additional review is not considered necessary. |
| 9.2.1 | Perform load test on the transfer cask trunnions annually | HI-STORM FSAR Section 9.1.2 describes initial HI-TRAC load testing requirements and compliance with ANSI N14.6 and NUREG-0612. ANSI N14.6 (6.3.1) requires that annual load testing be performed or (6.3.1. b) in cases where conditions allow, load testing may be omitted and dimensional checks, and nondestructive examination in accordance with Section 6.5 (PT or MT) shall be sufficient. NUREG-0612 recommends compliance with ANSI N14.6 for special lifting devices without additional requirements regarding load testing. Substitution of NDE for load testing is consistent with regulations and industry practices. In addition, |

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|---|---|---|
| | | inspections of the trunnions are performed in accordance with IPEC PMs, work orders, and/or procedures prior to each use. This departure from the requirements specified in the Holtec FSAR has been evaluated under 10 CFR 72.48 and approved by OSRC. Meeting # 07-022 |

### F.5.10 FSAR Chapter 10, Radiation Protection

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| NA | | No deviations or discussion required. |

### F.5.11 FSAR Chapter 11, Accident Analysis

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| 11.2.1.4, 11.2.2.4, and 11.2.3.4 | Special handling procedures will be developed by the ISFSI operator to upright the HI-TRAC/overpack after a handling or tipover event | The FSAR states that "special" procedures will be developed. "Special" meaning tailored to the as-found condition after the event. IPEC Health Physics will perform a radiological assessment of the area in the event of a tip-over accident and take appropriate actions in accordance with site radiation protection procedures. IPEC will then assess the damage and develop a special handling procedure to upright the HI-TRAC or overpack taking into account all radiological and environmental conditions. |

### F.5.12 FSAR Chapter 12, Operating Controls and Limits

| SECTION | Requirement | DISCUSSION |
|---|---|---|
| 12.2.2 | Dry Run Training On Receipt Inspection Of Cask Components | Receipt inspection of cask components consists of several phases. First is a fabrication source inspection performed by the Entergy Supplier QA group prior to the components leaving the shop based on procurement documents. The next phase is the site receipt inspection that checks for completeness of order and shipping damage. Last is the pre-use cleaning and equipment checkout controlled by procedures 2-DCS-003-GEN "HI-STORM Receipt Inspection," 2-DCS- |

| | | |
|---|---|---|
| | | 002-GEN "HI-TRAC Receipt Inspection," and 2-DCS-001-GEN "MPC Receipt Inspection." These procedures incorporate the inspection requirements noted in Chapter 9 of the FSAR. Training is provided to those individuals responsible for execution of the procedure requirements. No specific dry run training on receipt inspection of cask components will be performed. The MPC and HI-TRAC transfer cask will be inspected and fit-up tests performed, however, as part of the loading operations dry run. . |

**F.5.13 FSAR Chapter 13, Quality Assurance**

| SECTION | REQUIREMENT | DISCUSSION |
|---|---|---|
| NA | | No deviations or discussion required. |

## SECTION F.6
## 72.48 REVIEWS & OUTSTANDING
## CASK LICENSING BASIS DOCUMENT CHANGES

The primary licensing documents of record used in this 72.212 evaluation report are the HI-STORM CoC Amendment 2, issued in June 2005, and Revision 4 of the HI-STORM 100 System FSAR, issued in April 2006.

**License Amendment 3 has was approved by the USNRC in May 2007. This amendment does not apply to the Unit 2 2007/2008 fuel loading campaign or the design and fabrication of systems, structures, components, or procedures used during the campaign. .**

### F.6.A HOLTEC GENERIC HARDWARE CHANGES

### F.6.B HOLTEC GENERIC CHANGES TO FSAR REVISION 4

| FSAR CHANGE INDEX FILTER RESULTS | | | | |
|---|---|---|---|---|
| **Chapter Number** | **FSAR Section** | **Holtec 72.48 No.** | **Affected Revision** | **Notes** |
| 2 | Table 2.2.1 | 831 | 4.00 | Modified 1/5/2007 |
| 3 | 3.1.2.3 | 765 | 4.00 | Modified 9/19/2006 |
| 3 | Table 3.1.18 | 765 | 4.00 | Modified 9/19/2006 |
| 3 | 3.4.4.3.3.3 | 812 | 4.00 | Modified 1/5/2007 |
| 3 | Table 3.4.5 | 812 | 4.00 | Modified 1/5/2007 |
| 3 | Table 3.4.9 | 812 | 4.00 | Modified 1/5/2007 |
| 3 | 3.4.7.2 | 831 | 4.00 | Modified 1/5/2007 |
| 8 | Table 8.1.5 | 812 | 4.00 | Modified 9/7/2006 |
| 1 | Dwg 3927 | 718 Rev 1 | 5.00 | |
| 3 | 3.4.4.3.3.3 | 812 Rev 2 | 5.00 | |
| 1 | Dwg 3927 | 816 Rev 1 | 5.00 | |
| 1 | Dwg 4116 | 833 | 5.00 | |
| 1 | Dwg 3996 | 837 | 5.00 | |
| 3 | 3.1/3.3 | 838 | 5.00 | |
| | | | | |

### F.6.C SITE-SPECIFIC IPEC HARDWARE CHANGES

1. The opening in the top lid of the HI-TRAC has been enlarged from a diameter of 27" to 33" to preclude interferences with the MPC lift rigging. The top lid was shipped back to the fabrication shop and this modification was performed by Holtec.

2. Weld protrusions on the HI-TRAC upper flange were ground down to preclude interferences between the HI-TRAC and the MPC's. This modification was performed on site by Holtec. The manufacturing tolerances of the MPC's and the HI-TRAC are such that very little margin exists between the outer diameter of the MPC's and the inner diameter of the HI-TRAC.

ATTACHMENT N

to

ATTACHMENT 1

**Holtec Report HI-2084118 Rev 2, "Shielded Transfer Canister Structural Calculation Package"**

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

[PROPRIETARY TEXT REMOVED]

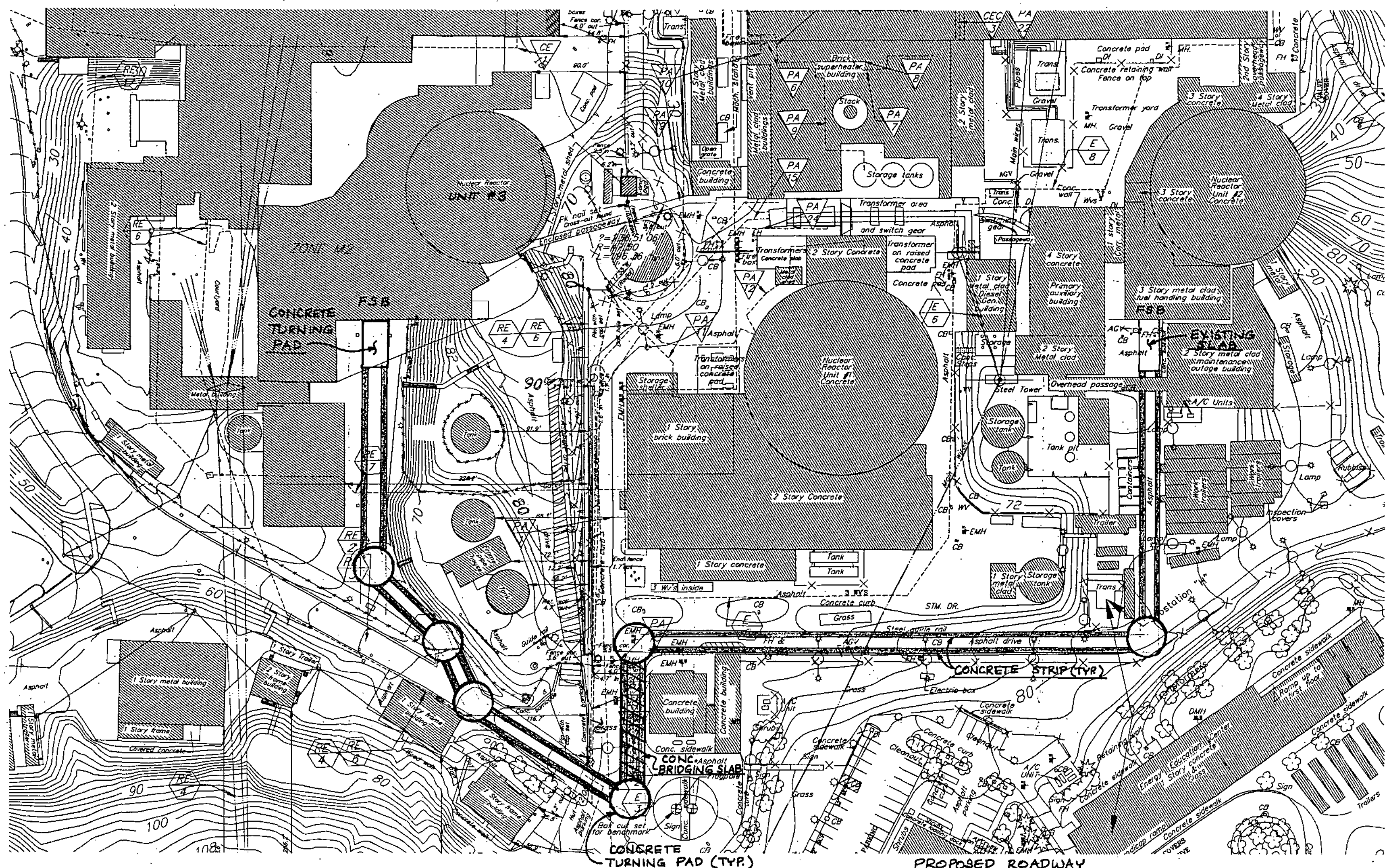
ATTACHMENT O

to

ATTACHMENT 1

**Proposed haul path design for fuel transfer operation**

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




PROPOSED ROADWAY
08-18-'09
J.O. 13634201

ATTACHMENT P

to

ATTACHMENT 1

## MCNP Sample Input file for STC regionalized loading

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

[PROPRIETARY TEXT REMOVED]

ATTACHMENT Q

to

ATTACHMENT 1

**Holtec Report HI-2022847 Rev 5, "Spent Nuclear Fuel Source Terms"**

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

[PROPRIETARY TEXT REMOVED]