STATE OF VERMONT
PUBLIC SERVICE BOARD

Petition of Entergy Nuclear Vermont Yankee,	)	
LLC, and Entergy Nuclear Operations, Inc.,	)	
For a Certificate of Public Good Pursuant to	)	
30 V.S.A. § 248 and 10 V.S.A. § 6522 to	)	PSB Docket No. 8300
Construct a Second Independent Spent Fuel	)	
Storage Installation ("ISFSI") at the Vermont	)	
Yankee Nuclear Power Station	)	

SUPPLEMENTAL PREFILED TESTIMONY AND EXHIBITS OF GEORGE THOMAS

STATE OF VERMONT
PUBLIC SERVICE BOARD

Petition of Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc., For a Certificate of Public Good Pursuant to 30 V.S.A. § 248 and 10 V.S.A. § 6522 to Construct a Second Independent Spent Fuel Storage Installation ("ISFSI") at the Vermont Yankee Nuclear Power Station	)	PSB Docket No. 8300

SUPPLEMENTAL PREFILED TESTIMONY AND EXHIBITS OF GEORGE THOMAS

Q1. Please state your name, occupation and business address.

A1. My name is George Thomas. I am employed by Entergy Nuclear Operations, Inc. as a Senior Project Manager at the Vermont Yankee Nuclear Power Station (or "VY Station"). I am the Project Manager responsible for managing the engineering and construction schedule and cost aspects of the VY Station's proposed second dry fuel, or Independent Spent Fuel Storage Installation (or "ISFSI"), storage pad, which I will refer to as the "Second ISFSI" or the "Project." My business address is P.O. Box 250, 320 Governor Hunt Road, Vernon, Vermont.

Q2. Did you previously submit prefiled testimony in this matter?

A2. Yes. I submitted prefiled testimony dated June 30, 2014.

Q3. What is the purpose of your supplemental testimony?

A3. The purpose of my supplemental testimony is to update my previous testimony to describe several changes to the Project since Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (together, "Entergy VY") submitted their petition in this matter on June 30, 2014, and to explain the engineering studies completed by Entergy VY since that time. In my previous testimony, I indicated that engineering calculations associated with the seismic response of the Second ISFSI storage pad were being finalized and that there was a possibility that the location of the pad may have to be adjusted based on those calculations. In October 2014, Entergy VY reported to the Public Service Board that it was in the process of completing ongoing engineering studies that had a potential to alter the design of the Project. The purpose of my supplemental testimony is to update my previous testimony, explain the engineering studies completed by Entergy VY and to describe several changes to the Project.

Q4. Please describe the engineering analysis and design work completed on the Second ISFSI storage pad since your initial prefiled testimony was submitted on June 30, 2014.

A4. As mentioned, at the time of my June 30, 2014 testimony the engineering calculations related to the seismic design of the pad were being finalized. Subsequently, Entergy VY concluded that the soil data taken adjacent to the pad's proposed location did not provide sufficient data to allow the engineers to conclude that the pad's design would meet U.S. Nuclear Regulatory Commission (or "NRC") requirements codified at 10 C.F.R. Part 72. Therefore, the decision was made to perform additional analyses of the soil conditions.

Beginning in early September 2014, Entergy VY performed nineteen soil soundings using cone penetration testing technology to more precisely measure the soil characteristics under and adjacent to the proposed pad location. The cone penetration data is collected by hydraulically pushing a steel cone containing instrument sensors into the soil and acquiring data at approximately 2 inch depth increments as the cone moves vertically below the surface. In addition, five conventional soil borings using a drill rig were performed, four of which were adjacent to soundings. The soil samples from the soil borings were analyzed by an off-site laboratory and their results confirmed the data from the soundings.

Entergy VY's consulting engineering firm, Sargent & Lundy, used the data from the soil soundings to perform engineering analyses based on the same design basis seismic loads that were used to analyze and design the first ISFSI pad, including Soil/Structure interaction, liquefaction, cask stability and cask sliding investigations. Their evaluations indicated that there are a few deep isolated "pockets" of soil of limited thickness beneath the location selected for the second ISFSI Pad that under design basis earthquake conditions were determined to be potentially liquefiable with a limited level of associated settlement. These potentially liquefiable pockets were incorporated and accounted for in the design of the reinforced concrete Second ISFSI storage pad.

Sargent & Lundy also developed an analytical model of the soil characteristics under and adjacent to the proposed location of the pad. The analytical model was utilized to evaluate the response of the pad during both static and dynamic conditions. The dynamic response included twenty-nine (29) Soil/Structure interaction analyses that verified that casks stored on the pad will not tip over or undergo excessive sliding under design basis earthquake conditions and that the pad would comply with the applicable requirements of 10 C.F.R. Part 72 and the NRC-issued Certificate of Compliance ("CoC") for the Holtec HI-STORM 100 dry storage system. The analysis was based on the same design basis seismic loads that were used to analyze and design the first ISFSI pad.

In summary, since the completion of soil testing, Sargent & Lundy has completed five engineering calculations utilizing the newly acquired soil data and has verified that the pad design meets applicable regulatory requirements. Sargent & Lundy completed its design analysis on May 1, 2015.

Q5. Based on the results of the updated engineering analyses, does Entergy VY propose any changes to the location or dimensions of the Second ISFSI storage pad?

A5. No. The location and dimensions of the Second ISFSI storage pad will not change from that originally proposed in June 2014. The updated engineering analyses determined that a pad with the original location and dimensions can be constructed in compliance with NRC regulations and the Holtec CoC given the soil conditions at that location.

Q6. Has the construction schedule changed as a result of the additional engineering analysis?

A6. Yes. Entergy VY previously planned to start construction in mid-July, 2015 and complete it prior to July 31, 2017. Assuming we receive the CPG requested in this proceeding and any other necessary approvals, our plan now is to begin construction in mid-June 2016 and complete construction in mid-November, 2017.

Q7. Will the new construction schedule impact Entergy VY's planned dry-cask loading campaigns?

A7. No. As noted in my June 30, 2014 testimony, Entergy VY plans to conduct two dry-cask-loading campaigns—one in 2019 and one in 2020—which will move all the fuel from the spent fuel pool to dry cask storage. The current plan is described in Entergy VY's updated Spent Fuel Management Plan, which I am sponsoring as Exhibit EN-GT-3. Prior to each loading campaign, the overpacks and canisters will be delivered to the site. Construction of the Second ISFSI storage pad must be completed before the end of 2017 so that the overpacks and empty canisters can be delivered and inspected, the overpacks filled with high density concrete and the overpacks and canisters (without fuel) placed on the pad during 2018. Once the overpacks are filled with high density concrete, they must be transported using the Vertical Cask Transporter which is designed to shuttle to and from the ISFSI pads.

If Entergy VY is not able to complete construction of the Second ISFSI storage pad before the end of 2017, the dry-cask loading campaign may be delayed. Based on Entergy VY's current Decommissioning Cost Estimate, the impact of delaying the transfer of all fuel onto the pads beyond 2020 is estimated at approximately $1.7 million per month. These costs include site security, maintenance and support staff, insurance, electricity, fuel oil, NRC fees, taxes and other costs that will be decreased or eliminated once fuel has been removed from the Spent Fuel Pool and the site Protected Area reduced. Entergy VY expects that these costs would be paid out of the plant's nuclear decommissioning trust fund, delaying the time when major decommissioning activities could begin.

Q8. Have there been any other developments with respect to the Project plans since your initial testimony was submitted?

A8. Yes. The Project design relating to the 200 kW security diesel generator has changed in several ways.

First, to meet the requirements of 10 C.F.R. § 73.55, the 200 kW diesel generator enclosure will be surrounded on three sides by a 16-foot high by 8-foot thick barrier wall. The wall will be approximately 59 feet long on the northwest side and 25 feet, 6 inches long on the northeast and southwest sides. The exterior surface of the wall will be Cor-Ten steel. Cor-Ten steel is corrosion-resistant and forms a stable, dark brown, rust-like appearance when exposed to weather.

Second, to accommodate the barrier wall, the location and orientation of the 200 kW diesel generator have changed somewhat. The 200 kW diesel generator enclosure has been rotated approximately 45 degrees counterclockwise and has been moved approximately 51 feet to the northeast. The diesel generator will be mounted in a metal enclosure/foundation base measuring approximately 12 feet wide by 35 feet long by 12 feet high and will have a ventilation hood with an approximately 52-inch overhang installed along the southeast side of the enclosure. The ventilation hood was previously designed to be located on the south side of the enclosure.

Finally, based on updated information provided by the vendor, the size of the fuel tank has been reduced from a 1,250 gallon above-ground double-wall fuel storage tank to a 1,200 gallon above-ground double-wall fuel storage tank surrounded by a 1,350 gallon rupture basin which contains a leak detection switch to detect a fuel oil leak from the storage tank.

The revised location and orientation, as well as the newly proposed barrier wall, are depicted on an updated site plan and location site plan sponsored by John Goodell as Exhibits EN-JG-7 and EN-JG-8, respectively.

Q9. Do you have any other updates to provide at this time?

A9. Yes. In my June 30, 2014 testimony, I discussed alternatives to the Second ISFSI storage pad considered by Entergy VY. One alternative considered was to use the Holtec HI-STORM 100U, an underground storage design by Holtec that was licensed by the NRC in December of 2009. At the time of my testimony, one facility planned to use an underground system. At this time, there are two facilities that plan to use or have begun to install underground systems. One facility, the Callaway Nuclear Plant, is located in Fulton, Missouri and has significantly different site conditions than the VY Station. Specifically, an area adjacent to the operating plant had been previously excavated for purposes of installing a second nuclear unit. The second unit was subsequently cancelled, leaving the excavated area vacant. Utilizing that area will greatly facilitate the installation of an underground system. The second facility is the San Onofre Nuclear Generating Station ("San Onofre"), a two-reactor site located in Southern California. It is my understanding that San Onofre selected an underground system based on site-specific conditions that are not applicable to the VY Station, such as site space limitations and tsunami protection, as well as commercial considerations.

Q10. Does the experience of other facilities change the evaluation of the Holtec 100U System for the VY Station?

A10. No. The experience of other facilities substantiates the conclusion that the cost to install an underground dry cask storage system at Vermont Yankee would be considerably more expensive than the above ground HI-STORM 100 system. Additionally, I understand that utilities suing the U.S. Department of Energy ("DOE") for breach of its contracts to remove spent fuel from their sites are required to take reasonable steps to mitigate the damages incurred as a result of the breach. It is therefore unlikely that the cost of a spent fuel storage system that is significantly more costly than another available alternative can be recovered from DOE. Entergy VY continues to believe that the HI-STORM 100U system not only would be significantly more difficult and substantially more expensive to install than the above-ground HI-STORM 100 system, but also carries significant schedule and cost risks associated with an unproven system.

Q11. Does this conclude your testimony?

A11. Yes.

Entergy Nuclear Vermont Yankee, LLC

SPENT FUEL MANAGEMENT PLAN

Revision 5

May 2015

This Spent Fuel Management Plan for the Vermont Yankee Nuclear Power Station ("VY" or "Station") was initially prepared and filed with the Vermont Public Service Board (the "Board") pursuant to the requirements of Chapter 157 of Title 10, Vermont Statutes Annotated, and in particular 10 V.S.A. § 6522 (b)(3), as interpreted by the Board in its Order and accompanying Certificate of Public Good ("CPG") issued on April 26, 2006, in Docket No. 7082. Section 6522 (b)(3) requires that prior to the issuance of a certificate of public good for a new spent-nuclear-fuel-storage facility, the Board shall find that Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (collectively referred to as "Entergy VY") "has developed and will implement a spent fuel management plan that will facilitate the eventual removal of those wastes in an efficient manner."

The Spent Fuel Management Plan ("Plan") is a living document subject to revision in accordance with the requirements of Chapter 157 as interpreted by the Board, changes in federal regulation, as well as changing circumstances affecting the subject matter of the Plan.

In submitting this report, Entergy VY notes that, notwithstanding the provisions of state law being applied in this case, 10 V.S.A. § 6522, the NRC has "exclusive authority over [commercial nuclear] plant construction and operation." The U.S. Supreme Court in *Pacific Gas & Electric Co. v. State Energy Resources Conservation and Development Commission*, 461 U.S. 190, 206-07 (1983) found that the NRC "was given exclusive jurisdiction to license the transfer, delivery, receipt, acquisition, possession and use of nuclear materials" and "[u]pon these subjects no role was left for the states." "Under the federal licensing scheme . . . it is not the states but rather the NRC that is vested with authority to decide under what conditions to license a [spent nuclear fuel] storage facility." *Skull Valley Band of Goshute Indians v. Nielson*, 376 F.3d 1223, 1250 (2004), *cert. denied*, 546 U.S. 1060 (2005). By submitting this report, Entergy VY does not waive any rights under federal law.

Entergy Nuclear Vermont Yankee, LLC

SPENT FUEL MANAGEMENT PLAN

May 2015

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LIST OF ATTACHMENTS

1.0 GENERAL

1.1 Definitions

- **72.212 Report** refers to the site-specific evaluation, required by 10 CFR § 72.212, that Entergy VY completed prior to operating the dry fuel storage ("DFS") system at the Station.
- **CAB** refers to the Containment Access Building at the Station.
- **CFR** refers to the Code of Federal Regulations.
- **CPG** means a "**Certificate of Public Good**" issued by the Vermont Public Service Board.
- **CTF** refers to a cask-transfer facility.
- **DOE** refers to the U.S. Department of Energy.
- **Full Core Offload** refers to the operation of removing all fuel assemblies from the reactor vessel. This could be required during the term of the operating license if a major reactor vessel repair or certain other maintenance were required. It will be required as an early step in the decommissioning process. It will be Entergy VY's goal to sequence dry cask loading campaigns such that Full Core Offload capability will be available during the license renewal period. Full Core Offload capability will continue to require the use of a temporary spare fuel rack.
- **FSAR** is an acronym for Final Safety Analysis Report. The FSAR for the SNF dry storage system was prepared by Holtec per requirements specified in 10 CFR Part 72, and is a compilation of information and analyses to support the NRC licensing review. The information provided in the FSAR includes a general description of the design of the dry storage system, its operation, and the supporting analyses for the demonstration of its performance under various operating, hypothetical accident and extreme environmental accident conditions, and its compliance with the regulatory requirements for the assurance of the public health and safety.
- **HOLTEC** refers to Holtec International.
- **HI-STAR** refers to Holtec's system used to transport SNF off-site.
- **HI-STORM Overpack** or "**Storage Overpack**" means the cask that receives and contains the sealed MPCs containing SNF. It provides the radiation shielding, ventilation passages, missile protection and protection against natural phenomena and accidents for the MPC. The HI-STORM overpack is

approximately 200 tons in weight, 11 feet in diameter, and approximately 19 feet high.

- **HI-TRAC transfer cask or HI-TRAC** means the transfer container used to house the MPC during MPC fuel loading, unloading, drying, sealing and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC, allowing loading operations to be performed while limiting radiation exposure to personnel.

- **Independent Spent Fuel Storage Installation** ("**ISFSI**" or "**Facility**") means a facility designed, constructed and licensed for the interim storage of SNF and other radioactive materials associated with spent-fuel storage in accordance with 10 CFR Part 72.

- **Multi-Purpose Canister** (or "**MPC**") means the sealed canister consisting of a honeycombed fuel basket for SNF storage, contained in a cylindrical canister shell.

- **NEI** is an acronym for the Nuclear Energy Institute.

- **Non-mechanistic tip-over** is a postulated cask tip-over event that is not based on any previously observed causal event or mechanism, but which is assumed in order to provide conservatism in risk assessments to prove that no fuel damage will occur as a result of the tip-over.

- **NRC** is an acronym for the U.S. Nuclear Regulatory Commission.

- **Reactor Building** refers to the reactor building at the Station.

- **Regulatory Guides** (or "Reg. Guides") are guidance documents drafted by NRC (and before the NRC, by the AEC) staff to provide guidance to licensees on implementing specific parts of the NRC regulations, techniques used by NRC staff in evaluating specific problems or technical issues.

- **SNF** is an acronym for Spent Nuclear Fuel.

- **Station** means the Vermont Yankee Nuclear Power Station.

1.2 General Facility Description

Use of the ISFSI or Facility involves several major areas at the Station owned by Entergy Nuclear Vermont Yankee, LLC, in Vernon, Vermont: the Reactor Building where the SNF is currently stored in the Station's spent-fuel pool; the CAB, the ISFSI pad (to which the fuel will be transported for interim storage in a Storage Overpack); and the transfer path between the CAB and the ISFSI pad. The ISFSI pad is a highly-engineered structure which has been designed and constructed to support loaded Storage Overpacks (which weigh about 200 tons each) and to ensure that no damage to the SNF occurs as a result of "non-mechanistic tip-over." The specific soil properties, soil depths, concrete properties and pad thickness have been thoroughly evaluated; the results are documented in the 72.212 Report and the ISFSI pad has been constructed to meet these standards. See Tab 2. The transfer path between the CAB and the ISFSI pad has been evaluated, and it has been determined that the existing roadway is adequate to protect underground utilities.

1.3 Location

The ISFSI pad is located north of the Reactor Building, approximately 210 feet west of the high water mark of the Connecticut River as shown on the Site Plan attached as Tab 1 to this Plan. As agreed in paragraph 1 of the Memorandum of Understanding between Entergy VY and the Vermont Department of Public Service dated June 21, 2005 ("DFS MOU"), a line-of-sight barrier has been constructed on the north and east sides of the ISFSI pad (See Tab 3).

1.4 Loading Campaigns

1.4.1 2008 Loading Campaign

During the summer of 2008, Entergy VY loaded 340 SNF assemblies into five Holtec Storage Overpacks and placed them on the ISFSI pad such that any one of the five casks can be retrieved without relocation of any of the other casks, providing access to individual casks. The 2008 loading campaign restored Full Core Offload capability.

1.4.2 2011 Loading Campaign

During the summer of 2011, Entergy VY loaded 272 SNF assemblies into four Holtec Storage Overpacks and placed them on the ISFSI.

1.4.3 2012 Loading Campaign

During the summer of 2012, Entergy VY loaded 272 SNF assemblies into four Holtec Storage Overpacks and placed them on the ISFSI. These four Overpacks (#10, #11, #12 and #13) were placed south of the Overpacks (#2, #3, #4 and #5) that were placed into storage on the ISFSI pad in an earlier campaign.

1.5 Future Loading Campaigns

As of January 2015, there are 2995 SNF assemblies remaining in the spent-fuel pool, configured so that high-decay-heat assemblies of SNF are surrounded by low-decay-heat assemblies of SNF, as agreed in paragraph 9 of the DFS MOU.

There are no SNF assemblies in the Reactor Core. The Planned Loading Schedule details VY's plan to reduce the number of fuel rods stored in the Spent Fuel Pool based on, among other factors, the normal cooling period for SNF after it is unloaded from the Reactor Core and efficient work scheduling. The current schedule for future loading campaigns is as follows:

PLANNED LOADING SCHEDULE		
Loading Campaign Years	Number of Casks to be Processed	SNF Assemblies Discharged to ISFSI Pad
2019	32	2176
2020	13	820*

*Includes one Damaged Fuel Container consisting of fuel debris

2.0 OPERATING PROCEDURES

2.1 Design and Operational Requirements

The Facility is sited, designed and operated in compliance with the applicable NRC licensing requirements found at 10 CFR Part 72, Subparts E, F & G, NRC Regulatory Guides and the FSAR for the SNF prepared by Holtec as well as the DFS MOU, the design and/or operational requirements set forth by the Board in its Order and CPG dated April 26, 2006, and such other requirements that were set forth in applicable permits from the Vermont Agency of Natural Resources or other governmental authority having jurisdiction over the Facility.

2.2 Facility Security

Security for the Facility is established and operated in compliance with the NRC licensing requirements found at 10 CFR Part 72, Subpart H and Part 73, and other applicable NRC Regulatory Guides.

2.3 Staffing

The operators of the Facility are trained in compliance with the training and certification requirements found at 10 CFR Part 72, Subpart I.

3.0 LONG TERM PLANNING FOR STORAGE OF SPENT NUCLEAR FUEL

3.1 Construction of Separate ISFSI Pad

The ISFSI described in Section 1.2 is an 8 x 5 array that was constructed to support 36 Storage Overpacks. (NOTE: Four storage locations are unused to allow retrieval of any casks should the need arise). As of permanent shutdown on December 29, 2014, VY will need 58 total Storage Overpacks to store the 3879 SNF assemblies (and one additional assembly for fuel debris) that were generated during VY's lifetime. Hence, an additional ISFSI pad is required. VY has submitted a CPG Request for a separate ISFSI pad to be constructed as part of decommissioning activities. All necessary regulatory approvals will be obtained prior to construction of this new ISFSI. The SNF remaining in the spent-fuel pool will be transferred to one of the two ISFSI pads and managed pursuant to the operating procedures outlined in Section 2.0 above, pending removal by the federal government. The DOE has not provided a schedule to accept SNF that would realistically preclude the need for the new pad.

The additional ISFSI pad has been designed and is planned to be located 30 feet to the west of the existing ISFSI pad.

3.2 License Renewal; Amendment of Spent Fuel Management Plan

Since VY was permanently shut down on December 29, 2014, there is no need to amend the Spent Fuel Management Plan for the obtained License Renewal.

3.3 Long Term Storage of Spent Nuclear Fuel at the Station

It is uncertain when DOE will begin taking fuel from the Station. In its April 26, 2006, Order and accompanying CPG in Docket No. 7082, the Board required VY to address the possibility that SNF could remain at the Station through 2082. Because of the uncertainty of future events over the relevant time period, it is expected that this section will need to be amended over time to address changed circumstances as they arise.

To address the Board's requirements in Docket No. 7082, Entergy VY will perform the following actions:

- Entergy VY will comply with all applicable NRC requirements for the storage of SNF in dry or wet storage until DOE meets its obligation under the federal law to take title to the SNF.
- By 2018 (ten years after the first fuel loading into an MPC/Overpack) Entergy VY will develop a formal inspection and maintenance program

for the MPC/Overpack assemblies with recognition that MPC/Overpack assemblies could be stored on site as long as 75 years.

- In advance of 2028 (twenty years after the first fuel loading into an MPC/Overpack), Entergy VY will undertake a program to seek renewal of applicable cask Certificates of Compliance as provided in 10 CFR 72.212 if the Certificate of Compliance has not been renewed by Holtec.
- If all SNF is not removed from the Station by 2047 (forty years after construction of the ISFSI pad), aging-management procedures for the ISFSI pad will be developed consistent with the NRC License Renewal requirements for concrete structures for reactors.
- If all SNF is not removed from the Station by 2048 (forty years after placing into service of the ISFSI temperature monitoring system) aging management procedures for the ISFSI pad temperature monitoring system will be developed consistent with the NRC License Renewal requirements for similar instrumentation systems for reactors.

4.0 TRANSFER AND CLOSURE PROCEDURES

4.1 General

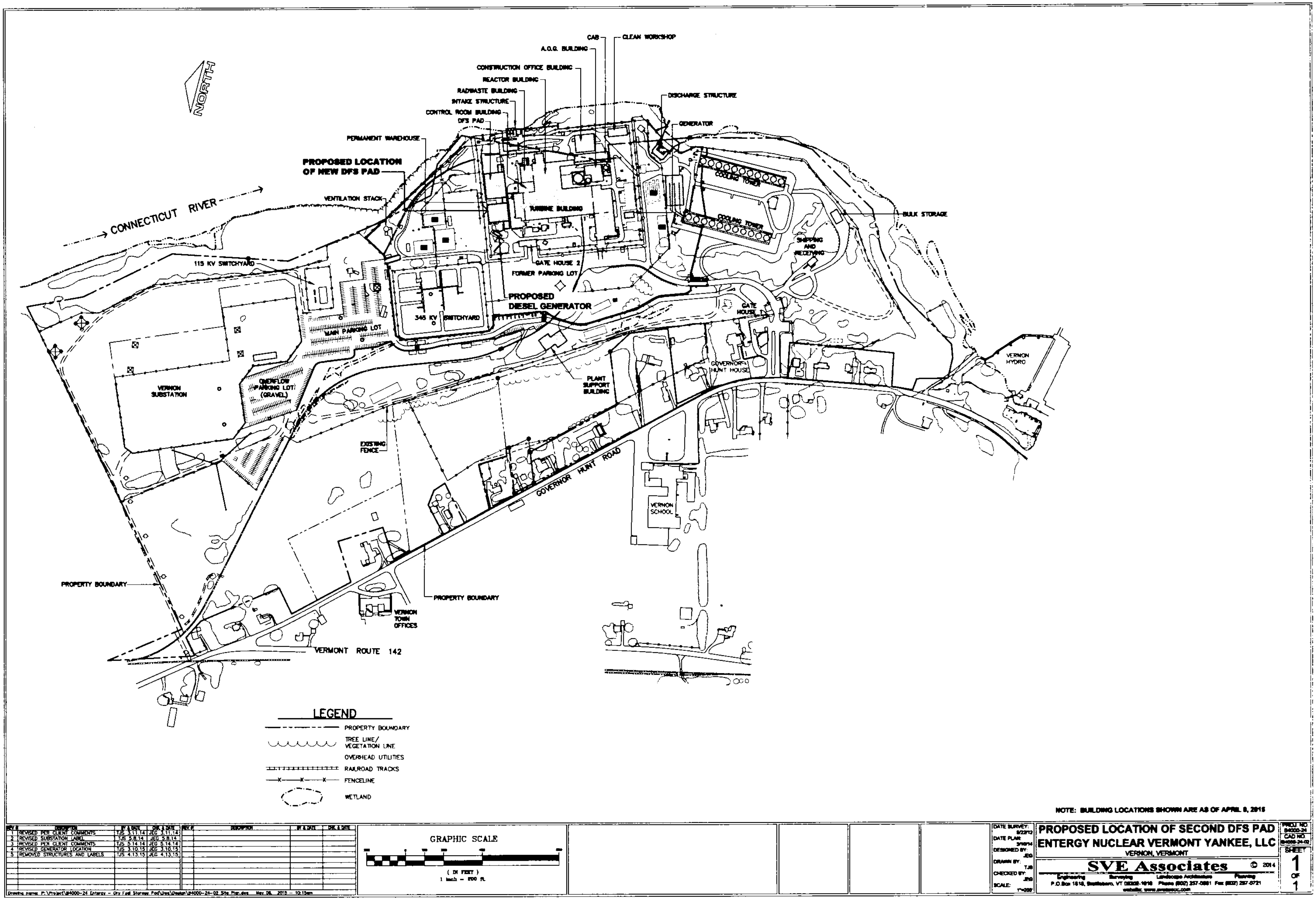
Under federal law, removal of SNF from the Station is the legal responsibility of DOE. VY has agreed in memoranda of understanding with the State of Vermont in Docket Nos. 6545 and 7082 to use its commercial best efforts to ensure that high-level SNF stored at the Station is removed from the site in a reasonable manner and as quickly as possible to an interim or permanent location outside of Vermont. Entergy VY, individually and through the NEI, also continues to work with state and federal officials and authorities to support the prompt implementation of a federal repository for SNF.

4.2 Transfer Procedure from ISFSI to Long-Term Repository

Entergy VY would likely construct a CTF to facilitate transportation of SNF off site. The transportation sequence would be as follows:

- The HI-TRAC would be placed over the Overpack;
- The MPC would be raised out of the Overpack and into the HI-TRAC;

- The HI-TRAC/MPC would be placed over the HI-STAR;
- The MPC would be lowered into the HI-STAR; and
- The HI-STAR would be removed from the chamber, if used, and readied for shipment.
- The HI-STAR would then be loaded on either specially designed rail or over-road transportation vehicles for transportation to the designated federal SNF repository.



TAB 1 - SITE PLAN

TAB 2

VERMONT YANKEE NUCLEAR POWER STATION

10 CFR 72.212 Report

Docket 72-0059

HI-STORM 100 SYSTEM

Licensing Basis Document

REVISION 5

Vermont Yankee Nuclear Power Station

VY 10 CFR 72.212 Evaluation Report

Docket 72-0059

Licensing Basis Document

HI-STORM 100 System

Revision 5

APPLICABLE SITES

ANO Unit 1:	☐	GGNS:	☐	W-3:	☐
ANO Unit 2:	☐	RBS:	☐	ECH:	☐
IPEC Unit 1:	☐	JAF:	☐	VY:	☐
IPEC Unit 2:	☐	VY:	X	WPO:	☐
IPEC Unit 3:	☐				

Safety-Related: ____ Yes
X No

Prepared by:	Brian Gutherman/ [signature] Responsible Engineer	Date:	10/22/14
Reviewed by:	Mike O'Brien/ [signature] Reviewer	Date:	10/29/14
Reviewed by:	Phil Couture/ [signature] Site 10 CFR 72.48 Coordinator	Date:	10/29/14
Approved by:	Ken Swanger/ [signature] Dry Fuel Storage Project Manager	Date:	10/29/14

RECORD OF REVISION

Issue Date	Revision Number	Affected Sections	Description
04/07/08	0	N/A	Initial Release
05/01/08	1	10.2, 10.7, Appendix B Appendix B Section 3.8, Appendix D	• LBDCR LIC 08-04 • Section 10.2 revised to enhance basis for tornado comparison • Section 10.7 added to address snow and ice • Appendix B Section 3.8 revised to note that monitoring for combustible gas will be perform during the entire MPC lid welding operation • Appendix D was revised as a result of a change to the Hazards Analysis
03/12/09	2	15, 18 Appendix B Appendix D	• LBDCR LIC 09-02 • Section 15 and 18 revised to reference EN-DC-212 • Section 18 revised to change revision of reference 31 from rev 2 to rev 3 due to EC 12270 which revised the hazards evaluation • Appendix B was revised to reference EN-DC-212 • Appendix D was revised due to the revised hazards analysis setback distances
5/31/11	3	8.3 12 14 18 Appendix B Appendix C Appendix D	• LBDCR LIC 11-01 • Sections 8.3 and 18 were revised to reflect results of new 72.104 Evaluation • Section 12 was revised due to changes to 10CFR73.55 • Section 14 was revised to include a table referencing the Engineering Report associated with each ISFSI component. • Section 18 was revised to reflect new hazards evaluation per EC 24349 • Section 18 was revised to reference EC 28665 for the stack-up restraints • Section 18 was revised to reference EC 28768 for the 2011 loading campaign • Re-titled Appendix B to "List of Procedures" and identified additional related procedures • Procedures were removed from Section 18 to eliminate duplication with Appendix B • Appendix C was revised to reflect that VY will perform temperature monitoring and annual inspections of the vent screens • Appendix C and section 18 were revised based on the final disposition of the MPC Helium leak test issue • Appendix C was revised to discuss disposition of applicable HIBs • Appendix C was revised to reflect the adoption of HOLTEC 72.48 No. 812 for bolt torque requirements.

			• Appendix D was revised to reflect new hazards evaluation per EC 24349 • Editorial changes made throughout to recognize that first campaign is complete and to add discussion on the second campaign
5/9/12	4	Various 7.5 7.11 8.0 Appendix B Appendix C	• LBDCR LIC 12-02 • Revised various sections due to changes in the formatting of 10CFR72.212 (Revision bars not included). This also resulted in the renumbering of sections. • Updated Section 7.5 for 2011 annual dose values • Updated Table in Section 7.11 showing the engineering report numbers for the additional HI-STORMS and MPCs. • Added Section 8 Reference to EC 34606 for 2012 Loading Campaign • Updated Appendix B list of procedures due to procedure revisions. • Updated Appendix B to incorporate ECOs 5014-192 and 5014-199 • Updated Appendix C to note that visual inspection and NDE is an acceptable alternative to load testing of the trunnions per ANSI N 14.6
10/27/14	5	All	• Reformatted into Stand-Alone Licensing Basis Document. Moved information in Appendices A, B and C into the body of the document and re-lettered Appendix D to Appendix A. Revision bars are not used because the document was completely re-formatted. • Clarified the compliance statements for 10 CFR 72.106(a) and (c) in Sections 3.27 and 3.28 and deleted the deviation from HI-STORM FSAR Section 5.1 • Clarified that the fleet QA program governed part 72 activities until October 2014 when the VY-specific program began governing. • Clarified that the safety classification of the ISFSI pad was ITS-B for design and construction and re-classified as ITS-C after construction. • Documented VY review of ECOs, SMDRs, and HIBs.

VERMONT YANKEE NUCLEAR POWER STATION

HI-STORM 100 SYSTEM
10 CFR 72.212 EVALUATION REPORT

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1 - INTRODUCTION

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 (DFS) system at the Vermont Yankee Nuclear Power Station (VY) site Independent Spent Fuel Storage Installation (ISFSI). Before dry fuel storage project implementation, all spent fuel storage at VY was stored under water in a pool in the Reactor Building (RB). 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 spent fuel disposal contract signed by Yankee Atomic Company for VY. Under the existing provisions of the contract, and based upon recent DOE public statements, the start of fuel acceptance by DOE from any U.S. nuclear plant is not expected to begin before 2025.

Currently, the DOE is still not accepting spent fuel. Thus, additional spent fuel storage capacity for VY is necessary. The Holtec HI-STORM 100 System was selected for use at VY and is designed to contain up to 68 BWR fuel assemblies. The HI-STORM 100 System consists of a stainless steel multi-purpose canister (MPC) with a welded baseplate and lid, which is placed inside a coated carbon steel and concrete overpack that is placed on a concrete pad for storage. The VY ISFSI pad is designed to store 2,448 VY spent fuel assemblies in 36 casks, 68 assembles per cask.

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 nine times since that time. VY began moving spent fuel to dry storage in 2008 in accordance with Amendment 2 to this CoC and Revision 4 to the HI-STORM 100 FSAR (the site CFSAR). VY continues to load casks under HI-STORM CoC Amendment 2 and FSAR Revision 4. The VY ISFSI docket number is 72-0059. The VY general license number is SFGL-36.

The VY site implementation review can be broken 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 report to provide a background for the required reviews. Other 10 CFR 72.212 requirements are documented as appropriate in clearly identified report sections.

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).

2) A transfer cask (HI-TRAC) (ITS-A) and HI-TRAC lift yoke (ITS-A) that are used to move the HI-TRAC/MPC assemblage containing fuel from the spent fuel 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 natural ventilation heat removal, radiation shielding, and structural protection for the MPC during storage operations (ITS-B).

4) A mating device (ITS-B) 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 (NITS)) used to transport the loaded HI-STORM overpack from the Containment Access Building (CAB) to the ISFSI pad.

6) Ancillaries consisting of: 1) a fluid pump-down system (NITS) to remove the bulk water from the MPC, 2) a vacuum drying system (VDS) (NITS) to remove the remaining moisture from the fuel cavity, and 3) a helium backfill system (HBS) (NITS) to fill the fuel cavity with an inert gas.

7) The ISFSI concrete storage pad on which the loaded overpacks (HI-STORMs) are placed for long-term storage operations was designed and constructed as ITS-B and was re-classified at ITS-C subsequent to its construction.

Existing major VY plant equipment used for dry fuel storage includes the single failure proof RB crane (Augmented Quality), Refueling Bridge (Safety-Related), and the site heavy-haul transport route between the CAB and the ISFSI pad (NITS).

1.3 VY Cask Loading Sequence

The general sequence of events for loading activities to move VY spent fuel from the Spent Fuel Pool (SFP) to the ISFSI 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. Prior to beginning DFS operations the Low Profile Transporter (LPT) is placed on the rail system in the RB railroad airlock and CAB.

2. DFS operations begin with empty MPCs being brought into the RB at Elevation 252' using the LPT and staged to a designated storage area on the refuel floor of the RB at Elevation 345' using the RB 110-ton single-failure-proof crane. The RB crane with the HI-TRAC lift yoke is then used to stage the HI-TRAC. The empty MPC is then moved from its storage location on the RB refuel floor and placed in the HI-TRAC.

3. Using the RB crane and lift yoke, the HI-TRAC/MPC assemblage is lifted and located to the Cask Washdown Area (CWA) on the refuel floor where they are prepared for submergence in the SFP.

4. The MPC is filled with water to prevent annulus water pressure from lifting the MPC during filling of the annulus and to avoid splash-back of SFP water during placement into the pool. The HI-TRAC/MPC annulus is then filled and the annulus seal and overpressure systems installed.

5. Following verification that the HI-TRAC neutron shielding water jacket is drained of water, the RB crane with Lift yoke is used to move the HI-TRAC/MPC to the cask loading area of the SFP. Due to water volume displacement, Operations and DFS personnel coordinate to maintain SFP water level while the HI-TRAC/MPC is lowered into the pool. The lift yoke is then disengaged from the HI-TRAC and removed from the pool.

6. Sixty-eight (68) pre-selected VY fuel assemblies meeting the limits of the HI-STORM 100 System Certificate of Compliance (CoC) are then loaded into the MPC using the plant's refuel bridge. Verification of the fuel loading is then performed.

7. The RB crane, lift yoke and MPC lid rigging slings are then used to lower the MPC lid with its drain line installed into place in the MPC. The MPC lid provides sufficient shielding to raise the HI-TRAC/MPC out of the spent fuel pool.

8. The lift yoke then engages the HI-TRAC and the RB crane and lift yoke are used to move the HI-TRAC/MPC from the SFP cask loading area to the CWA for MPC closure processing. Due to water volume displacement, Operations and DFS personnel coordinate to maintain SFP water level while the HI-TRAC/MPC is removed from the pool.

9. The MPC lid rigging slings are removed and the lift yoke is disengaged from the HI-TRAC. The MPC lid rigging slings are stored and the lift yoke placed on its storage stand.

10. The HI-TRAC/MPC is decontaminated as necessary and preparations are made to support MPC lid welding, draining, drying, and backfilling.

11. A small amount of water was drained from the MPC prior to its complete removal from the SFP to create a space between the water and the underside of the MPC lid. The welding purge and combustible gas monitoring systems are installed to provide for combustible gas control under the MPC lid during welding. The MPC Lid-to-Shell (LTS) weld is completed during which dye penetrant (PT) nondestructive examinations (NDE) are performed. The purge and combustible gas monitoring systems are removed after the completion of MPC LTS welding.

12. Removable Valve Operating Assemblies (RVOAs) are installed on the MPC vent and drain access port nozzles and MPC fluid processing systems installed including the multi-purpose pumping, hydrostatic test, vacuum drying system (VDS), and blowdown systems. The MPC is then pressure tested with required NDE performed then drained, dried, and backfilled with helium.

13. The vent and drain port caps are closed and the fluid processing systems and RVOAs removed. The vent and drain port covers are then welded in place and PT and helium leak testing performed. Finally, the MPC closure ring segments are welded in place and

PT inspections of these welds performed. At this time the fuel is totally confined within the MPC.

14. The MPC lift cleats, download slings and the HI-TRAC top lid are installed. The RB crane with the lift yoke is used to lift the HI-TRAC/MPC. Before the HI-TRAC/MPC is moved from the CWA, the exterior of the HI-TRAC is surveyed and decontaminated if necessary. The inner shell of the HI-TRAC and the outer shell of the MPC is clean due to the clean water used 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 CWA by taking smears from the MPC lid and the top several inches of the MPC shell exterior.

15. The HI-TRAC/MPC is then moved to the RB 252' elevation to a point above the HI-STORM overpack which was previously moved into the Reactor Building 252' elevation. The HI-STORM overpack will have previously been staged on the LPT with the mating device secured to the top of the overpack body by bolted connections and the seismic restraints installed.

16. The HI-TRAC is placed on top of the mating device. The lift yoke is disconnected from the HI-TRAC lifting trunnions and the download slings used to raise the MPC off the HI-TRAC pool lid. Air bags on the mating device are inflated to allow the HI-TRAC pool lid bolting to be removed.

17. Following pool lid bolt removal the pool lid is lowered into the mating device drawer when the air bags are deflated. 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 to be lowered from the HI-TRAC and into the HI-STORM.

18. After the MPC is lowered into the HI-STORM the download slings are detached from the lift yoke and allowed to rest on top of the MPC. The RB crane and lift yoke are then used to move the empty HI-TRAC off of the mating device once the seismic restraints are removed and place it in a storage location. The mating device remains bolted to the HI-STORM to provide shielding during MPC lift cleat and download sling removal.

19. After the cleats and slings are removed from the MPC lid, the mating device drawer containing the pool lid is closed and the HI-TRAC is removed. The mating device is returned to the RB 345' elevation where the HI-TRAC lid is removed from the mating device and re-attached to the HI-TRAC. The HI-TRAC is returned to its designated location for future use.

20. The HI-STORM overpack lid is installed and the overpack is moved on the LPT from the RB 252' elevation, through the railroad RB airlock, into the CAB.

21. Once the HI-STORM has been moved out of the RB into the CAB, the Vertical Cask Transporter (VCT) is moved into position around the HI-STORM.

22. The VCT HI-STORM lift brackets are then connected to the overpack and the VCT lifts and moves the HI-STORM off and away from the LPT.

23. 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. The overpack lid is then secured in place for long-term storage operations.

1.4 VY Cask Unloading Sequence

Although unlikely to be necessary, provisions are in place to transport the cask back to the Reactor Building and unload fuel from the MPC due to some unforeseen event. 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 in the SFP 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 SFP 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 (e.g., MPC pressure monitoring during re-flooding). Steps that are the reverse of loading operations are not repeated here. The following steps would be implemented to remove the fuel and return it to the spent fuel pool storage racks:

1. Once it is decided that an MPC needs to be unloaded, the overpack is moved back to the location in the CAB just outside the RB using the VCT. The LPT is in place on the rails. The overpack is placed on the LPT and the railroad airlock door is opened.

2. The LPT and overpack are moved through the railroad airlock and to the RB 252' elevation.

3. The HI-STORM lid is removed, the mating device and HI-TRAC are installed, and the MPC is transferred into the HI-TRAC using the RB crane. The HI-TRAC pool lid is re-installed while on the mating device.

4. The HI-TRAC now containing the MPC to be unloaded is moved back to the CWA.

5. After setting the HI-TRAC in the CWA, additional cooling of the MPC fuel cavity may be performed in any manner appropriate for the hardware and for the rate of heat removal deemed necessary. For example, additional heat removal may be accomplished by use of the Annulus Circulation System for circulating water through the HI-TRAC/MPC annulus to increase heat rejection.

6. The vent and drain ports are accessed with the use of a drilling operation that also drills through the closure ring segments so that the RVOAs can be installed.

7. A gas sample is taken from the MPC to determine whether there is any failed fuel in the MPC and establish appropriate radiological controls.

8. 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 may be vented using an appropriate radioactive waste vent path.

9. Cooldown using a Helium cooling system, such as the FHD system, may commence if necessary.

10. Water is introduced into the MPC up to a level that is appropriate to allow lid weld cutting to proceed, while MPC cavity pressure is monitored. Following removal of the RVOAs and any attached fluid processing systems, purging or exhausting of the space under the MPC lid must be commenced before the MPC lid-to-shell weld is removed via cutting.

11. Upon completion of LTS weld cutting, the annulus is then filled and the annulus seal and overpressure systems installed. Using the RB crane and lift yoke, with the MPC lid lifting slings in place, the HI-TRAC/MPC is the moved to the SFP cask loading area.

12. Once the HI-TRAC/MPC is in place in the SFP, lift yoke is detached from the HI-TRAC and the MPC lid is removed. The contained fuel assemblies are returned to the SFP storage racks.

1.5 VY 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

The VY RB cask crane lifting system is a single-failure-proof lifting system. This is based on a comparison of the design features of the crane, lifting devices, and interfacing lift points against the guidance of NUREG-0612, Section 5.1.6. Heavy loads are handled over the cask loading area of the spent fuel pool, including over irradiated fuel in the cask, during MPC lid installation. Based on the single-failure-proof design of the cask lifting system, drops of the transfer cask and the MPC lid are of sufficiently low probability to be considered not credible. The sole drop event of concern is the drop of a fuel assembly during movement from the storage racks to the MPC during cask loading operations.

Drop of a Fuel Assembly into an MPC

This accident is similar to the accident evaluated in Section 14.6.4 of the VY Part 50 UFSAR in which a fuel assembly is dropped on top of the core from a height of approximately 30 feet during refueling operations in the Reactor Building. The primary concern is configurations for criticality of the fuel or gross damage to the fuel such that a radiological contamination spread becomes possible. The conditions and configurations assumed in the VY USAR Chapter 14 safety analyses envelope loading of fuel in the MPC because the fuel assemblies will only be suspended a few feet above the MPC during loading (just enough to clear the top of the transfer cask). The HI-TRAC/MPC can be viewed as a spent fuel storage rack with approximately the same fuel interface conditions.

Sealing and Cleanup

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 CWA (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 DFS procedures. The MPC is pressure tested, the remaining water is removed from the MPC fuel cavity and the VDS is connected to the MPC. The MPC fuel cavity is vacuum 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., deconners, riggers, welders, etc.), discharge of water from the cask, and releases of gases from the cask. MPC preparation activities will be performed under the same administrative control procedures used at VY 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 appropriate dose and dose rate limits, dosimetry and protective clothing for the activities.

Hydrogen Gas Ignition during MPC Lid Welding and Cutting

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.

To support the welding activities, welding procedure specifications (WPSs) are qualified in accordance with VY requirements. The qualified WPSs are for manual and automatic welding. The welding contractor's WPSs fully meet Holtec International's welding specifications and Entergy's welding program requirements.

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 pool 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 explosive 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 by pumping argon into the space and directing the gas from the lid vent through a hydrogen monitor using an explosion proof vacuum.

Argon purging and combustible gas monitoring continues until all lid-to-shell welding operations are complete. Continuous sampling also includes all repairs to the lid weld, if required. If welding of the lid to the MPC shell is interrupted for any reason, combustible gas concentrations are verified to be less than the lower explosive limit prior to continuing welding.

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 and monitored for combustible gases 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.

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 Vertical Cask Transporter (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 VCT is designed in accordance with ANSI N14.6 and employs redundant drop protection features. Thus, no lift height limit needs to be established when moving the loaded HI-STORM overpack. However, for defense in depth, 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.

HI-STORM in Storage

Elsewhere in this report, the effects of natural phenomena (e.g., tornadoes, earthquakes, and floods) and man-made events (e.g., cask drops, fire and explosion) on the loaded storage cask are discussed in the context of CoC requirements and compliance with the cask FSAR. See Sections 3.40, 3.42, and 4.3.6.

2.A - LIST OF SITE-SPECIFIC PROCEDURES

VY uses the following site-specific procedures in order to implement specific fuel transport, processing, and storage activities at the facility. These procedures (latest revision unless otherwise noted) for VY are listed below.

DFS PROCEDURES

1. DP 2220, MPC Off-loading, Storage, and Handling
2. DP 2221, MPC Prior to Use Inspections and Handling
3. DP 2222, HI-STORM 100S Overpack Storage, Prior to Use Inspections, and Handling
4. OP 2223, MPC Fuel Loading Preparations and Sealing Operations
5. OP 2224, MPC Transfer Operations and HI-STORM Transport
6. OP 2225, MPC Unloading Operations
7. DP 2226, MPC Alternate Cooling
8. DP 2227, Vacuum Drying System (VDS) Operation
9. DP 2228, Low Profile Transporter (LPT) Operation
10. DP 2229, Vertical Cask Transporter (VCT) Operation
11. DP 2230, HI-TRAC Offloading, Storage, Prior to use Inspections and Handling
12. EN-DC-215, Fuel Selection for Holtec Dry Cask Storage
13. EN-RE-210, Reactor Core and MPC Cask Fuel Verification
14. OP 2505, Handling Transfer and Storage of Site Radioactive Material
15. OP 2530, Radiological Monitoring Requirements for the HI-STORM 100 Dry Fuel Storage System
16. OP 3127, Natural Phenomenon
17. DP 3201, DFS Equipment Handling and Storage Abnormal Conditions
18. AP 3125, Emergency Plan Classification and Action Level Scheme
19. AP 0010, Situational Reporting Requirements
20. AP 0042, Plant Fire Protection and Fire Prevention

21. AP 0069, Routine Report due to state and Federal Agencies
22. AP 0156, Notification of Significant Events
23. AP 0172, Work Schedule Risk Management – On Line
24. OP 1100, Refuel Platform Operation
25. OP 1101, Management of Refueling Activities and Fuel Assemble Movement
26. OP 4530, Dose Rate Radiation Surveys
27. DP 4531, Radioactive Contamination Surveys
28. AP 4601, Environmental Radiation Surveillance Program
29. PP 7023, Control of Heavy Loads Program Document
30. EN-LI-108, Event Notification and Reporting
31. EN-AD-103, Document Control and Records Management Program
32. OP 5241, Lifting Fixtures and Equipment
33. EN-DC-212, Caskloader Computer Code – Model Development and Updating
34. Deleted
35. EN-MA-100, Fundamentals of Maintenance
36. EN-MA-118, Foreign Material Exclusion
37. EN-MA-119, Material Handling Program
38. EN-DC-211, Dry Fuel Storage Management
39. EN-LI-115, HI-STORM 100 Independent Spent Fuel Storage Installation Licensing Document Preparation and Control
40. EN-LI-112, 10CFR72.48 Evaluations
41. EN-NF-104, Special Nuclear Material Program
42. EN-NF-200, Special Nuclear Material Control
43. EN-NF-201, Special Nuclear Material Reporting

44. EN-IS-109, Compressed Gas Cylinder Handling and Storage

45. PP 7206, Use of Lifting Systems

46. EMMP-INSP-5240-11 Reactor Building Crane Electrical Periodic Inspections

47. EMMP-INSP-5240-13 Reactor Building Crane Hoist Limit Switch Adjustments for Refueling Outage Settings

48. EMMP-INSP-5240-14 Reactor Building Crane Hoist Limit Switch Adjustments for DFS Cask Handling Settings

49. MMMP-INSP-5240-11 Reactor Building Crane Periodic And Wire Rope Inspections

50. MMMP-INSP-5240-13 Reactor Building and Turbine Building Crane Frequent Inspection Checklist

51. EN-FAP-OU-108, Fuel Handling Process

52. EN-DC-160, Dry Fuel Storage Document Control

53. EN-OP-116, Infrequently Performed Tests or Evolutions

54. OP 2200, Operation of Reactor and Turbine Bridge Cranes

55. OPOP-NFPC-2184, Normal Fuel Pool Cooling System

2.B - VY KEY HOLTEC CASK SYSTEM DRAWINGS

Drawing Series	Component	Applicable Revisions
3996	HI-STORM 100S Version B Assembly	See Documentation Packages for Applicable HI-STORM Serial Number and Drawing Revision Number
1402	MPC-68/68F Enclosure Vessel Construction	See Documentation Packages for Applicable MPC Serial Number and Drawing Revision Number
1401	MPC-68/68F Basket Construction	See Documentation Packages for Applicable MPC Serial Number and Drawing Revision Number
4130	100 Ton HI-TRAC 100D Assembly	Revision S/N 1026-6
5079	100 Ton HI-TRAC Lifting Yoke	Revision 2
2602	HI-STORM Lift Bracket	Revision 13
2511	MPC Lift Cleat	Revision 13
4131 4151	Mating Device – 100D for HI-STORM 100S Ver. B and HI-TRAC 100D	Revision 4 Revision 3

2.C - CASK SYSTEMS APPROVED FOR VY ISFSI

The tables below identify those dry fuel storage systems licensed by the NRC which are approved to be used at the VY Independent Spent Fuel Storage Installation, and provide the applicable CoC amendment, and FSAR revision for each MPC, overpack and the transfer cask, by serial number.

This information, in conjunction with changes to the MPC, overpack, and transfer cask initiated by: (1) the CoC holder as listed on the corresponding Component Completion Record (CCR); and (2) VY as described in Section 6 of this report, compose the licensing bases for each MPC, overpack, and transfer cask.

Information regarding casks to be loaded after the date of this report is subject to change, and will be reflected in subsequent revisions to this report.

HI-STORM 100 Cask Multi-Purpose Canister (MPC)-68– CoC 1014									
MPC SN 1021-	**ISFSI Pad Location**	**VY Asset ID Number**	**Contents**	**Date Placed in STORAGE OPERATIONS**	**CoC Amend**	**FSAR HI-2002444 Revision**	**ECO Number 1021-**	**SMDR Number**	**Engineering Report**
58	H-5	OP-DFS-01	68 BWR intact fuel assemblies	2008	2	4	76R1, 77R1, 78, 80, 83R1, 89	1602R1	VY-RPT-08-0003
59	G-5	OP-DFS-02	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 58	N/A	VY-RPT-08-0009
60	F-5	OP-DFS-03	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 58	1548	VY-RPT-08-0010
61	E-5	OP-DFS-04	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 58	N/A	VY-RPT-08-0011
62	D-5	OP-DFS-05	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 58	N/A	VY-RPT-08-0012

HI-STORM 100 Cask Multi-Purpose Canister (MPC)-68– CoC 1014									
MPC SN 1021-	**ISFSI Pad Location**	**VY Asset ID Number**	**Contents**	**Date Placed in STORAGE OPERATIONS**	**CoC Amend**	**FSAR HI-2002444 Revision**	**ECO Number 1021-**	**SMDR Number**	**Engineering Report**
305	C-5	OP-DFS-06	68 BWR intact fuel assemblies	2011	2	4	76R1, 77R1, 78, 80, 83R1, 89, 95, 96R1, 97, 99, 100	N/A	VY-RPT-11-0009
306	B-5	OP-DFS-07	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 305	N/A	VY-RPT-11-00010
307	A-5	OP-DFS-08	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 305	1705R1	VY-RPT-11-00011
308	H-4	OP-DFS-09	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 305	1705R1	VY-RPT-11-00012
362	G-4	OP-DFS-10	68 BWR intact fuel assemblies	2012	2	4	76R1, 77R1, 78, 80, 83R1, 89, 95, 96R1, 97, 99, 100, 102, 103, 104, 106, 107, 109	1705R2	VY-RPT-12-0009
363	F-4	OP-DFS-11	68 BWR intact fuel assemblies	2012	2	4	Same as S/N 362	1705R2	VY-RPT-12-00010

HI-STORM 100 Cask Multi-Purpose Canister (MPC)-68– CoC 1014									
MPC SN 1021-	**ISFSI Pad Location**	**VY Asset ID Number**	**Contents**	**Date Placed in STORAGE OPERATIONS**	**CoC Amend**	**FSAR HI-2002444 Revision**	**ECO Number 1021-**	**SMDR Number**	**Engineering Report**
364	E-4	OP-DFS-12	68 BWR intact fuel assemblies	2012	2	4	Same as S/N 362	1705R2	VY-RPT-12-00011
365	D-4	OP-DFS-13	68 BWR intact fuel assemblies	2012	2	4	Same as S/N 362	1705R2	VY-RPT-12-00012

HI-STORM 100S Version B HI-STORM Overpack – CoC 1014									
HI-STORM SN 1024-	**ISFSI Pad Location**	**VY Asset ID Number**	**Contents**	**Date Placed in Storage**	**CoC Amendment**	**FSAR HI-2002444 Revision**	**ECO Number 1024-**	**SMDR Number**	**Engineering Report**
65	H-5	OP-DFS-01	68 BWR intact fuel assemblies	2008	2	4	119, 121, 123R1, 126, 131R1	N/A	VY-RPT-08-0003
66	G-5	OP-DFS-02	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 65	N/A	VY-RPT-08-0004
67	F-5	OP-DFS-03	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 65	N/A	VY-RPT-08-0005
68	E-5	OP-DFS-04	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 65	1574	VY-RPT-08-0006
69	D-5	OP-DFS-05	68 BWR intact fuel assemblies	2008	2	4	Same as S/N 65	1573, 1577, 1595	VY-RPT-08-0007
240	C-5	OP-DFS-06	68 BWR intact fuel assemblies	2011	2	4	119, 121, 123R1, 126, 131R1, 134, 135R1, 139, 141	1835	VY-RPT-11-0005

HI-STORM 100S Version B HI-STORM Overpack – CoC 1014									
HI-STORM SN 1024-	**ISFSI Pad Location**	**VY Asset ID Number**	**Contents**	**Date Placed in Storage**	**CoC Amendment**	**FSAR HI-2002444 Revision**	**ECO Number 1024-**	**SMDR Number**	**Engineering Report**
241	B-5	OP-DFS-07	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 240	N/A	VY-RPT-11-0006
242	A-5	OP-DFS-08	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 240	N/A	VY-RPT-11-0007
461	H-4	OP-DFS-09	68 BWR intact fuel assemblies	2011	2	4	Same as S/N 240	N/A	VY-RPT-11-0008
569	G-4	OP-DFS-10	68 BWR intact fuel assemblies	2012	2	4	119, 121, 123R1, 126, 131R1, 134, 135R1, 139, 141, 142, 151	2179	VY-RPT-12-00005
570	F-4	OP-DFS-11	68 BWR intact fuel assemblies	2012	2	4	Same as 569	2179	VY-RPT-12-00006
571	E-4	OP-DFS-12	68 BWR intact fuel assemblies	2012	2	4	Same as 569	2179	VY-RPT-12-00007
572	D-4	OP-DFS-13	68 BWR intact fuel assemblies	2012	2	4	Same as 569	2179	VY-RPT-12-00008

HI-STORM 100 Cask HI-TRAC 100D Transfer Cask – CoC 1014								
HI-TRAC Serial Number	**CoC Amendment**	**VY Asset ID Number**	**FSAR HI-2002444 Revision**	**Effective Date**	**Expiration Date**	**ECO Number 1026-**	**SMDR Number**	**Engineering Report**
1026-6	2	FTC-DFS-01	4	6/7/05	5/31/20	41	1564R1	VY-RPT-08-0013

3 - COMPLIANCE WITH 10 CFR PART 72

VY compliance with the requirements specified in 10 CFR 72 is provided below.

3.1 §72.1 – Purpose
§72.2(a)(1), (b), (c), and (e) – Scope
§72.3 – Definitions

These regulations are not repeated in their entirety here due to their length. These regulations set forth the purpose and scope of 10 CFR 72 and define key terms used in this part of the regulations.

These regulations are general requirements and require no specific action by Entergy-VY to demonstrate compliance.

3.2 §72.4 – Communications

Except where otherwise specified, all communications and reports concerning the regulations in this part and applications filed under them should be sent by mail addressed: ATTN: Document Control Desk, Director, Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; by hand delivery to the NRC's offices at One White Flint North, 11555 Rockville Pike, Rockville, Maryland between 7:30 a.m. and 4:15 p.m. eastern time; or, where practicable, by electronic submission, for example, via Electronic Information Exchange, or CD-ROM. Electronic submissions must be made in a manner that enables the NRC to receive, read, authenticate, distribute, and archive the submission, and process and retrieve it a single page at a time. Detailed guidance on making electronic submissions can be obtained by visiting the NRC's Web site at http://www.nrc.gov/site-help/e-submittals.html; by e-mail to MSHD.Resource@nrc.gov; or by writing the Office of Information Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The guidance discusses, among other topics, the formats the NRC can accept, the use of electronic signatures, and the treatment of nonpublic information. If the submission deadline date falls on a Saturday, or Sunday, or a Federal holiday, the next Federal working day becomes the official due date.

All communications and reports are sent ATTN: Document Control Desk, Director, Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and as specified by 10 CFR 72 requirements.

Entergy-VY has been assigned docket number 72-0059 for the ISFSI, general license number SFGL-36. It is also noted that the Division of Spent Fuel Storage and Transportation is now known as the Division of Spent Fuel Management.

3.3 §72.5 – Interpretations

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of the Commission, other than a written interpretation by the General Counsel, will be recognized to be binding upon the Commission.

These regulations are general requirements and require no specific action by Entergy-VY to demonstrate compliance.

3.4 §72.6(a), (b), and (c)(1) – License Required; Types of Licenses

These regulations are not repeated in their entirety here due to their length. These regulations establish the types of licenses required for ISFSIs and Monitored Retrievable Storage (MRS) facilities.

These regulations are general requirements and require no specific action by Entergy-VY to demonstrate compliance. In §72.6(b), "specific license" refers to the 10 CFR 50 license held by the facility.

3.5 §72.7 - Exemptions

The Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

General licensees such as Entergy-VY are permitted to request exemptions from the Part 72 regulations provided the need is demonstrated and the criteria in §72.7 are met. Exemption requests are prepared and submitted by Entergy-VY Licensing using the same process as Part 50 exemption requests.

3.6 §72.8 – Denial of Licensing by Agreement States
§72.9 – Information Collection Requirements: OMB Approval

§72.8: *Agreement States may not issue licenses covering the storage of spent fuel and reactor-related GTCC waste in an ISFSI or the storage of spent fuel, high-level radioactive waste, and reactor-related GTCC waste in an MRS.*

§72.9: *(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). OMB has approved the information collection requirements contained in this part under control number 3150-0132.*

(b) The approved information collection requirements contained in this part appear in §§ 72.7, 72.11, 72.16, 72.22 through 72.34, 72.42, 72.44, 72.48 through 72.56, 72.62, 72.70 through 72.80, 72.90, 72.92, 72.94, 72.98, 72.100, 72.102, 72.103, 72.104, 72.108, 72.120,

72.126, 72.140 through 72.176, 72.180 through 72.186, 72.192, 72.206, 72.212, 72.218, 72.230, 72.232, 72.234, 72.236, 72.240, 72.242, 72.244, 72.248.

(c) In §72.79, Form N-71 and associated forms are approved under control number 3150-0056, and DOC/NRC Forms AP-1, AP-A, and associated for are approved under control number 0694-0135.

These regulations are general requirements and require no specific action by Entergy-VY to demonstrate compliance.

3.7 §72.10 – Employee Protection

This regulation is not repeated in its entirety here due to its length. In summary, this regulation states that employees engaging in protected activities are prohibited from retaliatory or discriminatory actions being taken against them for engaging in such activity.

Entergy Nuclear Management Manual (NMM) Procedure EN-EC-100, *Guidelines for Implementation of the Employee Concerns Program*, encompasses the requirements of this regulation.

3.8 §72.11 – Completeness and Accuracy of Information

This regulation is not repeated in its entirety here due to its length. This rule requires all information submitted to the NRC by Entergy-VY *to be complete and accurate in all material respects. It also requires* Entergy-VY *to notify the NRC of any information having a significant implication for public health and safety or common defense and security.*

NMM Procedure EN-LI-106, *NRC Correspondence*, specifies the requirements of this article. Also, NRC Form 3, *Notice to Employees* contains relevant information.

3.9 §72.12 – Deliberate Misconduct

This regulation is not repeated in its entirety here due to its length. This rule states that employees, contractors, or subcontractors of Entergy-VY may not engage in deliberate misconduct that would cause a violation of the NRC's regulations.

NMM Procedure ENS-HR-135, *Disciplinary Action*, encompasses the requirements of this article.

3.10 §72.13(a) and (c) – Applicability

These regulations identify those sections under Part 72 that apply to the activities associated with a general license.

These regulations are general requirements and require no specific action by Entergy-VY to demonstrate compliance.

3.11 §72.30(c) and (d) – Financial Assurance and Recordkeeping for Decommissioning

These regulations are not repeated in their entirety here due to their length. These rules establish decommissioning financial requirements for general licensees, such as Entergy-VY.

Because Entergy-VY is a Part 50 licensee and must meet the applicable provisions of 10 CFR 50.75, Entergy-VY complies with this regulation per 10 CFR 72.30(c)(5). No unique Part 72 decommissioning financial requirements are necessary to comply with this rule. This regulation is complied with under the Entergy-VY Part 50 decommissioning program. The information is to be submitted with Entergy-VY's decommissioning plan five or more years prior to decommissioning in accordance with §§50.54(bb), 50.82, and 72.218.

3.12 §72.32(c) and (d) – Emergency Plan

(c) *For an ISFSI that is: (1) located on the site, or (2) located within the exclusion area as defined in 10 CFR part 100, of a nuclear power reactor licensed for operation by the Commission, the emergency plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.*

(d) A licensee with a license issued under this part may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.

Entergy-VY's emergency plan meets the requirements of §50.47. See the evaluation under 10 CFR 72.212(b)(10) for additional information. The requirements of 10 CFR 72.32(d) are similar to those in 10 CFR 50.54(x) for which training is provided to personnel on when a departure from a license condition or technical specification is appropriate to take.

3.13 §72.44(b) (d) and (f) – License Conditions

These regulations are not repeated in their entirety here due to their length. 10 CFR 10 CFR 72.44(b) establishes the over-arching license conditions for Part 72 licensees. These requirements address requirements such as the NRC's authority to revoke or suspend the license and to request information from licensees to be submitted under oath. 10 CFR 72.44(d), although not applicable to general licensees per 10 CFR 72.13, is applicable to Entergy-VY through the HI-STORM 100 CoC (Appendix A, Section 5.4). 10 CFR 72.44(d) establishes requirements for radioactive effluent control and reporting. 10 CFR 72.44(f) requires the ISFSI to be operated under an NRC-approved emergency plan and changes to that plan may be made by the Energy-VY without prior NRC approval provided the changes do not decrease the effectiveness of the plan.

Compliance with the general requirements in 10 CFR 72.44(b)(1) through (b)(3) require no specific action by Entergy-VY to demonstrate compliance. Compliance with the training requirements in 10 CFR 72.44(b)(4) and (b)(5) are discussed in Sections 3.32 and 4.1.10 of this report. The bankruptcy reporting requirements in 10 CFR 72.44(b)(6) are covered by Entergy-VY's compliance with 10 CFR 50.54(cc). Compliance with 10 CFR 72.44(d) is discussed in Section 4.2.2 of this report. Compliance with 10 CFR 72.44(f) is discussed in Section 3.44 of this report.

3.14 §72.48 – Changes, Tests, and Experiments

This regulation is not repeated in its entirety here due to its length. This regulation establishes the requirements for when an activity requires NRC review and approval prior to implementation and reporting requirements for §72.48 evaluations.

Entergy-VY will evaluate any proposed cask design changes and ensure the HI-STORM CoC envelopes them. Entergy-VY will report any changes evaluated under this section in periodic submittals in compliance with NMM Procedure EN-LI-112, *10 CFR 72.48 Evaluation Program*, and NMM Procedure EN-EV-115, *Environmental Reviews and Evaluations*.

3.15 §72.50(a) – Transfer of License

No license or any part included in a license issued under this part for an ISFSI or MRS shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission gives its consent in writing.

This regulation states that the VY ISFSI license cannot be transferred to another entity without prior NRC approval. No specific action by Entergy-VY to demonstrate compliance is required.

3.16 §72.52(a), (b), (d), and (e) – Creditor Regulations

These regulations are not repeated in their entirety here due to their length. These regulations establish requirements for creditors of Part 72 licensees, with particular emphasis on the restrictions on transfer of special nuclear material contained in spent fuel.

For Entergy-VY, compliance with these regulations is demonstrated through compliance with 10 CFR 50.81.

3.17 §72.60 – Modification, Revocation, and Suspension of License

This regulation is not repeated in its entirety here due to its length. This regulation establishes NRC authority to modify, revoke, or suspend a Part 72 license and the reasons permitting such action.

No specific action by Entergy-VY to demonstrate compliance.

3.18 §72.62 – Backfitting

This regulation is not repeated in its entirety here due to its length. This regulation establishes the circumstances that must exist for the NRC to require the addition, elimination, or modification of SSCs associated with the ISFSI after the license has been issued. For a general licensee, the license is considered "issued" when the first cask loaded with spent fuel is placed at the ISFSI pad. Energy-VY may choose to contest an NRC request or demand to add, eliminate, or modify the SSCs of an ISFSI if they believe the request or demand does not meet the criteria in this regulation.

No statement of compliance is required. If Entergy-VY wishes to seek a backfit evaluation from the NRC, they would follow a process similar to the 10 CFR 50.109 process for the Part 50 facility.

3.19 §72.72 – Material Balance, Inventory, and Records Requirements for Stored Material

This regulation is not repeated in its entirety here due to its length. This regulation requires licensees to maintain an accounting system for all special nuclear material (SNM) at the ISFSI. The SNM primarily resides in the spent fuel, but can be found in greater-than-class-C (GTCC) waste. This regulation requires records to be kept and a physical inventory to be performed.

NMM Procedure EN-NF-104, *Special Nuclear Materials Program*, contains inventory, transfer, and storage requirements for dry fuel storage. A documentation review of VY fuel in dry storage is conducted yearly in accordance with this procedure. The dual storage requirement in 10 CFR 72.72(d) is satisfied by procedure EN-AD-103, "Document Control and Records Management Program."

3.20 §72.74 – Reports of Accidental Criticality of Loss of Special Nuclear Material

(a) Each licensee shall notify the NRC Operations Center within one hour of discovery of accidental criticality or any loss of special nuclear material.

(b) This notification must be made to the NRC Operations Center via the Emergency Notification System if the licensee is party to that system. If the Emergency Notification System is inoperative or unavailable, the licensee shall make the required notification via commercial telephonic service or any other dedicated telephonic system or any other method that will ensure that a report is received by the NRC Operations Center within one hour. The exemption of § 73.21(g)(3) of this chapter applies to all telephonic reports required by this section.

(c) Reports required under § 73.71 of this chapter need not be duplicated under the requirements of this section.

NMM Procedure EN-LI-108, *Event Notification and Reporting*, requires compliance with 10 CFR 72.74.

3.21 §72.75 – Reporting Requirements for Specific Events and Conditions

These regulations are not repeated in their entirety here due to its length. These regulations establish the requirements for reporting events associated with the ISFSI and/or the dry storage cask design and operation.

NMM Procedure EN-LI-108, *Event Notification and Reporting*, includes the reports required by §72.75.

3.22 §72.76 – Material Status Reports

(a) Except as provided in paragraph (b) of this section, each licensee shall complete in computer-readable format and submit to the Commission a Material Balance Report and a Physical Inventory Listing Report as specified in the instructions in NUREG/BR-0007 and NMMSS Report D--24 "Personal Computer Data Input for NRC Licensees.". Copies of these instructions may be obtained either by writing the U.S. Nuclear Regulatory Commission, Division of Fuel Cycle Safety and Safeguards, Washington, DC 20555-0001, or by e-mail to RidsNmssFcss@nrc.gov. These reports, as specified by §74.13 or 40.64 of this chapter, provide information concerning the special nuclear material and/or source material possessed, received, transferred, disposed of, or lost by the licensee. Each report must be submitted within 60 days of the beginning of the physical inventory required by §72.72(b). The Commission may, when good cause is shown, permit a licensee to submit Material Balance Reports and Physical Inventory Listing Reports at other times. Each licensee required to report material balance and inventory information as described in this part, shall resolve any discrepancies identified during the report review and reconciliation process within 30 calendar days of notification of a discrepancy identified by NRC. The Commission's copy of this report must be submitted to the address specified in the instructions. These prescribed computer-readable forms replace the DOE/NRC forms 742 and 742C which have been previously submitted in paper form.

(b) Any licensee who is required to submit routine material status reports pursuant to § 75.35 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit such reports only as provided in that section instead of as provided in paragraph (a) of this section.

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 10 CFR 74.13.

3.23 §72.78 – Nuclear Material Transfer Reports

a) Except as provided in paragraph (b) of this section, whenever the licensee transfers or receives or adjusts the inventory, in any manner, of special nuclear material as specified by §74.15 and/or source material as specified by §40.64 of this chapter, the licensee shall complete in computer-readable format a Nuclear Material Transaction Report as specified in the instructions NUREG/BR-0006 and NMMSS Report D-24, "Personal Computer Data Input for NRC Licensees." Copies of these instructions may be obtained either by writing the U.S. Nuclear Regulatory Commission, Division of Fuel Cycle Safety and Safeguards, Washington, DC 20555-0001, or by e-mail to RidsNmssFcss@nrc.gov. Each licensee who transfers the material shall submit a Nuclear Material Transaction Report in computer-readable format as specified in the instructions no later than the close of business the next working day. Each licensee who receives the material shall submit a Nuclear Material Transaction Report in computer-readable format in accordance with instructions within ten (10) days after the material is received. Each ISFSI licensee who receives spent fuel from a foreign source shall complete both the supplier's and the receiver's portion of the Nuclear Material Transaction Report, verify the identity of the spent fuel, and indicate the results on the receiver's portion of the form. These prescribed computer-readable forms replace the DOE/NRC Form 741 which have been previously submitted in paper form.

(b) Any licensee who is required to submit Nuclear Material Transactions Reports pursuant to § 75.34 of this chapter (pertaining to implementation of the US/IAEA Safeguards Agreement) shall prepare and submit the reports only as provided in that section instead of as provided in paragraph (a) of this section.

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 10 CFR 74.15. 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*). However, movement of fuel from the spent fuel pool to dry storage within the protected area is not considered "transfers" under §74.15 requirements.

3.24 §72.80(a) through (f) – Other Records and Reports

These regulations are not repeated in their entirety here due to their length. These regulations require records to be kept and reports to be made in accordance with the conditions of the license and other requirements, not specifically addressed elsewhere in the Part 72 regulations. It also establishes requirements for the transfer of records to the NRC or the new licensee if the Part 72 license is terminated or transferred, respectively. For a Part 72 general license, this could only take place if the Part 50 license is transferred.

Records are maintained in accordance with procedures that meet the VY Quality Assurance Program document control requirements for Engineering Reports (ERs), department procedures, and licensing documents. Financial report requirements are met by compliance with §50.71(b). An annual report is required to be submitted by all Part 50 licensees per 10 CFR 140.21. Meeting these Part 50 obligations meet the Part 72 obligation.

3.25 §72.82 – Inspections and Tests
§72.84 – Violations
§72.86 – Criminal Penalties

These regulations are not repeated in their entirety here due to their length. These regulations establish that the VY ISFSI and Entergy must facilitate NRC inspections, provide office space, and perform (or allow the NRC to perform) tests upon request. They also establish that Entergy is subject to violations and civil and criminal penalties for violations of the regulations.

Entergy-VY complies with this requirement by complying with similar requirements under 10 CFR 50.70.

3.26 §72.104 – Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS

This regulation is not repeated in its entirety here due to its length. This regulation establishes the requirements for control of radiation from the ISFSI and dose limits at or beyond the ISFSI controlled area due to normal and off-normal ISFSI operations and other nearby fuel facility operations.

Compliance with this requirement is addressed by §72.212(b)(5)(iii) with the required evaluation described in Section 3.41 of this report and Entergy-VY's compliance with 10 CFR 50 requirements for control of direct radiation.

3.27 §72.106 - Controlled Area of the ISFSI

(a) For each ISFSI or MRS site, a controlled area must be established.

As defined by 10CFR72, 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 VY ISFSI is located within the plant protected area. The ISFSI controlled area boundary is the same as the site owner-controlled area (OCA), which is described in VY UFSAR Section 2.2.4. The minimum distance from the ISFSI to the OCA boundary is approximately 873 feet, or 266 meters and is shown in Appendix C to Holtec Report HI-2073701, "Dose Versus Distance at the Vermont Yankee ISFSI."

(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. 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 ANSI N14.5. Therefore, there is no gaseous effluent dose contribution to the calculated normal, off-normal, or accident offsite accident dose.

The minimum distance of 266 meters from the VY ISFSI to the controlled area boundary meets the minimum requirement of 100 meters, as discussed in the compliance statement under 10 CFR 72.106(a) above.

(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 VY ISFSI is completely contained within the plant protected area. There are no public highways or railroads traversing the ISFSI controlled area. The Connecticut River runs through the plant owner controlled area and access is controlled by plant security in accordance with the site security plan.

3.28 §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. The Reactor Building crane is a Part 50 component designated as "Augmented Quality Program" (QP). Activities involving ITS or QP components or subcomponents are conducted in accordance with Holtec International's 10 CFR 72, Subpart G quality assurance program, or the VY 10 CFR 50 Appendix B quality assurance program.

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, 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:

(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 VY ISFSI 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 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 VY 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.

Consideration of the most severe of the natural phenomena reported for the site is as follows:

<u>Tornado</u>

See Section 3.42 of this report.

<u>Flooding</u>

See Section 4.3.6.4 of this report.

<u>Earthquake</u>

See Sections 3.40 and 4.3.6.3 of this report.

<u>72.122(c): 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 VY fire protection plan and fire hazards analysis have been reviewed and revisions to the program and procedures were made as appropriate to support cask loading, movement, and ISFSI operations. A fire suppression system is not used at the VY ISFSI because there are no combustible materials stored at the ISFSI. Fires due to transient combustibles will be extinguished with portable fire suppression equipment. A site-specific evaluation of fire and explosion hazards for the ISFSI is required by the HI-STORM 100 CoC. That evaluation is discussed in Section 4.3.6.5 of this report.

72.122(d): Sharing of structures, systems, and components. 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. Temperature monitoring of the ISFSI pad area ambient temperature and outlet vent air temperature validates heat removal for the cask in accordance with technical specifications. The ISFSI, including the cask system shares no structures, systems, or components important to safety with any other facility.

72.122(e): Proximity of sites. 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 VY ISFSI is co-located within the protected area of the Part 50 facility. The cumulative effects of combined operations of the plant 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 negligible due to the significant distance between the ISFSI and the site boundary and partial shielding provided by the power plant structures. The HI-STORM 100 System is designed not to release any gaseous effluents under normal, off-normal, or accident conditions, so there is no additional gaseous effluent dose to the public from ISFSI operations.

72.122(f): Testing and maintenance of systems and components. 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. No in-situ periodic inspections or testing of the storage systems during storage operations is required. 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): Emergency capability. 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 VY 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 via gate. 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): 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. Temperature monitoring of the inlet and outlet cooling air assures adequate heat removal for the cask in accordance with technical specifications. 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.

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 or control systems. However, a temperature monitoring system is employed to verify adequate cask cooling.

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 VY 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 room or control area is provided.

72.122(k): Utility or other services. (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. Temperature monitoring of the ISFSI pad area ambient temperature and outlet vent air temperature validates heat removal for the cask in accordance with technical specifications. If the temperature monitoring system loses power, the inlet and outlet vents may be visually inspected to assure adequate cooling is taking place. The ISFSI, including the cask system shares no structures, systems, or components important to safety with any other facility.

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 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 VY site.

3.29 §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 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 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.

72.124(b) Methods of criticality control. 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 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 of the MPC-68 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): Criticality Monitoring. 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 spent fuel pool; moved to the cask decontamination area; and drained, dried, and backfilled with helium), criticality monitoring is provided by dose rate detectors specifically designed to provide alarm functions in the event of a criticality incident. Criticality monitoring is addressed in site procedure OP 2530.

3.30 §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;

(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 cask loading/unloading procedure OP 2530. The VY radiation protection program and application of these procedures also supports VY commitment to compliance with the requirements of 10CFR20 for occupational dose considerations. See Section 2.A of this report for a detailed list of DFS 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 MPC is fully welded and emits no gaseous effluents once the MPC is prepared for storage. Existing alarm systems for radiological monitoring fuel loading operations in the Reactor Building has 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 pool area and area monitors in the Reactor Building, monitoring is performed during onsite transportation as required by the site DFS procedures.

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 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 gaseous effluents from the HI-STORM casks is required.

An area environmental dose monitoring device is installed in the ISFSI area to meet the requirements of procedure AP-4601, "Environmental Radiation Surveillance Program."

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 limit given in §72.106. Systems designed to monitor the release of radioactive materials must have a 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 gaseous 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 "leaktight" 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 gaseous effluent controls or dose analysis is required.

3.31 §72.140 through §72.176 – Quality Assurance

These regulations are not repeated in their entirety here due to their length. These regulations set forth the criteria to be met by the general licensee's Quality Assurance program governing Part 72-related activities.

In anticipation of the permanent shutdown of the VY power plant, Entergy has created a site-specific VY Quality Assurance Program Manual (QAPM). The VY QAPM, Revision 0, duplicates the Entergy fleet QAPM, which satisfies applicable criteria of Appendix B to 10 CFR 50 and was approved by the NRC (Reference CNRI 2001-00007, dated 12/5/01). Entergy notified NRC of its intent to use the Entergy fleet QAPM for dry spent fuel storage activities at its VY ISFSI as permitted in Section 72.140(d) via letter BVY 07-031 "Intent to use 10CFR50 Appendix B Quality Assurance Program for Independent Spent Fuel Storage Installation Notification Pursuant to 10CFR72.140(d)", dated April 18, 2007. Applicable ISFSI-related activities were governed by the Entergy fleet QAPM through about mid-October 2014. Beginning in October 2014 the VY site-specific QAPM will be used to govern ISFSI-related activities. Entergy informed the NRC of this change via letter BVY 14-072, dated October 27, 2014.

The Corrective Action Program is controlled in accordance with NMM Procedure EN-LI-102, *Corrective Action Process.* The Holtec Quality Assurance Program is described in Section 13.3 of the HI-STORM 100 CFSAR, which has been approved by the NRC.

3.32 §72.44, §72.144, §72.190 – 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.

The VY Training Program was reviewed to assure compliance to the requirements of 10CFR72 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 VY has been reviewed and verified to meet these requirements and 10CFR72 requirements. Specifics to this review are listed below.

- HI-STORM FSAR Section 8.0 requires training procedures in place to account for operation of an ISFSI. The VY DFS 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 of the HI-STORM 100 System & ISFSI. The VY DFS 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. VY performed dry run training in accordance with this FSAR section and CoC condition before the first fuel loading campaign in 2008. See Section 4.1.10 of this report for additional discussion.

- 10CFR72.44(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 FSAR & license. The VY DFS Training Program ensures compliance with these requirements.

- 10CFR72.44(b)(5) and 10CFR72.190 require that only trained and certified personnel operate equipment and controls that have been identified as important to safety in the FSAR & license. The VY DFS Training Program ensures compliance with these requirements.

- 10CFR72.144 (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 VY DFS Training Program
    - ASNT-SNT-TC-1A
    - ANSI-45.2.6
    - VY Quality Assurance Program Manual
    - NDE 2.10, Certification of NDE Personnel.
    - NDE 2.12, Certification of Visual Testing (VT) Personnel.

- EN-QV-111, Quality Control Certification.

3.33 §72.194 – Physical Requirements

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 physical condition and health of personnel involved with operation of equipment under this section are subject to requirements of:

- NMM Procedure EN-NS-102, *Fitness For Duty Program*
- NMM Procedure EN-NS-112, *Medical Program*
- NMM Procedure EN-MA-119, *Material Handling Program*

3.34 §72.210 – General License Issued

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 Certificate of Compliance CoC 72-1014 for the HI-STORM 100 Cask System effective June 1, 2000. Entergy is authorized to possess nuclear fuel in accordance with Part 50 license DPR-28 issued by the NRC for the operation of VY. Thus, Entergy-VY is eligible to be a Part 72 general licensee in accordance with this rule.

10 CFR 72.212 – CONDITIONS OF GENERAL LICENSE ISSUED UNDER §72.210

3.35 §72.212(a) – General Requirements

(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-VY is authorized to possess nuclear fuel at VY in accordance with the Part 50 license issued by NRC for 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 a certified dry storage cask system listed in 10 CFR 72.214 (CoC 1014).

(a)(3) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance shall commence upon the date that the particular cask is first used by the general licensee to store spent fuel, shall continue through any renewals of the Certificate of Compliance, unless otherwise specified in the Certificate of Compliance, and shall terminate when the cask's Certificate of Compliance expires. For any cask placed into service during the final renewal term of a Certificate of Compliance, or during the term of a Certificate of Compliance that was not renewed, the general license for that cask shall terminate after a storage period not to exceed the length of the term certified by the cask's Certificate of Compliance. Upon expiration of the general license, all casks subject to that general license must be removed from service.

The beginning of the twenty-year VY 10 CFR 72 general license for each HI-STORM cask is when the cask is placed in storage operation (see Section 2.C of this report). Current options upon approaching the end of licensed life for each cask 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).

VY tracks the in-service life of each cask in this 10 CFR 72.212 Evaluation Report. Through Entergy's participation on the Holtec User Group, Entergy maintains cognizance of Holtec's plan and schedule for renewing the HI-STORM CoC. Should Holtec choose not to renew the CoC, alternate actions will be developed at that time. For example, VY could choose to convert to a Part 72 specific license.

3.36 §72.212(b)(1) and (b)(2) – Notification and Cask Registration

(b) The general licensee must:

(1) 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 notified the NRC of its intent to store spent fuel using the HI-STORM 100 storage cask at VY as documented in Letter BVY 07-013, dated March 13, 2007.

(2) 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 number, the CoC amendment number to which the cask conforms, unless loaded under the initial certificate, cask model number, 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-VY registers each cask within 30 days following loading in accordance with direction in site procedure OP 2224, "MPC Transfer Operations and HI-STORM Transport."

3.37 §72.212(b)(3) – Use of an NRC-Approved Cask

The general licensee must ensure that each cask used by the general licensee conforms to the terms, conditions, and specifications of a CoC or an amended CoC listed in §72.214.

The HI-STORM 100 Cask System is listed in 10 CFR 72.214 under CoC number 1014. See Section 4 for detailed discussion of how VY complies with the terms, conditions, and specification in the HI-STORM CoC, including compliance with CoC Appendices A and B.

3.38 §72.212(b)(4) – Applying Amended CoCs to Previously Loaded Casks

In applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, register each such cask with the Nuclear Regulatory Commission no later than 30 days after applying the changes authorized by the amended CoC. 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 number, the CoC amendment number to which the cask conforms, cask model number, 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-VY has not applied the requirements of an amendment to the HI-STORM 100 CoC to casks loaded under a previous amendment to the CoC.

3.39 72.212(b)(5)(i) – Review of the CoC

The general licensee must perform written evaluations, before use and before applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that the cask, once loaded with spent fuel or once the changes authorized by an amended CoC have been applied, will conform to the terms, conditions, and specifications of a CoC or an amended CoC listed in §72.214.

See Section 4 of this report for detailed discussion of compliance with all CoC Conditions, Appendix A, "Technical Specifications," and Appendix B, "Approved Contents and Design Features" that apply to the VY ISFSI.

3.40 §72.212(b)(5)(ii) – ISFSI Pad Design

The general licensee must perform written evaluations, before use and before applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which 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 or other soil instability due to vibratory ground motion.

The VY ISFSI is located on the north-end of the VY site within the existing protected area. The ISFSI consists of a 76' x 132' x 3' thick storage pad with appropriate access. Access to the storage pad is provided by a 30' x 132' concrete apron located adjacent to the south edge of the storage pad. There is a concrete access ramp on the east side and west side of the concrete apron.

The concrete pad provides storage locations for thirty-six (36) Holtec HI-STORM 100 Overpacks arranged in a 5 x 8 array. This arrangement provides four (4) extra spaces that will allow access to any HI-STORM 100 Overpack.

The ISFSI storage pad is a reinforced concrete slab founded on engineered fill placed on existing soils. Seismic input at VY is at bedrock which lies from 20 to 30 feet below the pad. Strain-compatible soil properties were determined and acceleration time histories were developed at the bedrock elevation. The results of these analyses were used as design input to the soil structural interaction analysis and the assessment of liquefaction potential. The results demonstrated that liquefaction will not occur at VY. The structural interaction analysis output was used as design input to the seismic analysis of the storage pad and the sliding and tipping analysis of the casks on the storage pad.

<u>Design Criteria</u>

The VY ISFSI pad analysis uses the following primary parameters found in the HI-STORM 100 FSAR, Table 2.2.9, Parameter Set "A":

- Maximum compression strength of concrete (f'_c) at 28 days curing = 4.2 ksi
- Minimum compression strength of concrete (f'_c) at 28 days curing = 3.0 ksi.
- Maximum modulus of elasticity (E_{sg}) of subgrade = 28 ksi.
- Maximum slab thickness (t) = 36"
- Pad reinforcement (top and bottom, both directions): 60 ksi yield strength ASTM material

The analysis assessed the pad design for the condition where the overpacks are fully "glued" to the pad surface and therefore delivers all the movement that results from the seismic accelerations. The sliding analysis computes the maximum displacements of the casks for various soil conditions and coefficients of friction (from 0.0 to 0.8) between the cask and storage pad to ensure the casks will not impact each other or slide off the pad.

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 100 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 engineered backfill meets the design basis established in the Holtec FSAR. The design basis for the engineered backfill requires the material stiffness of the supporting backfill material to limit fuel deceleration due to a postulated HI-STORM 100 overpack tip-over event. The source materials, in-place properties, and method of placement of the engineered backfill are identified in the GZA Geotechnical Engineering Report for the ISFSI site dated January 2004 and satisfy the Holtec design basis requirements. The GZA report limits the net allowable bearing pressure under the storage pad to 2 tons/ft^2. In accordance with the GZA report recommendations, a minimum 4'0" thick sub-base course was placed under the storage pad to provide frost line protection. Therefore, the storage pad and engineered backfill are founded upon a non-frost susceptible sub-grade.

Because VY specified and verified the ISFSI pad properties in the HI-STORM 100 FSAR, Table 2.2.9, Parameter Set "A", the cask deceleration limit of less than or equal to 45 g's for a cask tip-over was met.

VY is utilizing a Vertical Cask Transporter (VCT) to carry and position the overpacks on the storage pad. The VCT and all components associated with the lifting and transport of the overpacks will prevent the drop of an overpack by design compliance with ANSI N14.6 combined with the use of redundant drop protection features, including hydraulic check valves and wedge locks or by enhanced safety margins. Therefore, the 11-inch drop event is not applicable to VY.

Applicable Loads & Combinations

The load combinations used in the design of the reinforced concrete ISFSI pad are in compliance with the HI-STORM 100 FSAR and NUREG-1536. 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.

Design Basis Earthquake (DBE) Spectra

The response spectra used as input at the bedrock elevation under the pad was the VY design basis DBE horizontal and vertical (2/3 of the horizontal) spectra at 5% damping. The ZPA for the horizontal spectrum is defined as 0.14 g. This was used as the "target" spectra for the development of the required time histories.

Finite Element Design/Analysis Models

Analysis of the ISFSI pad was performed using finite element techniques. The computer code used in this analysis was ANSYS.

As demonstrated by the above analyses, all HI-STORM 100 CoC and FSAR analysis requirements are met for normal storage, seismic, tipover, and sliding considerations:

- The requirements of HI-STORM 100 FSAR, Table 2.2.9, Parameter Set "A" were met
- The load combinations used in the design of the reinforced concrete ISFSI pad are in compliance with the HI-STORM 100 FSAR and NUREG-1536
- Seismic evaluations addressed the effects of soil structural interaction and liquefaction

Conclusion

The VY ISFSI pad is adequately designed to support both static and dynamic loads of the stored casks for loading conditions ranging from a single HI-STORM 100 cask to a fully-loaded 40-cask configuration. The analyses appropriately considered potential amplification of earthquakes through soil-structure interaction and evaluated the potential for soil liquefaction and other instability due to and vibratory ground motion. The ISFSI pad is designed for the loads and load combinations specified in NUREG-1536. The analyses show that the ISFSI pad design meets the requirements of 10 CFR Part 72 and Holtec's HI-STORM 100 FSAR and CoC for pad stability. The ISFSI haul path was also evaluated and it was determined that it is adequate to support loads associated with the tracked VCT carrying a HI-STORM 100 storage cask. Therefore, the requirements of 10 CFR 72.212(b)(5)(ii) and the HI-STORM licensing basis are met.

3.41 §72.212(b)(5)(iii) – Dose Limitations per §72.104

The general licensee must perform written evaluations, before use and before applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which 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 under §72.210.

A radiological dose rate evaluation was conducted to confirm that the ISFSI site has adequate standoff distances for maintaining dose rates in compliance with the criteria in 10 CFR 72.104(a) at the ISFSI controlled area boundary. The discussion that follows demonstrates that doses to an individual beyond the controlled area boundary from a) ISFSI gaseous effluents, b) radiation from casks at the ISFSI, and c) radiation from the VY, are within the limits of 10 CFR 72.104(a).

10 CFR 72.104(a)

10CFR72.104 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.

Chapter 5 of the HI-STORM 100 Cask System FSAR 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 a design loaded cask at 47,500 MWD/MTU burnup and 3-year cooling. The results of the analysis are shown in Table 5.4.7 of the FSAR.

A site-specific analysis (Holtec Report HI-2114926, Revision 1 – VY-RPT-11-00015) was conducted for the VY ISFSI to determine the dose associated with loading the first 13 casks. The analysis assumes high-density concrete in the overpacks and a conservative placement of the casks on the pad relative to the off-site boundary. The analysis determined, based on regionalized loading of the MPC, that the dose contribution from the 13 casks was 1.86 mr/year and based on uniform MPC loading the dose contribution would be 3.30 mr/year.

VY calculates the dose from all plant sources per the ODCM methodology at location DR-53. For 2011 the total dose from all plant sources was determined to be 16.21 mrem. Conservatively, adding the dose associated with the 13 casks results in a total dose of 18.07 mrem/year for regionalized loading and 19.51 mrem/year for uniform loading.

Additionally, VY reports the results of direct exposure measurements taken at the west site boundary to the Vermont Department of Health [Letter from S.P. Skibniowsky (VY) to Dr. William Irwin (VY Department of Health), dated 1/6/12]. For 2011, the direct exposure measurement was 15.25 milliRoentgen which is equivalent to 9.15 millirem (.6 X 15.25). This measurement demonstrates that the ODCM methodology used to determine dose is conservative.

10 CFR 72.104(b) and (c)

The design of the MPCs in the HI-STORM 100 cask system ensures that gaseous effluent releases are not credible for the reasons discussed above. Fuel assembly selection and DFS loading operations are both conducted such that personnel doses are maintained as low as reasonably achievable (ALARA).

Section 5.7 of Appendix A of the HI-STORM 100 Cask System CoC places specific requirements on the cask user to ensure the cask is loaded properly and that dose will be ALARA. See Section 4.2.4 of this report for additional discussion of the Radiation Protection Program.

Conclusion

The above radiological dose evaluation demonstrates that there is reasonable assurance that the annual dose equivalent to any real individual who is located beyond the ISFSI controlled area boundary during normal operations and anticipated off-normal occurrences will not exceed 10 CFR 72.104(a) dose limits, including doses from the ISFSI and the VY plant (the other fuel cycle operations in the region). Also, per 10 CFR 72.104(b) and 10 CFR 72.104(c) requirements, the combination of HI-STORM cask design features, Entergy procedures, and the VY Radiation Protection Program will meet ALARA objectives for radioactive materials in effluents and direct radiation levels associated with cask loading and storage operations to ensure that 10 CFR 72.104(a) dose limits will be met.

Therefore, Entergy-VY complies with the requirements of 10 CFR 72.212(b)(5)(iii).

3.42 §72.212(b)(6) – Review of the Cask FSAR and SER

The general licensee must review the Safety Analysis Report referenced in the CoC or amended CoC 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)(5) of this section.

The HI-STORM FSAR, Revision 4, and the NRC's Safety Evaluation Report (as supplemented through CoC Amendment 4) have been reviewed and a determination made that the reactor site parameters at VY 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 certain parameters required to be addressed by the CoC are addressed elsewhere in this report.

Section 5 of this report provides a chapter-by-chapter assessment of deviations between requirements in the cask FSAR and implementation of dry storage at the VY site. All cask FSAR requirements of the licensee that are not listed in Section 5 of this report are met. Deviations from cask FSAR requirements have been evaluated under the VY 10 CFR 72.48 program, as applicable.

Ambient Temperature

See Sections 4.3.6.1 and 4.3.6.2 of this report.

Earthquake

See Sections 3.40 (ISFSI pad design) and 4.3.6.3 (cask sliding and tipping) of this report.

Flooding

See Section 4.3.6.4 of this report

Fire and Explosion

See Section 4.3.6.5 of this report.

Tornado Wind and Missiles

A review of the required design values indicates that the VY site-specific tornado wind and missile design criteria are enveloped by the Holtec generic design criteria as listed in Holtec FSAR Tables 2.2.4 and 2.2.5. Holtec analyzed a maximum tornado wind speed of 360 mph (290 mph rotational and 70 mph translational) with a pressure drop of 3 psi. The VY site tornado parameters include a tornado with a 300 mph maximum wind speed with a 3 psi pressure drop.

Section 2.3.6 of the VY UFSAR defines the maximum tornado wind speed as 300 mph. VY UFSAR Section 12.2.1 specifies that Class I structures are designed against penetration by tornado-generated missiles. The missiles that were considered were a 4 inch x 4 inch x 16 foot long wood post and a 2 inch x 12 inch x 16 foot long wood plank. The following is a comparison of the VY site specific missiles and the Holtec HI-STORM design missiles:

	Missile description	Mass (lbs)	Velocity (mph)	Comment
Holtec FSAR Missiles	Automobile	3960	126	High kinetic energy missile which deforms on impact
	8 inch diameter rigid solid steel cylinder	275	126	Rigid missile to test penetration resistance
	1 inch diameter steel sphere	0.48	126	Small rigid missile to pass through any opening.
VY UFSAR Missiles	4 inch x 4 inch x 16 foot long wood post	89	300	
	2 inch x 12 inch x 16 foot long wood plank	134	300	

The Holtec eight-inch diameter solid steel cylinder was used to verify the penetration resistance of the storage cask. The VY wooden missiles do not have the same capability of the solid steel missile to penetrate the cask. EPRI testing (EPRI NP-440, July 1997) demonstrated that a large portion (50-70 inches) of a wood pole missile disintegrated into splinters upon impact with 12 and 16 inch reinforced concrete panels. The faces of the panels were undamaged. The storage cask wall is 27.5 inch thick high density concrete confined within 1" thick steel inner and outer shells (total thickness of wall is 29.5 inches). Therefore, it is concluded that the VY wood missiles will disintegrate into splinters upon impact with the cask causing no significant damage.

However, the wood missiles will also transfer their kinetic energy to the cask. The VY wooden missiles are comparable to the HI-STORM high kinetic energy missile which deforms on impact (automobile).

- (Kinetic energy of 2 x 12 x 16 wood plank) / (Kinetic energy of automobile) = $[(300)^2 * (134)] / [(126)^2 * (3960)]$

- (Kinetic energy of 2 x 12 x 16 wood plank) / (Kinetic energy of automobile) = 0.192

The kinetic energy imposed on the storage cask by a VY design missile is only 19.2% of the HI-STORM design automobile missile. The HI-STORM high energy missile bounds the VY wooden missiles.

VY does not have any design missiles comparable to the HI-STORM solid steel rigid missile to test penetration or a small rigid missile to pass through any cask opening. Therefore, the HI-STORM design tornado missile analysis envelopes the VY UFSAR missiles in relationship to the design of the HI-STORM casks.

Soil Temperature

HI-STORM 100 FSAR Section 2.2.1.4 and Table 2.2.2 require that the normal soil temperature (bounding annual average) does not exceed 77°F. Soil temperature is not monitored at the VY site. The VY ISFSI is located in a relative cold climate. BY engineering judgment, it can be assumed that the average soil temperature through the year is well below 77°F

Hurricane

The VY site is located well west of the Atlantic Ocean and far north of the tropics. Hurricanes are not a threat to the VY ISFSI. Further, the 360 MPH design wind speed for the HI-STORM cask bounds any credible tropical storm-created wind speed at the site.

Tsunami and Seiche

Tsunamis and seiches are not considered credible at VY because they are very low probability events for this ISFSI's geographic location on the Connecticut River in Vernon, Vermont.

Lightning

The HI-STORM 100 system is a large metal/concrete cask stored in an unsheltered ISFSI. If the HI-STORM overpack is hit with lightning, the lightning will discharge through the steel shell of the overpack to ground. The MPC provides the confinement boundary for the spent nuclear fuel. The effects of the lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground via the installed grounding system. The MPC will be unaffected. This event does not result in the release of activity.

Burial Under Debris

The VY ISFSI is not located near potential sources of mudslides or volcanoes that could result in burial of the casks. Therefore, this event is not considered credible for the VY ISFSI.

Snow and Ice Load

The HI-STORM 100 System is designed to withstand a snow and ice loading of 100 pounds per square foot (CFSAR section 2.2.1.6). The VY UFSAR, Section 2.3.5.3, documents that the worst case snow loading is 70 pounds per square foot. Based on this, the HI-STORM design conditions bound the site specific UFSAR requirements.

3.43 §72.212(b)(7) – Changes to 212 Evaluations

The general licensee must 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.

As indicated in Entergy Procedure EN-LI-112, any changes to the written evaluations required by 10CFR72.212(b)(6) will be evaluated using the requirements of §72.48(c) and a copy of this record will be retained per Entergy Procedure EN-AD-103 until spent fuel is no longer stored under the general license issued under §72.210.

3.44 §72.212(b)(8) – Changes to 10 CFR Part 50 Technical Specifications

The general licensee must, before 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)(5) of this section.

Several design modifications, including an Engineering Change (EC) that governed the ISFSI installation, have received a 10CFR50.59 review, and have been implemented in support of dry fuel storage at VY. None of these modifications required a change to the VY operating license or technical specifications. An integrated 10CFR50.59 Evaluation (included in EC 1604, "2008 Loading Campaign") was completed on the initial ISFSI installation, and it was concluded that an NRC review is not required. Subsequent changes have been evaluated under 10CFR50.59.

3.45 §72.212(b)(9) – Protection Against Radiological Sabotage

The general licensee must 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(e) 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 the purposes of this general license, searches required by §73.55(h) 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(i)(3) of this chapter as applied to a new protected area may be provided by a guard or watchman on patrol in lieu of video surveillance technology;

(v) For the purpose of this general license, the licensee is exempt from requirements to interdict and neutralize threats in §73.55 of this chapter; and

(vi) Each general licensee that receives and possesses power reactor spent fuel and other radioactive materials associated with spent fuel storage shall protect Safeguards Information against unauthorized disclosure in accordance with the requirements of §73.21 and the requirements of §73.22 or §73.23 of this chapter, as applicable.

The ISFSI is located within the VY protected area. During transport to the ISFSI pad from the Reactor Building, the overpack will not exit the protected area. The site Security Plan and Physical Protection Program provide the appropriate controls during this transient activity. The VY Security Plan has been updated to include the VY ISFSI in accordance requirements of 10 CFR 50.54(p).

In addition to the above security measures, VY has applied interim safeguards and security compensatory measures, and implemented additional security measures associated with access authorization for the VY ISFSI (EC 1604, "2008 Loading Campaign").

3.46 §72.212(b)(10) – Review of the Emergency Plan, Quality Assurance Program, Training Program, and Radiation Protection Program

Each general licensee must 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 VY 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. 10 CFR 50.47(b) lists the sixteen planning standards that must be met in the Emergency Plan. The VY Emergency Plan was reviewed to determine if changes were required to support implementation of the Holtec HI-STORM 100 System at the VY ISFSI will decrease the effectiveness of the Emergency Plan.

Construction and implementation of the ISFSI was determined to not decrease the effectiveness of the VY Emergency Plan or implementing procedures as it relates to Part 50 activities. However, the HI-STORM 100 FSAR does contain provisions to establish procedures to:

- Address establishing emergency action levels and implementation of the emergency action program. (Section 8.0)

- Include written procedures to account for such things as emergency response. (Section 8.0)
- Address removal of material blocking the air inlet ducts prior to the fuel clad reaching its short-term temperature limit.
- Include emergency action plan provisions for corrective actions for cask burial under debris

The VY Emergency Plan and Procedure AP 3125 were revised to address ISFSI action levels and implementation of the emergency response. Site Procedures OP 3127 and DP 3201 address corrective actions for abnormal events (e.g., blocked HI-STORM 100 air inlet ducts). When an event is declared, the Plant Emergency Director's/Site Recovery Manager's responsibility is to make the notification and 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

ISFSI-related activities at VY through 2014 that were subject to the site 10 CFR 72 QA program were conducted in accordance with the Entergy fleet QAPM established under 10 CFR 50, Appendix B. The Entergy fleet QAPM was reviewed and revised as needed to assure compliance with the requirements of 10 CFR 72 for the handling, transporting and storage of dry fuel storage canisters. After 2014, in anticipation of the permanent shutdown of the power plant, a VY site-specific QAPM was developed and now governs applicable ISFSI-related activities. Entergy-VY informed the NRC of this change by letter BVY 14-072, dated October 27, 2014.

The Holtec HI-STORM 100 Storage System quality assurance requirements in the HI-STORM 100 FSAR and CoC 1014 impose the requirements of 10CFR72 on both licensees and certificate holders. The VY QAPM applies to all activities associated with structures, systems, and components which are safety related or controlled by 10CFR72. The methods of implementation of the requirements of the VY QAPM are commensurate with the item's or activity's importance to safety. The applicability of the requirements of the VY QAPM to other items and activities is determined on a case-by-case basis. The VY QAPM implements 10CFR50, 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 100 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 VY quality-related procedures.

Training Program

The Holtec HI-STORM 100 Storage System training program requirements are found in the HI-STORM 100 FSAR and CoC 1014, which invoke the requirements of 10CFR72 and require cask design-specific topics for personnel training. The VY Training Department used the Systematic Approach to Training, which is based on 10CFR50.120, to develop and revise training programs to meet these requirements. The training program addresses all training requirements identified in the HI-STORM FSAR and CoC 1014, as well as requirements in other parts of the 10CFR72 regulations. Corresponding training modules shall include the following elements, at a minimum:

- HI-STORM 100 System Design (overview)
- ISFSI Facility Design (overview)
- SSCs Important to Safety (overview)
- HI-STORM 100 FSAR (overview)
- NRC SER (overview)
- CoC 1014 Conditions
- HI-STORM 100 Tech Specs, Approved Contents, Design Features
- Regulatory Requirements (e.g., 10CFR72.48, 10CFR72 Subpart K, 10CFR 20, 10CFR73)
- Required Instruments and use
- Operating Experience Reviews
- HI-STORM and ISFSI Procedures (list is in Appendix B)

The Systematic Approach to Training was employed to determine what positions and what level of training was required for the VY staff.

Radiation Protection Program

The Holtec HI-STORM 100 Storage System radiological protection requirements are found in the HI-STORM 100 FSAR and CoC 1014, which invoke the requirements of 10CFR72 and provide cask-specific requirements. The VY Radiological Protection Program has been reviewed and implementing procedures developed or modified as necessary to address dry spent fuel cask loading, unloading, and storage operations. In addition, the requirements in HI-STORM 100 CoC, Appendix A, Section 5.7 "Radiation Protection Program," have been addressed.

3.47 §72.212(b)(11) – Maintain a Copy of the CoC

The licensee shall maintain a copy of the CoC and, for those casks to which the licensee has applied the changes of an amended CoC, the amended CoC, and the documents referenced in such Certificates, 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, conditions, and specifications of the CoC and, for those casks to which the licensee has applied the changes of an amended CoC, the terms, conditions, and specifications of the amended CoC, including but not limited to, the requirements of any AMP put into effect as a condition of the NRC approval of a CoC renewal application in accordance with § 72.240.

HI-STORM Certificate of Compliance No. 1014 and documents referenced therein are controlled and maintained by the document control process until turnover to VY. Once turned over to VY the documents are maintained per Procedure EN-AD-103.
For additional information (specific documents, drawings, ECOs and 72.48 screens and evaluations) on any of the ISFSI components refer to the referenced Engineering Report identified in the table in Section 2.C of this report. VY has not applied the changes in an amended CoC to casks previously loaded under an earlier CoC amendment. VY does not have any casks in service that are being operated under a renewed CoC.

3.48 §72.212(b)(12)(i) – Cask Vendor/Lessor Name and Address

The licensee shall 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 name and address of the cask vendor or lessor.

Entergy uses the Holtec International HI-STORM 100 Cask System at VY. The vendor address is:

Holtec International
1 Holtec Drive
Marlton, NJ 08053

3.49 §72.212(b)(12)(ii) – Listing of Spent Fuel in the Cask

The licensee shall 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 listing of spent fuel stored in the cask.

All Special Nuclear Material (SNM) possessed by VY is closely controlled, inventoried, and reported semi-annually per NRC requirements (10CFR74). VY approves all fuel assemblies for dry storage and ensures all related documentation (DOE/NRC 741 & 742 forms) is processed and maintained per the requirements of Entergy Procedures EN-NF-104, EN-NF-200, and EN-NF-201. SNM inventory updates will be made in a timely manner to reflect the exact location of any reportable quantity of SNM. The location of specific fuel assemblies loaded into a given cask is determined and documented in accordance with Procedures EN-DC-215, EN-RE-210 and EN-DC-212.

3.50 §72.212(b)(12)(iii) – Maintenance of Records

The licensee shall accurately maintain the record provided by the cask supplier for each cask that shows, in addition to the information provided by the cask vendor, any maintenance performed on the cask.

Maintenance performed on the casks will be documented in accordance with the maintenance procedures under the controls provided in the VY QAPM.

3.51 §72.212(b)(13) – Procedures

The licensee shall conduct activities related to storage of spent fuel under this general license only in accordance with written procedures.

It is the policy of VY, and required by the VY QAPM, that VY use approved written procedures for important to safety activities. See Section 2.A of this report for a listing of DFS-related procedures.

3.52 §72.212(b)(14) – Availability of Records

The licensee shall make records and casks available to the Commission for inspection.

Records and documentation are stored at VY in accordance with the VY Document Control and Records Management Program, and are available to the NRC for review upon request.

3.53 §72.212(c) – Record Requirements

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.

Maintenance and testing will be performed via approved procedures including those identified in EN-MA-100 that provide documentary evidence that the maintenance testing has been conducted under the VY QAPM, which has been approved by the NRC. Each facility's component/equipment database provides the component classification for input into any required maintenance and testing.

3.54 §72.212(d) – Cask Sale, Lease, or Loan

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.

A record of casks that are sold, leased, loaned, or otherwise transferred to another registered user will be maintained by VY, as required

3.55 §72.212(e) - Fees

The general licensee shall pay fees for inspections related to spent fuel storage under this general license as shown in §170.31 of this chapter.

NRC inspection fees are incurred and paid by Entergy-VY on a case-by-case basis.

3.56 §72.214 – List of Approved Spent Fuel Storage Casks

The regulation lists all approved spent fuel storage cask designs and the associated CoC number.

The Holtec HI-STORM 100 System is approved for storage of spent fuel under the conditions specified in the HI-STORM 100 System CoC.

Certificate Number: 1014
SAR Submitted by: Holtec International
SAR Title: Final Safety Analysis Report for the HI-STORM 100 Cask System
Docket Number: 72-1014
Certificate Expiration Date: May 31, 2020

Model Number: HI-STORM 100

3.57 §72.218 – Termination of Licenses

(a) The notification regarding the program for the management of spent fuel at the reactor required by §50.54(bb) of this chapter must include a plan for removal of the spent fuel stored under this general license from the reactor site. The plan must show how the spent fuel will be managed before starting to decommission systems and components needed for moving, unloading, and shipping this spent fuel.

(b) An application for termination of a reactor operating license issued under 10 CFR part 50 and submitted under §50.82 of this chapter, or a combined license issued under 10 CFR part 52 and submitted under § 52.110 of this chapter, must contain a description of how the spent fuel stored under this general license will be removed from the reactor site.

(c) The reactor licensee shall send a copy of submittals under § 72.218(a) and (b) to the administrator of the appropriate Nuclear Regulatory Commission regional office shown in appendix D to part 20 of this chapter.

No action required at this time. Entergy-VY has no plans to terminate its Part 50 license at this time. An active Part 50 license maintains an active Part 72 general license.

3.58 §72.220 – Violations

This general license is subject to the provisions of § 72.84 for violation of the regulations under this part.

No action required.

4 - COMPLIANCE WITH HI-STORM 100 CASK SYSTEM CERTIFICATE OF COMPLIANCE

4.1 Certificate of Compliance Conditions

Compliance with Amendment 2 of the Holtec HI-STORM 100 System Certificate of Compliance (CoC) terms, conditions, and specifications is discussed below.

4.1.1 Condition 1 – Cask

This condition describes the HI-STORM 100 System and its three main components: the HI-STORM overpack, the HI-TRAC transfer cask, and the Multi-Purpose Canister (MPC). The details of the CoC are not repeated here for convenience.

The cask system and components that will be used for the VY ISFSI are the MPC-68, the HI-TRAC 100D transfer cask and the HI-STORM 100S Version B storage overpack. These components were designed and fabricated in accordance with CoC No. 1014, Amendment 2 and HI-STORM FSAR, Revision 4 requirements.

4.1.2 Condition 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.

Written procedures for the above activities are identified in Appendix B of this evaluation report. These procedures are consistent with the technical basis provided in Chapter 8 of the HI-STORM 100 FSAR.

4.1.3 Condition 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.

Written cask acceptance tests were developed consistent with the technical basis described in Chapter 9 of the FSAR; and are included in the list of procedures in Section 2.A of this evaluation report. Maintenance activities on the casks will be performed in accordance with VY's maintenance procedures; and the specific activities identified in Chapter 9 of the FSAR have been integrated into the VY Maintenance program.

4.1.4 Condition 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.

Entergy notified NRC of its intent to use the Entergy fleet QAPM for dry spent fuel storage activities at its VY ISFSI as permitted in Section 72.140(d) via letter BVY 07-031 "Intent to use 10CFR50 Appendix B Quality Assurance Program for Independent Spent Fuel Storage Installation Notification Pursuant to 10CFR72.140(d)", dated April 18, 2007. ISFSI-related activities were governed by the Entergy fleet QA program through approximately mid-October, 2014. In anticipation of the permanent shutdown of the VY power plant, Entergy has created a VY-specific QAPM. Beginning in October 2015, the VY-specific QAPM is now used to govern applicable ISFSI-related activities. Entergy informed the NRC of this change via letter BVY 14-072, dated October 27, 2014.

4.1.5 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 Section 3.5 of Appendix B to this certificate, as applicable.

Lifting of the loaded HI-TRAC and transfer of the MPC from the HI-TRAC to the HI-STORM 100 are integral to a structure (Reactor Building) governed by 10CFR50 regulations. The Reactor Building crane, which performs the lifts in the Reactor Building, is single-failure-proof. All lifts will be in accordance with the site's heavy load control program PP 7023, "Control of Heavy Loads Program Document," which is consistent with VY commitments to NUREG-0612.

A Low Profile Transporter (LPT) will be used to transport the loaded HI-STORM 100 overpack from the Reactor Building to the Containment Access Building (CAB). The LPT is a device providing support from underneath (i.e. steel platform mounted on rollers) with no lifting capabilities.

The Vertical Cask Transporter (VCT) is utilized to lift the loaded HI-STORM 100 overpack from the LPT and transport it from the CAB to the storage pad for final placement. The VCT and all components associated with the lifting and transport of the overpacks will prevent the drop of an overpack by design compliance with ANSI N14.6, combined with the use of redundant drop protection features including hydraulic check valves and wedge locks or by enhanced safety margins. VY is, therefore, in compliance with Section 5.5 of Appendix A by meeting the requirements specified in Section 5.5.a.3 as allowed by Note 3 on Table 5-1.

4.1.6 Condition 6 – Approved Contents

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

The fuel selection specifications are implemented by procedure EN-DC-215, "Fuel Selection for Holtec Dry Cask Storage." This procedure ensures that the requirements of CoC 1014, Appendix B, Section 2, "Approved Contents" are satisfied.

4.1.7 Condition 7 – Design Features

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B to this certificate.

Entergy-VY has reviewed the site, cask, and ancillary equipment against the requirements in CoC Appendix B, Section 3, "Design Features." See Section 4.3 of this report for additional details.

4.1.8 Condition 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.

4.1.9 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 thermally 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.

Thermal Test

The portion of this CoC requirement pertaining to cask heat load applies to all general licensees using the Holtec HI-STORM 100 System. VY has confirmed that this CoC requirement has been successfully implemented by other HI-STORM 100 System users based on the heat loads of MPC-68/68F/68FFs to date. Energy Northwest loaded an MPC-68 series canister with a heat load greater than 16 kW at Columbia Generating Station. Therefore, no temperature data are required to be taken, and no reports need to be submitted.

Supplemental Cooling

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. VY does not plan to load high burnup fuel at this time. Therefore, this requirement of the CoC is not applicable.

4.1.10 Condition 10 – Pre-Operational Training 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.)*

* Not applicable to VY at this time.

The demonstrations were performed in an alternate step sequence from the actual procedures. The demonstration criteria identified in the Certificate of Compliance Condition 10 are satisfied in the following defined scenarios:

- Scenario #1 – MPC Inter-Cask Transfer Operations (Stack-up)
- Scenario #2 – HI-STORM Handling and Transport Operations
- Scenario #3 – HI-TRAC/MPC Spent Fuel Pool Placement and Fuel Loading
- Scenario #4 – MPC Sealing Operations
- Scenario #5 – MPC Welding Operations
- Scenario #6 – MPC Unloading Operation

The extent to which the above activities are simulated is documented in the Dry Fuel Storage Project Cask Handling Operation NRC Demonstration Plan. In some cases (e.g., welding) the activity was simulated by setting up welding equipment but not actually completing the activity. The extent of simulation was discussed with the NRC to ensure that the intent of the CoC condition was satisfied.

4.1.11 Condition 11 – Exemption from 10 CFR 72.236(f)

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. This CoC condition documents an exemption granted to the CoC holder permitting use of the Supplemental Cooling System (SCS), when required by the technical specifications. VY does not require a supplemental cooling system at this time.

4.1.12 Condition 12 – Authorization

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 specific MPC-68s and HI-STORM 100S Version B overpacks to be loaded under CoC Amendment 2 have been certified for use under CoC Amendment 2 as documented in the

Component Completion Records (CCRs) for those serial number components. The HI-TRAC transfer cask is a one-time fabrication that was also certified for use at VY under CoC Amendment 2. VY HI-TRAC transfer cask serial number 1026-6 is certified for use under CoC Amendment 2 as shown in its CCR. The CCRs for the MPCs and overpacks, and this FSAR section for the HI-TRAC transfer cask demonstrate compliance with this CoC condition.

4.2 CoC Appendix A – Technical Specifications

Compliance with the Holtec HI-STORM 100 System CoC, Appendix A, "Technical Specifications" at VY, is discussed below.

4.2.1 Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs)

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 VY Part 50 standard technical specifications. No specific actions or implementation procedures are required. Personnel are trained on the use of the LCO and SR 3.0 series of technical specifications.

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. It refers to Tables 3-1 and 3-2 for specific limits.

With regard to MPC cavity drying, VY choose to use a vacuum drying system. Therefore, the drying acceptance criteria for the Forced Helium Drying system in Table 3-1 of HI-STORM 100 CoC Appendix A do not apply. The helium backfill pressure range for MPC-68/68F/68FF in Table 3-2 of HI-STORM 100 CoC Appendix A is used as the acceptance criterion rather than the "gram-moles/liter" acceptance criterion. Compliance with this LCO is controlled by procedure OP 2223 "MPC Fuel Loading Preparations and Sealing Operation." See Section 1.3 for a summary of the cask loading operational sequence where this LCO would apply.

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. The LCO permits either temperature monitoring or inspections of the air inlet and outlet vent duct.

For loaded HI-STORM overpacks stored on the ISFSI pad, a temperature monitoring system is used to ensure adequate fuel decay heat removal. Significant inlet or outlet air duct blockage or damage would reduce air flow and increase the temperature rise of the cooling air. This periodic surveillance is controlled by ESOMS Electronic operator rounds. If any air duct blockage or damage is found, the air ducts will be restored to an operable condition within the completion time established for the condition in the LCO.

4.2.1.4 LCO 3.1.3 – Fuel Cooldown/MPC Cavity Re-flooding

This LCO establishes requirements for MPC cavity bulk helium temperature prior to re-flooding of the MPC. A maximum cavity helium temperature limit of 200°F is mandated by the LCO prior to initiating re-flooding operations.

Compliance with this LCO is controlled by procedure OP 2225, "MPC Unloading Operations." See Section 1.4 for a summary of the cask unloading operational sequence where this LCO would apply.

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 > 45 GWD/MTU) are loaded into the MPC.

Currently, VY does not plan to load any high burnup fuel assemblies. Therefore, this LCO is not applicable to dry fuel storage at VY at this time. Loading of casks meeting the CoC criteria for use of the SCS will require a revision to this document and site procedures, and training in the use of the SCS.

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.

Compliance with this LCO is controlled by procedure OP 2224, MPC Transfer Operations and HI-STORM Transport.

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 (PWR) plants.

Because VY is a BWR plant, this LCO does not apply to dry fuel storage at VY.

4.2.2 Section 5.4 - Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

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.

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.

An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

An annual report shall 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 VY Procedure AP 0069, "Routine Reports Due to State and Federal Agencies."

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.).

Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport route conditions.

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 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.

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 specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.

Lifting of the loaded HI-TRAC and transfer of the MPC from the HI-TRAC to the HI-STORM 100 are integral to a structure (Reactor Building) governed by 10 CFR 50 regulations. The Reactor Building crane, which will perform the lifts in the Reactor Building, is single-failure-proof. All lifts will be in accordance with the site's heavy load control program PP 7023, which is consistent with VY commitments to NUREG-0612.

A Low Profile Transporter (LPT) is used to transport the loaded HI-STORM 100 overpack from the Reactor Building to the Containment Access Building (CAB). The LPT is a device providing support from underneath (i.e., steel platform mounted on rollers) with no lifting capabilities.

The Vertical Cask Transporter (VCT) is then utilized to lift the loaded HI-STORM 100 overpack from the LPT and transport it from the CAB to the storage pad for final placement. The VCT and all components associated with the lifting and transport of the overpacks will prevent the drop of an overpack by design compliance with ANSI N14.6, combined with the use of redundant drop protection features including hydraulic check valves and wedge locks or by enhanced safety margins. VY is, therefore, in compliance with Section 5.5 of Appendix A of the CoC by meeting the requirements specified in Section 5.5.a.3 as allowed by Note 3 on Table 5-1.

b. For the transport of OVERPACKS to be anchored to the ISFSI pad...

Subsection b of this specification is not applicable to VY because VY does not use anchored casks.

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)* [now §72.212(b)(5)(iii)], *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 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:*

   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 Section 5.7.3, the measured dose rates on a loaded OVERPACK shall not exceed the following values:*

   a. *20 mrem/hr (gamma + neutron) on 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 Section 5.7.8 for comparison against the limits established in Section 5.7.3 or Section 5.7.4, whichever are lower.*

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:*

   a. *Administratively verify that the correct contents were loaded in the current 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 72.104 to be exceeded.*
   c. *Perform a written evaluation within 30 days to determine why the surface dose rates were exceeded.*

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 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 has been performed in accordance with the HI-STORM CoC Appendix A, Sections 5.7.2 and 5.7.3. Based on this analysis, the overpack and transfer cask surface dose limits to be used for the VY site are as follows:

CoC Appendix A Reference		**Location**	**Site-Specific Calculated Overpack Surface Dose rate (mrem/hr)**		
			Neutron	**Gamma**	**Total**
5.7.3	a(1)	The top lid of the transfer cask	31.83	17.80	49.63
	a(2)	The top lid of the overpack i. center of top lid ii. middle of top lid	0.78 0.567	0.74 0.77	1.52 1.44
5.7.3	b(1)	The side of the transfer cask (mid-height)	32.54	96.40	128.94
	b(2)	The side of the overpack i. mid-height ii. 60 in. above mid-height iii. 60 in. below mid-height	0.41 0.04 0.41	0.54 0.39 0.53	0.95 0.43 0.95
5.7.3	c(1)	The inlet air duct of the overpack	5.02	4.96	9.99
	c(2)	The outlet air duct of the overpack	1.02	1.64	2.66

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.

VY procedure OP 2530, "Radiological Requirements for the HI-STORM 100 Dry Fuel Storage System" 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, Section 5.7.6 and 5.7.7 (if required) will be implemented.

4.3 CoC Appendix B – Approved Contents and Design Features

Section 2.0 of Appendix B to the HI-STORM CoC, "Approved Contents," establishes the limits on fuel and other hardware to be stored at the ISFSI in the HI-STORM System. The full text of this CoC section is not repeated here due to its length.

Entergy-VY procedure EN-DC-215, Fuel Selection For Holtec Dry Cask Storage, is used to select fuel assemblies for storage in the HI-STORM 100 System that meet all applicable requirements of CoC No. 1014, Amendment 2, Appendix B, Section 2.

Section 3.0 of Appendix B to the HI-STORM CoC, "Design Features," establishes requirements for site parameters and design features of the ISFSI, cask system, and certain ancillary equipment.

Section 3.0 to Appendix B of the CoC is addressed in Sections 4.3.3 through 4.3.10 below.

4.3.1 Section 1.0 – Definitions

This CoC section provides terms specifically defined for use within the cask technical specifications.

The cask operating procedures ensure the definitions are implemented correctly. See Section 2.A of this report for a list of dry cask operating procedures.

4.3.2 Section 2.0 – Approved Contents

4.3.2.1 Section 2.1.1 – Fuel Specifications and Loading Conditions

CoC Appendix B, Section 2.1.1.a specifies that fuel and other contents meeting the definitions of INTACT FUEL ASSEMBLIES or DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE and the limits in Table 2.1-1 and other referenced tables may be stored in the HI-STORM 100 System. This CoC section is not repeated here due to its length.

Entergy-VY procedure EN-DC-215 ensures that any fuel and other contents loaded into the storage cask meets the requirements in this CoC section.

CoC Appendix B, Section 2.1.1.b through 2.1.1.d address fuel with stainless steel fuel cladding and certain BWR fuel.

Because VY fuel does not have stainless steel fuel cladding or the other fuel types mentioned, these CoC requirements do not apply. No further action is required.

4.3.2.2 Section 2.1.2 – Uniform Fuel Loading

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.

Entergy-VY procedure EN-DC-215 ensures that any fuel and other contents loaded into the storage cask meets the requirements in this CoC section.

4.3.2.3 Section 2.1.3 – Regionalized Fuel Loading

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 INTACT 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.

Entergy-VY procedure EN-DC-215 ensures that any fuel and other contents loaded into the storage cask meet the requirements in this CoC section

4.3.2.4 Section 2.2 – Violations

If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

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 VY procedure AP 0010, "Situational Reporting Requirements."

4.3.2.5 Section 2.3 – Not Used

4.3.2.6 Section 2.4 – Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel

CoC Appendix B, Section 2.4 and its tables provide a methodology for licensees to establish limits on the decay heat, burnup, and cooling time for ZR-clad fuel for each fuel storage location in all MPC models, for uniform and regionalized fuel loading configurations. This CoC section is not repeated here due to its length.

Entergy-VY procedure EN-DC-215 ensures that any fuel and other contents loaded into the storage cask meet the requirements in this CoC section.

4.3.3 Section 3.1 – Site and 3.1.1 – Site Location

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.

Entergy is a Part 50 license holder for VY (DPR-28) and therefore qualifies as a general licensee to use the HI-STORM 100 System.

4.3.4 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 and 3.2.3 through 3.2.5 pertain to the MPC-24, MPC-68F, MPC-24E/EF, and MPC-32/32F models, respectively. The MPC-68F is a BWR model not licensed for use by VY. The MPC-24, MPC-24E/EF, and MPC-32/32F are PWR MPC models. Therefore, these CoC sections are not applicable to the VY ISFSI and are not discussed further in this report. CoC Section 3.2.2 is applicable and is discussed in Section 4.3.4.1 of this report.

4.3.4.1 Section 3.2.2 – MPC-68/68FF

1. Fuel cell pitch $\geq$ 6.43 inches
2. ^{10}B loading in the neutron absorbers: $\geq$ 0.0372 g/cm^2 (Boral) and $\geq$ 0.0310 g/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-68 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.

4.3.4.2 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 VY fuel assembly design is 176.2 inches long with a maximum active fuel length of 150 inches. In accordance with HI-STORM FSAR, Table 2.1.10, no fuel spacers are required.

4.3.4.3 Section 3.2.7 - METAMIC B4C Content

The B_4C content in METAMIC shall be $\leq$ 33.0 wt.%.

The limit is verified to be met by Holtec International during the MPC fabrication process as documented in the Component Completion Records for the MPCs.

4.3.4.4 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 is verified to be met by Holtec International during the neutron absorber fabrication process as documented in the Component Completion Records for the MPCs.

4.3.5 Section 3.3 – Codes and Standards

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.

Compliance with design codes is the responsibility of Holtec and its fabricator. Compliance is documented in the Component Completion Record for each serial number component.

4.3.5.1 Section 3.3.1 – Alternatives to Codes, Standards, and Criteria

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.

This CoC section and its referenced table set down the NRC-approved alternatives to the ASME code of record for the HI-STORM overpack, MPCs and HI-TRAC transfer cask. No VY action required.

4.3.5.2 Section 3.3.2 – Construction/Fabrication Alternatives to Codes, Standards, and Criteria

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:

1. *The proposed alternatives would provide an acceptable level of quality and safety, or*

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.*

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

This CoC section establishes the process to be used by the CoC holder to request NRC approval of a new or revised alternative to the ASME code of record for the HI-STORM 100 cask system. No VY action required.

4.3.6 Section 3.4 – Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

4.3.6.1 Section 3.4.1 – Maximum Normal Ambient Temperature

The temperature of 80°F is the maximum average yearly temperature.

VY UFSAR, Table 2.3.2, provides temperature data for the Vernon, Vermont area during the time period from 1951 to 1960. The mean daily maximum when averaged for a year is approximately 59 degrees. Therefore, this requirement is considered met.

4.3.6.2 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

VY UFSAR, Table 2.3.2, provides temperature data for the Vernon, Vermont area. The extreme maximum temperature is 100 degrees F, which is less that the 125 degrees F. The extreme minimum is -33 degrees F. which is greater than the -40 degrees F. Therefore, this requirement is met.

4.3.6.3 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), GH, and vertical ZPA, GV, on the top surface of the ISFSI pad, expressed as fractions of 'g', shall satisfy the following inequality:

$$GH + \mu\ GV \leq \mu$$

where μ 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 μ, 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, GH and GV 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 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 tip over 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.

VY chose to perform a dynamic seismic analysis of the cask/ISFSI pad assemblage with appropriate recognition of soil/structure interaction effects to ensure that the casks will not tip over or undergo excessive sliding under the site's Design Basis Earthquake.

The ISFSI storage pad is a reinforced concrete slab founded on engineered fill placed on existing soils. Seismic input at VY is at bedrock which lies from 20 to 30 feet below the pad. Strain compatible soil properties were determined and acceleration time histories were developed at the bedrock elevation. The results of these analyses were used as design input to the soil structural interaction analysis and the assessment of liquefaction potential. The results demonstrated that liquefaction will not occur at VY. The structural interaction analysis output was used as design input to the seismic analysis of the storage pad and the sliding and tipping analysis of the casks on the storage pad.

The sliding and tipping analysis determined that the maximum horizontal displacements of the overpacks for any condition are significantly smaller than half of the free space between adjacent overpacks and also significantly less than the distance between the external edge of the overpack and the edge of the ISFSI storage pad. Therefore, the overpacks will not interact with each other and will not slide off the pad. In addition, this calculation determined that the overpacks will not tip over. The analysis used coefficients of friction ranging from 0.0 to 0.8, which covers the range of potential pad surface conditions (such as due to icing).

4.3.6.4 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.

VY is designed to withstand the effects of the probable maximum flood (PMF). The maximum PMF still water elevation is 252.5 feet mean sea level (MSL). Wave effects, including run-up, could produce flooding as high as 254 feet MSL. The storage pad is at elevation 254 feet MSL and the grade surrounding the pad is at elevation 252 feet MSL. The bottom of the loaded HI-STORM 100 System in position on the ISFSI pad is at an elevation of 254 feet. Therefore, the loaded cask is not submerged or subjected to water flow. The HI-STORM FSAR 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 at VY.

4.3.6.5 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 100 FSAR postulated fire event for the overpack was performed using the following key inputs, as described in HI-STORM 100 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 ISFSI 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. Flammable fuel will be in the vicinity of the overpack during transport from the Reactor Building to the ISFSI pad and at the pad while the cask is being placed in its designated position. Combustible and flammable materials will be under administrative controls and be evaluated.

The fire protection requirements for the ISFSI are contained in 10CFR72.122(c), Subpart F, General Design Criteria, *Protection Against Fires and Explosions.* A review of the 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. The Hazard Evaluation for the ISFSI and Haul Path (Enercon Report ENTV-PR-001, "Hazard Evaluation for the ISFSI and Haul Path – VT-RPT-08-0017) evaluates the fire hazards, and identifies sources of explosions, respectively, involved in the onsite transportation and dry storage of spent fuel in MPCs enclosed in HI-STORM 100 ventilated overpacks at the VY site (see Appendix A to this report). 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 Analysis consisted of the following steps:

1) Identify the design basis fire as established in Holtec HI-STORM 100 Final Safety Analysis Report,

2) Identify the travel path for the HI-STORM 100 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 100 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 100 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 100 outer shell reaches 570°F, which is below the 600°F short-term temperature limit for the outer shell steel specified in HI-STORM 100 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.

Since the ISFSI pad and overpack transport pathway is exterior to and sufficiently separated from plant structures, no automatic fire detection or suppression systems were incorporated into the design of the ISFSI. Several yard fire hydrants are installed along the transport pathway. Two additional fire hydrants are located east of the ISFSI pad. 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.

See Appendix A of this evaluation report for additional details on fire hazards and also details on explosion hazards.

4.3.6.6 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. 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.

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 45 g's at the top of the fuel for the HI STORM 100 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

In accordance with the HI-STORM 100 CoC, the ISFSI owner 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. The VY ISFSI pad is designed in accordance with the "Set A" requirements in HI-STORM 100 FSAR, Table 2.2.9.

The engineered backfill meets the design basis established in the Holtec FSAR. The design basis for the engineered backfill requires the material stiffness of the supporting backfill material to limit fuel deceleration due to a postulated HI-STORM overpack tip-over event. The source materials, in-place properties, and method of placement of the engineered backfill are identified in GZA GeoTechnical Engineering Report, dated January 2004 and satisfy the Holtec design basis requirements. The GZA Report limits the net allowable bearing pressure under the storage pad to 2 tons/ft^2. In accordance with the GZA Report recommendations a minimum 4'0" thick sub-base course was placed under the storage pad to provide frost line protection. Therefore, the storage pad and engineered backfill is founded upon a non-frost susceptible sub-grade.

Because VY specified and verified the properties in HI-STORM 100 FSAR, Table 2.2.9, parameter "Set A", the cask deceleration limit of less than or equal to 45 g's for an overpack tip-over was met.

VY is utilizing a Vertical Cask Transporter (VCT) to carry and store the overpacks on the storage pad. The VCT and all components associated with the lifting and transport of the overpacks will prevent the drop of an overpack by design compliance with ANSI N14.6, combined with the use of redundant drop protection features including hydraulic check valves and wedge locks or by enhanced safety margins. Therefore, the 11-inch drop criterion/restriction is not applicable to VY.

b. For anchored casks...

Not applicable to VY.

4.3.6.7 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 10CFR72.104(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 VY ISFSI and are not credited in the shielding analysis performed to demonstrate compliance with 10 CFR 72.104(a).

4.3.6.8 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 ≥ 0 °F.

Loading and unloading operations will be performed in the Reactor Building where the ambient temperature is controlled to be greater than 0°F. Procedures restrict Vertical Cask Transporter operations with the HI-STORM 100 overpack to temperatures greater than or equal to 10°F and a maximum temperature of 100°F. Procedures OP 2224, "MPC Transfer Operations and HI-STORM Transport" and OP 2225, "MPC Unloading Operations" implement this requirement.

4.3.6.9 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 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.

The VY ISFSI is designed to withstand the effects of the probable maximum flood (PMF). The maximum PMF still water elevation is 252.5 feet mean sea level (MSL). Wave effects, including runup, could produce flooding as high as 254 feet MSL. The storage pad is at elevation 254 feet MSL and the grade surrounding the pad is at elevation 252 feet MSL. The bottom of the loaded HI-STORM System in position on the ISFSI pad is at an elevation of 254 feet. Therefore, the ISFSI pad elevation is above the probable maximum flood level for the site, and flooding of the casks during storage operations is not a concern.

Therefore, there are no postulated site-specific design basis events (e.g., floods) that could potentially result in the blockage of any HI-STORM inlet or outlet air ducts for an extended period of time. Accordingly, this CoC requirement is not applicable to the VY ISFSI.

4.3.7 Section 3.5 – Cask Transfer Facility

This CoC section establishes the design requirements for cask users that employ a Cask Transfer Facility (CTF). The full text of the CoC section is not repeated here due to its length.

Lifting of a loaded TRANSFER CASK and MPC outside of structures governed by 10 CFR Part 50 will not be performed at VY. Thus, VY will not utilize a CTF with the loaded Holtec HI-STORM 100 cask system components. No VY action required.

4.3.8 Section 3.6 – Forced Helium Dehydration System

4.3.8.1 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.

VY uses the vacuum drying method to dry the MPCs. Therefore, these requirements are not applicable to VY.

4.3.8.2 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.

VY uses the vacuum drying method to dry the MPCs. Therefore, these requirements are not applicable to VY.

4.3.8.3 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.

VY uses the vacuum drying method to dry the MPCs. Therefore, these requirements are not applicable to VY.

4.3.9 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 VY 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 and site procedures will require revision to address the use of a supplemental cooling system prior to loading any fuel assemblies classified as high burnup fuel.

4.3.10 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 OP 2223 "MPC Fuel Loading Preparations and Sealing Operation." Continuous sampling for combustible gas buildup is performed during the MPC lid-to-shell weld, including NDE. Continuous sampling is also maintained during any repairs to the weld, if required. If completion of the weld of the MPC is interrupted for any reason, combustible gas concentration is verified less than or equal to 2 % (50% of lower explosive limit) 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 OP 2225, "MPC Unloading Operation." 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 for flammable gases during unloading are first, minimization of gases by venting and purging, and second, the exclusion of ignition sources.

5 - COMPLIANCE WITH HI-STORM 100 CASK SYSTEM SER AND FINAL SAFETY ANALYSIS REPORT

10 CFR 72.212(b)(6): 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)(5) of this section.

The following documents VY'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 are identified. All described deviations have been evaluated under the VY 10 CFR 72.48 process, as applicable. Changes made by Holtec generically or site-specifically for VY are discussed in Section 6.

HI-STORM FSAR Revision 4 was used as the baseline for the table in Section 5.1.

5.1 SER and FSAR Chapter 1, General Description

SECTION	REQUIREMENT	CHANGE DISCUSSION
SER FOR ORIGINAL CoC		
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 VY, nitrogen is used for a pressure test in lieu of a hydrotest.
SER FOR CoC AMENDMENT 1		
NA		No deviations or discussion required.
SER FOR CoC AMENDMENT 2		
6.3.2	Licensee must perform tests on Metamic with B_4C concentration above 15% prior to use.	VY will not be performing tests on Metamic. The CoC holder, Holtec, is required to perform these tests.

FSAR CHAPTER 1		
NA		No deviations or discussion required.

5.2 FSAR Chapter 2, Principle Design Criteria

SECTION	REQUIREMENT	DISCUSSION
Section 2.2.1.4	Confirmation of soil annual average soil temperature is to be performed by the licensee.	VY does not perform regular soil sampling. The Holtec FSAR (CFSAR) assumes a bounding average soil temperature of 77 degrees F. The design concern here is that in warm climates there could be an adverse effect on the thermal analysis. For cold climates this concern is not applicable and VY is bounded by the CFSAR analysis.

5.3 Chapter 3, Structural Evaluation

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.4 FSAR Chapter 4, Thermal Evaluation

SECTION	REQUIREMENT	DISCUSSION
Section 4.5.6	During normal handling and onsite transfer operations the water jacket shielding water is contained in the water jacket.	This will normally be the case with the exception of when the cask is placed into and lifted out of the pool. This is acceptable because the cask will be monitored based on an established time to boil calculation and if additional cooling is necessary equipment will be pre-staged to accomplish cooling. In addition during this time neutron shielding will be provided by the water in the MPC. It should be noted that this is due to HI-TRAC trunnion loading limitations and has been done at other stations.

5.5 FSAR Chapter 5, Shielding Evaluation

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.6 FSAR Chapter 6, Criticality Evaluation

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.7 FSAR Chapter 7, Confinement

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.8 FSAR Chapter 8, Operating Procedures

SECTION	REQUIREMENT	DISCUSSION
Section 8.0	User-developed procedures and the design and operation of any alternate equipment must be reviewed by the Certificate Holder prior to implementation	HOLTEC review is not considered warranted and will be noted as an exception in the 72.212 report. VY used procedures that were used at another Entergy plant, had independent reviews by personnel that are very knowledgeable in dry fuel storage and completed compliance reviews against 10CFR72, Holtec CoC and Holtec FSAR requirements.
Section 8.1	The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize.	Air Temperature rise test not required per HOLTEC. See report HI-2043249 Rev.0 for results that justify not performing the test.
Table 8.1.5	HI-STORM 100 System Torque Requirements	Table 8.1.5 of the HOLTEC FSAR is revised to replace the torque requirements for the HI-TRAC Pool Lid Bolts, The HI-TRAC Transfer Lid Bolts, the MPC Lift Cleat Stud Nuts and the HI-STORM Lid Nuts to hand tight.

Section 8.1	The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize.	Air Temperature rise test not required per HOLTEC. See report HI-2043249 Rev.0 for results that justify not performing the test.

5.9 FSAR Chapter 9, Acceptance Criteria and Maintenance Program

SECTION	REQUIREMENT	DISCUSSION
Section 9.1.1	Fabrication and Nondestructive Examination (NDE)	References 21 and 22 document disposition of the Non-Helium leak tested MPC weld issue.
Section 9.1.1 Item 4	Inspection plan shall be reviewed and approved by Holtec in accordance with its QA program.	Holtec review and approval of the inspection plan for welds performed at VY does not provide significant increased assurance of compliance with the CoC, safety in loading or storing the cask, or the intent of the FSAR. VY had independent reviews by personnel that are very knowledgeable in Inspection requirements and completed compliance reviews against 10CFR72, Holtec CoC and Holtec FSAR requirements. Therefore, additional review is not considered necessary.
9.2.1	Perform Load Test of transfer cask trunnions annually	HI-STORM FSAR, Section 9.1.2 describes initial load testing requirements and requires compliance with ANSI N 14.6 and NUREG 0612. ANSI N 14.6, Section 6.3.1, requires that annual load testing be performed or in cases where surface cleanliness and conditions permit, load testing may be omitted and dimensional checks and nondestructive examination in accordance with Section 6.5 is sufficient. Substitution of NDE for load testing is consistent with the referenced standard and industry practice. Inspections and NDE of the trunnions are performed in accordance with written procedures.
Table 9.2.1	Monthly overpack vent screen visual examination for damage, holes, etc.	The CFSAR section 9.2.6 recognizes that temperature monitoring is an acceptable alternative to monthly vent screen monitoring. Therefore, VY performs temperature monitoring and an annual inspection of vent screens.

Section 9.2.4	The pressure relief valves used on the water jacket for the HI-TRAC transfer cask shall be calibrated on an annual basis (or prior to the next use if the period of out of use exceeds one year).	The pressure relief valves used on the water jacket are either recalibrated or replaced before each dry fuel storage campaign. They are purchased with the appropriate qualifications as required by the VY QAPM.

5.10 FSAR Chapter 10, Radiation Protection

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.11 FSAR Chapter 11, Accident Analysis

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.12 FSAR Chapter 12, Operating Controls and Limits

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

5.13 FSAR Chapter 13, Quality Assurance

SECTION	REQUIREMENT	DISCUSSION
NA		No deviations or discussion required.

6 - 72.48 REVIEWS AND OUTSTANDING CASK LICENSING BASIS DOCUMENT CHANGES

The primary licensing documents of record used in this 72.212 evaluation report are HI-STORM CoC Amendment 2, issued in June 7, 2005 and Revision 4 of the HI-STORM 100 System FSAR, issued in April, 2006. Since the time FSAR Revision 4 was issued, a number of changes have been made to the FSAR text, tables, or drawings that could affect the VY hardware and procedures.

Generic, permanent modifications to the Holtec hardware are documented by Holtec using the Engineering Change Order (ECO) process. That process requires a review of the change pursuant to 10 CFR 72.48 if the change affects the cask design or operation as described in the FSAR. Those permanent changes that affect the licensing drawings are incorporated on the drawings when the ECOs are approved. Updated licensing drawings and any changes to the FSAR text, tables, and figures authorized by an ECO are included in the periodic revisions to the HI-STORM FSAR. See Section 6.1 for a listing of ECO changes to hardware approved by Holtec.

Holtec changes to the FSAR Revision 4 text, tables and figures and their impact on VY, if any, are summarized in Section 6.2.

Holtec one-time fabrication deviations are documented in the Supplier Manufacturing Deviation Report (SMDR) process. Dispositions of SMDRs that are "accept-as-is" or "repair" are reviewed under 10 CFR 72.48 if the disposition constitutes a change that affects the cask design or operation as described in the FSAR. See Section 6.3.

A record of the applicable ECOs, SMDRs, and 72.48s can be found in the Component Completion Record (CCR) for each serial number component (i.e., MPC, overpack, and transfer cask). Permanent design changes to MPCs and overpacks authorized by ECOs and 72.48 that constitute changes from FSAR Revision 4 (per the CCRs) are summarized below by component. Serial number applicability is shown in Section 2.C of this report.

6.1 HOLTEC DESIGN CHANGES AND FABRICATION DEVIATIONS

6.1.1 MPCs

<u>ECO 1021-76R1</u>

This ECO makes 17 changes to the fabrication drawing for the MPC-68 canister. These changes all involve details of the internal MPC construction and upper fuel spacers. The MPCs used for VY fuel do not require upper fuel spacers. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

<u>ECO 1021-77R1</u>

This ECO modifies the MPC enclosure vessel licensing drawing to permit an optional, smaller diameter lid. This is compensated for by increasing the thickness of the upper MPC shell on these MPCs. This change addresses a manufacturing mistake by Holtec in which they manufactured lids with dimensions for the MPC-68F canister rather than the MPC-68. Rather than scrap the lids, they were authorized for use on a one-time basis for two MPCs not used at VY (Reference SMDRs 1021-1269 and 1021-1364). Therefore, there is no impact on VY.

ECO 1021-78

This ECO modifies the MPC-68 basket licensing drawing to permit either Boral or Metamic as a neutron absorber, permit certain manufacturing imperfections in the neutron absorber, and permit the neutron absorber in a given fuel basket panel to be made from one or two pieces. The use of Metamic was approved by the NRC in Amendment 2 to the CoC. The other changes involve details of the MPC fuel basket design and fabrication. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-80

This ECO modifies the MPC-68 basket licensing drawing to a) specify a minimum length for the neutron absorber and b) replace toleranced dimensions with nominal dimensions on the shim assemblies. These changes involve details of the MPC fuel basket design and fabrication. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-83R1

This ECO makes five editorial changes to the MPC enclosure vessel licensing drawing. These changes involve details of the MPC fuel basket design and fabrication. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-89

This ECO makes 12 changes to the MPC-68 fabrication drawing and one editorial change to the MPC-68 fuel basket licensing drawing. Changes 1, 2, 3, and 5 modify materials and dimensions, and add a new part as corrective actions to prevent the seal washer bolt under the vent and drain caps from falling out during MPC preparation activities. Other changes are either editorial corrections or fabrication details. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-95

This ECO makes a total of 13 changes to the MPC-68 shell fabrication drawing and the generic MPC shell licensing drawing, including changes to the PWR and BWR fuel spacer design. VY fuel does not require fuel spacers and the remaining changes are minor details that do not affect how the canister is loaded or otherwise used to store fuel. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-96R1

This ECO makes editorial changes to the MPC enclosure vessel licensing drawing to remove Indian Point Unit 1 information and has no effect on VY MPCs.

ECO 1021-97

This ECO makes two minor changes to the MPC-68 basket licensing drawing. The changes are minor details that do not affect how the canister is loaded or otherwise used to store fuel. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-99

This ECO changes the licensing drawing to allow use of an MPC bottom piece made from either stainless steel or carbon steel. All carbon steel surfaces that may otherwise come in contact with spent fuel pool water shall be coated or covered with stainless steel. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1021-100

This ECO permits an optional fuel spacer design for MPC-68. VY fuel is of a length that no fuel spacers are required. Therefore, this ECO has no effect on VY MPCs.

ECO 1021-102

This ECO affects only two specific MPC canisters (Serial # 172 and # 173) neither of which belong to VY. Therefore this ECO is not applicable to VY.

ECO 1021-103

This ECO deletes the option of fabricating neutron absorber panels using two panels. No MPCs have been manufactured with this option. Therefore this ECO is not applicable to VY.

ECO 1021-104

This ECO provides for minimum and maximum thickness of fuel spacer plates. VY does not utilize fuel spacers. Therefore this ECO is not applicable to VY.

ECO-1021-106

This ECO revises the MPC shell licensing drawing to add a Note 5 requiring the minimum yield strength of the MPC lid to be 33 ksi. This is a fabrication requirement. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO-1021-107

This ECO revises the MPC shell licensing drawing to change the contour finish of the non-structural weld surface between the MPC split lids from "concave" to "flush." This is a fabrication requirement. There is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO 1021-109

This ECO revises the MPC shell licensing drawing to remove the "all around" designation for the port cover plate and closure ring pug welds and to restore an inadvertently remove fuel spacer data table. While the plug welds are performed in the field, there is no change to the field welding practice. The drawing change matches field practice. There is no impact on VY site implementation documents or the evaluations summarized in this report.

6.1.2 Overpacks

ECO 1024-119

This ECO makes minor dimensional and bill-of-material (BOM) changes to the HI-STORM 100S Version B fabrication drawing and to increase the dimension of the outlet air duct opening by 1/16 inch. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-121

This ECO makes several minor dimensional and BOM changes to the HI-STORM 100S Version B fabrication drawing and adds a second lid closure bolt optional design to both the fabrication and licensing drawing. Any impacts on DFS lid installation procedures have been incorporated.

ECO 1024-123R1

This ECO makes several changes to the overpack licensing drawing to replace the specific grade of carbon steel (e.g., SA 516 Grade 70 or SA36) to a generic call out of "carbon steel." As a result of the material changes, the properties of the materials, allowable stresses, and the factors of safety from the results of the structural analyses changed. This required concomitant changes to tables and text pertaining to the structural analyses presented in the FSAR. These changes pertain only to the licensing basis structural analyses. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-126

This ECO changes the overpack lid bolt hole dimensions and makes an editorial change to the overpack fabrication drawing. The overpack licensing drawing is revised to make an editorial change, revise the lid closure bolt location, and permit drilling and tapping holes for temperature monitoring instrumentation. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-131R1

This ECO provides another optional overpack lid bolt configuration, allows either carbon steel or stainless steel studs for attaching the lid vent screens, and makes a number of fabrication detail changes to the fabrication drawing. This ECO also increases the lid lift block length dimension from 17.5 inches to 18.5 inches and makes three editorial changes to the licensing drawing. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-134

This ECO makes two editorial changes to the overpack licensing drawing. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-135R1

This ECO makes four editorial changes to weld callouts on the overpack licensing drawing. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-139R1

This ECO makes a total of 18 changes to the overpack licensing and fabrication drawings, including a change to the lid bolt configuration and permission to use stainless steel studs to attach the air duct screen to the overpack body. The lid bolt and stud material changes have been addressed in DFS procedures and spare parts. Other changes have no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-141

This ECO makes nine minor changes to dimensions on the overpack licensing drawing. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report.

ECO 1024-142

This ECO makes eight minor changes to dimensions on the overpack licensing drawing. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report

ECO 1024-151R1

This ECO adds several seal and tack welds to the overpack steel structure to assist with fabrication, makes editorial changes, allows fabricator relocation of stitch welds, allows the lid shear ring to be fabricated from multiple pieces, and allows an option to drill threaded holes into the overpack shell to use threaded fasteners rather than studs and nuts to secure the vent screens. These are fabrication details that have no impact on VY the evaluations summarized in this report.

6.1.3 Transfer Cask

ECO 1026-41

This ECO makes minor changes to the HI-TRAC lid bolt holes to address a fabrication issue. There are no interface points with plant equipment affected and no impact on DCS loading operations, fuel selection, ISFSI operating procedures, or the 212 Report

6.2 HOLTEC FSAR CHANGES

Holtec-initiated changes to the HI-STORM FSAR are documented using the ECO process and reviewed under 10 CFR 72.48, as appropriate. ECOs and 72.48s incorporated into FSAR Revisions 5 through 12 (as listed in Section 1.0 of those FSAR revisions) and those against Revision 12 approved as of September 30, 2014 are summarized below.

ECO 5014-124R1

This ECO originally removed the fabrication shop helium leakage test for the MPC from the HI-STORM FSAR. Revision 1 reinstated it for MPCs loaded with decay heat exceeding 20 kW. The original change resulted in an NRC violation, which is discussed in further detail in Section 6.5 of this report. The fabrication shop helium leakage test for all MPCs, irrespective of heat load, was later restored to the FSAR via ECO 5014-174.

ECO 5014-127

This ECO revises HI-STORM FSAR Chapter 3 to add impact testing requirements for components of the anchored overpack design (100A and 100SA). VY does not use the anchored overpack design. Thus, this ECO is not applicable to the VY ISFSI.

ECO 5014-128

This ECO makes an editorial change to correct the overpack steel temperature limit in Table 2.0.2 to match other tables in the FSAR. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-131

This ECO modifies the torque requirements for several connections from specific values to either hand-tight or wrench-tight. These changes have been incorporated into the appropriate VY DFS procedures.

ECO 5014-132

This ECO adds discussion an equivalent blast pressure wave of 60 psi for 85 milli-seconds to FSAR Table 2.2.1 and Section 3.4.7.2. This information relates to the structural response of an overpack to a pressure pulse. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-133

This ECO updates the structural analysis results in the FSAR to reflect changes made to the HI-TRAC transfer cask designs in a prior ECO. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-135

This ECO makes changes to structural information to reflect alternate carbon steel grades permitted for use in fabrication. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-136

This ECO makes revisions to the FSAR in support of CoC Amendment 3. VY continues to load casks in accordance with CoC Amendment 2. Thus, this ECO does not apply to VY.

ECO 5014-137

This ECO adds information to the HI-TRAC lifting trunnion inspection checklist. These changes have been incorporated as appropriate into VY inspection documents.

ECO 5104-138

The ECO changes the safety classification of the overpack concrete from ITC-C to ITS-B. This change reverses a previous change that downgraded the classification of the concrete but was never implemented by Holtec. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-139

This ECO makes changes to the HI-STORM 100U design. VY does not use the HI-STORM 100U overpack. Thus, this ECO does not apply to VY.

ECO 5014-144

This ECO makes editorial changes to the HI-STORM FSAR, including changing UST&D to HMD. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO 5014-147

This ECO removes the requirement to verify the HI-TRAC identification markings annually. This change has been incorporated into site DFS procedures, as appropriate.

ECO 5014-148

This ECO makes changes to the FSAR to add an alternate mesh size for the damaged fuel container design of 40x40 in addition to the previously value of 250x250. This is a design change that does not affect use of the DFC in the field. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-149</u>

This ECO makes changes to the concrete specifications for the overpack. Holtec is responsible for implementing this change. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5104-151</u>

This ECO makes editorial corrections to fix a typo and to make the FSAR consistent with the CoC technical specifications. VY's procedures already reflect the correct information in the CoC/TS. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-152</u>

This ECO removes the fuel rod buckling analysis from the FSAR. This is a licensing basis analysis that has no impact on use of the cask in the field. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-153</u>

This ECO makes changes in several places in the FSAR to remove references to lifting a loaded damaged fuel container. VY has not loaded any damaged fuel to date. If damaged fuel is loaded in the future, these changes will be considered.

<u>ECO 5104-154</u>

This ECO adds an alternative material for the HI-TRAC 125 lifting trunnions. VY uses the HI-TRAC 100 transfer cask. Thus, this ECO does not apply to VY.

<u>ECO 5014- 155</u>

This ECO adds a table to the FSAR that lists the amendments under which HI-TRAC transfer casks certified to a previous amendment may be used. This table will be used in the future if VY upgrades to a later CoC amendment for loading casks.

<u>ECO 5014-156</u>

This ECO modifies the testing requirements for overpack concrete. Holtec is responsible for implementing this change. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-159</u>

This ECO makes minor FSAR corrections for consistency between chapters and adds clarified requirements for purging simultaneous with exhausting when monitoring for combustible gases. VY has reviewed and modified its procedures as needed for combustible gas monitoring.

<u>ECO 5014-161</u>

This ECO makes FSAR changes to reflect the option of a two-part MPC lid. The two-part lid is not authorized under CoC Amendment 2, which VY uses. Thus, this ECO does not apply to VY.

<u>ECO 5014-162</u>

ECO 5104-162 adds a definition of "MPC transfer" to FSAR Table 1.0.1 as a confirming change to CoC Amendment 5 and adds MPC transfer as a function of the Vertical Cask Transporter (VCT) in FSAR Table 8.1.6. Neither of these changes has an impact on VY site implementation because the definition of MPC transfer was already included in the FSAR elsewhere and previously addressed in procedures, as needed. VY does not use the VCT for loaded MPC transfer from the HI-TRAC to the storage module.

<u>ECO 5014-164</u>

ECO 5014-164 makes an editorial change to MPC Enclosure Vessel Licensing Drawing 3923 to add missing ECO numbers to the drawing revision log. There is no impact on site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-166R1</u>

ECO 5014-166 modifies the HI-STORM FSAR to permit the use of as-rolled SA 516 Grade 70 carbon steel in fabricating the overpacks in addition to normalized steel of the same grade. This change affects material procurement and fabrication under the control of the CoC holder and does not affect site implementation documents or the evaluations summarized in this report.

<u>ECO 5014-167</u>

This ECO makes corrections to Chapter 4 of the HI-STORM FSAR, "Thermal Evaluation" to make it align with changes made in CoC Amendments 3 and 5. VY loads casks in accordance with CoC Amendment 2. Thus this ECO does not apply to VY.

<u>ECO 5014-169</u>

This ECO changes the word "crawler" to "Vertical Cask Transporter" in several locations in the FSAR. This is an editorial change and has no impact on the RBS site implementation or the evaluations summarized in this report.

<u>ECO 5014-170R1</u>

This ECO (along with ECO 1025-58) supports a design change that permits lead plates to be used in the HI-TRAC 125D transfer cask in lieu of using molten lead. VY uses the HI-TRAC 100D design, which was manufactured before this ECO was issued. Therefore, this ECO does not apply to VY.

<u>ECO 5014-171</u>

This ECO added the details of how to perform the thermal air flow test required by CoC Condition 9 to FSAR Chapter 8. This change applies to a condition of the CoC that was modified in an amendment later than Amendment 2. Therefore, this part of the ECO does not apply to VY. A second change to the FSAR made in this ECO provides a clarification to the requirements for the MPC pressure test. The language requiring no drop in pressure is changed to permit some drop in pressure provided the pressure does not fall below the minimum test pressure. This flexibility has been added to the hydrostatic test procedure.

ECO 5014-172

This ECO adds text to HI-STORM FSAR Section 3.5 addressing fuel rod integrity. This information is strictly for licensing purposes. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO 5014-173

This ECO modifies the operating procedures so that in the event the time-to-boil is exceeded that forced water circulation through the MPC is initiated. Previously, water flushing was suggested as one option. This revised requirement has been incorporated into VY DFS procedures.

ECO 5014-174

This ECO restored to the FSAR the fabrication shop helium leak test of the MPC that was removed by ECO 5014-124. The disposition of MPCs that were delivered to RBS without having been tested in the fabrication shop is discussed in Section 6.5 of this report. The restored FSAR leak testing requirement is implemented by Holtec at the fabrication shop. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO 5014-175

This ECO clarifies FSAR Section 3.1.2.1.1.4 that explosion hazards are required to be evaluated on a site-specific basis per the HI-STORM CoC using the overpressure acceptance criteria in FSAR Table 2.2.1. The overpressure acceptance criteria in FSAR Table 2.2.1 have not been changed by this ECO. Therefore, the explosion hazards analysis referred to in Section 4.3.6.5 of this report remains valid.

ECO-5014-177

This ECO addresses design modifications to the HI-STORM 100U underground storage system vertical ventilated module, which is not used at VY. Therefore, this ECO does not apply to VY.

ECO-5014-179

This ECO revises the HI-STORM FSAR to add constraints to use of the Supplemental Cooling System (SCS) that were made necessary by the increase in allowed maximum canister heat load approved in CoC Amendment 5. VY is loading casks in accordance with CoC Amendment 2 and is not currently loading fuel into the MPCs that requires using the SCS. Thus no action is required.

ECO-5014-180

This ECO adds more text to HI-STORM FSAR Section 3.5 addressing fuel rod integrity. This information is strictly for licensing purposes. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-182

ECO 5014-182 added a new set of material yield strength requirements for the MPC lid and updated results for lid lifting structural analyses. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO-5014-184

This ECO corrects the helium backfill range for MPC-32 to match the value in the CoC. This is an editorial change to an MPC model not used at VY and has no impact on VY site implementation or the evaluations summarized in this report.

ECO-5014-188R1

This ECO makes the following text changes to the HI-STORM FSAR: 1) Clarifies requirements for water conditions in the MPC-to-HI-TRAC annulus during vacuum drying as a result of HIB-45 and HIB-48, 2) Clarifies that draining the MPC may be performed with either nitrogen or helium, 3) Adds specific heat load values rather than references to other locations for the information, 4) Clarifies that pre-cooling the MPC contents before re-flooding for unloading is optional, not required, 5) Deletes requirements for MPC drying and replaces with a reference to the CoC/TS, and 6) Clarifies that use of the Supplemental Cooling System is required for more than just MPCs containing high burnup fuel assemblies. The changes pertaining to vacuum drying are applicable to VY. Other changes are clarifications to improve understanding or corrections to make the FSAR match requirements already in the CoC (some after Amendment 2). VY DFS procedures have been reviewed and appropriate changes have been made as a result of this ECO.

ECO-5014-189

This ECO clarifies range requirements for the HI-TRAC transfer cask water jacket hydrotest pressure gauge and requires that the total thickness of the HI-TRAC lead sheets meets the licensing drawing dimension, including tolerances. These are changes implemented by Holtec during the fabrication and testing of the HI-TRAC transfer cask. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-191

This ECO adds clarifying information for coating materials used on the HI-TRAC and HI-STORM. These are changes implemented by Holtec during the fabrication the HI-TRAC transfer cask and HI-STORM overpack. There is no impact on site implementation documents or the evaluations summarized in this report.

ECO-5014-192

This ECO adds information to the HI-STORM FSAR to 1) clarify that full vacuum is the most limiting thermal condition during vacuum drying, 2) clarify that nitrogen or helium may be used for canister blowdown operations but only helium is permitted for storage operations, and 3) require vacuum drying procedures to include temperature cycling limits for MPCs that are subject to a time limitation for drying. VY DFS procedures have been developed to reflect these requirements.

ECO-5014-193

This ECO modifies text in HI-STORM FSAR Section 3.5 addressing fuel rod integrity to address an NRC violation. The changes clarify that because no credit is taken for the fuel cladding to confine radioactive material inside the MPC in an accident condition, no analysis demonstrating this design function is required. This information is strictly for licensing purposes. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-194

This ECO restores information to the HI-STORM FSAR pertaining to HI-TRAC water jacket testing that was previously removed inadvertently. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-195

This ECO makes several changes to the HI-STORM 100U underground storage system design drawing. VY does not use the HI-STORM 100U System. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-196R1

This ECO adds text to the FSAR to provide guidance to the users on how to calculate the total MPC heat load consistent with the assumptions used in the thermal analysis for the system. These changes are made as a result of HIB-51, Revision 2, which identified that users were calculating MPC heat loads incorrectly for determining vacuum drying and helium backfill requirements in the CoC/TS. HIB 51 pertains to Amendment 5 and later. Because VY continues to load casks under CoC amendment 2, this ECO does not apply to VY.

ECO-5014-197

This ECO modifies the FSAR instructions pertaining to MPC cooldown for unloading MPCs that were loaded under CoC Amendment 3 or later. VY currently loads MPCs under CoC Amendment 2.

ECO-5014-198R1

This ECO makes several changes that align the information in the TS bases in FSAR Appendix 12.B with the TS in the CoC. The VY DFS procedures were developed to reflect the TS in Amendment 2. Therefore, these FSAR changes have no effect on VY DFS procedures or the evaluations in this report. One additional FSAR change made in this ECO clarifies the implementation of the Surveillance

Requirement in LCO 3.1.3 for MPC re-flooding. The VY MPC unloading procedure has been developed to reflect this FSAR (TS Bases) change.

ECO-5014-199

This ECO modifies the HI-STORM FSAR to reflect the permission granted by ANSI N14.6 to perform periodic NDE on lifting devices in lieu of additional load testing that was previously the only option permitted by the FSAR. VY lift device maintenance procedures will be developed accordingly.

ECO-5014-200

This ECO clarifies the HI-STORM FSAR to reflect the option to manufacture all HI-TRAC transfer cask models with lead sheets in lieu of poured lead. These are changes implemented by Holtec during the fabrication and testing of the HI-TRAC transfer cask. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-201

This ECO makes two editorial changes to the HI-STORM FSAR and adds the overpack base bottom plate to the list of steel components for which the material exempt from impact testing. The technical change only affects material procurement and not use of the overpack. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-202

This ECO adds text to the HI-STORM FSAR to 1) permit users to pre-cool the helium in the MPC prior to re-flooding, at their discretion and 2) to clarify that operational activities are intended to be carried out in a continuous fashion in a reasonable time frame. If VY plans to pre-cool the helium in the MPC, appropriate procedural instructions will be put in place consistent with the cask FSAR and this ECO. VY implements operational activities in a continuous fashion and reasonable time frame.

ECO-5014-203

This ECO clarifies the shielding model for BPRAs and thimble plugs in response to HIB-57. This change does not affect fuel selection or the operation of the cask in the field, and VY is a BWR. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-204

This ECO revises the HI-STORM FSAR to 1) clarify that the licensing basis fire analysis is only a single event postulated for licensing purposes and may not bound all site-specific fire threats. Users must evaluate site-specific fires on an individual basis, and 2) replace the overpack steel structure temperature acceptance criterion for the fire accident event with a requirement that the structure remain physically stable (50% of the component's melting temperature).

VY performed a fire and explosion hazards analysis for the initial DFS campaign in 2008. That analysis remains applicable to the VY ISFSI because VY continues to load dry storage casks in accordance with the same CoC amendment and FSAR revision in effect at the time the initial casks were loaded.

ECO-5014-205

This ECO clarifies the TS Bases for LCO 3.1.2, "SFSC Heat Removal System," in the HI-STORM FSAR to state that the heat removal system remains operable with up to 50 percent blockage of the air inlet and outlet air ducts together. This is consistent with the note modifying the LCO in the CoC/TS in Amendment 5 and is an editorial correction. VY loads casks in accordance with Amendment 2. Therefore, this ECO is not applicable to VY.

ECO-5014-206

This ECO adds new items to FSAR Table 2.2.6 and clarified operating instructions in Chapter 8 to reflect the new, optional vent and drain port cap design. If VY uses the new vent and drain port designs, appropriate procedures changes will be made. The new design does not otherwise affect how the canister is loaded or used to store fuel.

ECO 5014-207

This ECO makes changes to the FSAR text pertaining to the CTF design. VY does not use a CTF. Thus, no further action is required.

ECO-5014-208

This ECO makes changes to the FSAR text pertaining to the CTF design. VY does not use a CTF. Thus, no further action is required.

ECO-5014-209

This ECO makes several sets of changes to the FSAR: 1) Changes 1, 2, 11, 12, and 13 pertain to fabrication leak testing of the MPC, which is performed by Holtec during manufacturing. 2) Changes 5, 6, 10, 14, and 15 pertain to the MPC-68M model that is not authorized for use under CoC Amendment 2. 3) Changes 3 and 9 pertain to a PWR fuel type. VY is a BWR. 4) Changes 4, 7, and 16 are editorial. 5) Change 8 makes changes to the description of the criticality analysis. None of these changes impact VY site implementation documents or the evaluations summarized in this report.

ECO-5014-210

This ECO makes several editorial and substantive changes pertaining to the materials used in the cask design and credited in the structural analyses. This information is strictly for licensing purposes. Therefore, there is no impact on VY site implementation documents or the evaluations summarized in this report.

ECO-5014-211R2

This ECO makes editorial corrections to FSAR Table 2.1.19 and 2.1.24 and provides guidance in Chapter 5 for performing site-specific dose analyses. These changes have been considered in the VY dose analyses.

ECO-5014-212

This ECO adds text to the FSAR to require site-specific handling analyses if MPCs are to be partially loaded. VY does not plan to partially load any MPCs, but will consider this information if partial loading becomes necessary of desired.

ECO-5014-213

This ECO adds information to the FSAR pertaining to malfunctions of the FHD System and Supplemental Cooling System, now considered off-normal events in the licensing basis. It states that no action is required for a malfunction of the FHD System other that detecting it, because the MPC would revert to natural convection cooling in this case. VY does not currently use FHD for drying nor does VY currently load MPCs that require use of the SCS. Thus, no further action is required.

ECO-5014-216

This ECO makes three changes to the FSAR that pertain to the new MPC-68M model, which is not authorized for use under CoC Amendment 2, and, therefore, does not apply to VY. The fourth change adds a clarification to the Operations chapter that if the FHD System malfunctions and the MPC pressure is below the minimum required by the technical specifications, the MPC must be backfilled to at least the minimum TS-required backfill pressure. VY does not use FHD for drying. Thus, no further action is required.

5014-217

This ECO adds text to the FSAR to provide clarification on when to initiate the Supplemental Cooling System. VY does not currently load MPCs that require use of the SCS. Thus, no further action is required.

5014-219

This ECO makes a small number of editorial changes to clarify the FSAR. There is no impact on VY site implementation documents or the evaluations summarized in this report.

6.3 SITE-SPECIFIC HARDWARE CHANGES

Site-specific changes to the VY hardware (e.g., those fabrication deviations dispositioned using the SMDR process as “accept-as-is” or “repair”) are documented in the CCRs and documentation package for each serial number component. SMDRS 1548, 1602R1, 1705R1, and 1705R2 were issued against the MPCs being loaded in the VY loading campaigns. SMDRs 1573, 1574, 1577, 1595, 1831, and 2179 were issued for the VY overpacks. SMDR 1564R1 was issued during the fabrication of the VY HI-TRAC transfer cask (S/N 1026-6). See Table 2.C for the specific serial number components to which these SMDRs apply. The SMDRS are summarized below.

6.3.1 MPCs

SMDR 1548

This SMDR accepts minor damage to the neutron absorber within the limits of the MPC-68 basket licensing drawing. There is no degradation of the design function of the neutron absorber.

SMDR 1602R1

This SMDR repairs minor damage to an MPC lid that was bumped by a forklift. The lid, including the weld pre area, was repaired to meet requirements.

SMDR 1705R1

The SMDR accepts six plates of raw stainless steel that exceeded the maximum hardness requirement in the procurement specification, based on initial sample testing. Additional testing of plate material (not near the edges of the production rolls) met the hardness requirements.

SMDR 1705R2

This revision to the SMDR simply states that the plates in question (dispositioned in Revision 1 to the SMDR) may be used with the PWR MPC-32 designs.

6.3.2 HI-STORM Overpacks

SMDR 1573

This SMDR reworks a damaged overpack lid lifting hole by filling the original four holes with plugs and drilling a new set of holes 90 degrees apart that provide the same function.

SMDR 1574

This SMDR addresses an overpack anchor block stud hole thread engagement problem by running a tap into the holes to clear any blockage, then ensuring sufficient corrosion protection is provided for the threads.

SMDR 1577

The SMDR addresses an overpack anchor block stud hole thread engagement problem by running a tap into the holes to clear any blockage, then ensuring sufficient corrosion protection is provided for the threads. NOTE: This SMDR states that it applies only to Quad Cities overpack S/N 182. It is not clear why this SMDR appears on VY's CCR for S/N 69.

SMDR 1595

This SMDR addresses an overpack anchor block stud hole thread engagement problem by running a tap into the holes to clear any blockage, then ensuring sufficient corrosion protection is provided for the threads.

SMDR 1831

This SMDR accepts several plates of steel that did not meet chemistry requirements for manganese and tensile test requirements. The plate is accepted based on an evaluation of the structural and shielding requirements of the parts these pieces of steel are used to make.

6.3.3 HI-TRAC Transfer Cask

SMDR 1564

This SMDR addresses indications found during MT exams of welds in the water jacket bottom plate. The indications were ground out and re-worked to restore compliance with the design requirements.

6.4 Evaluation of Holtec International Bulletins (HIBs)

6.4.1 MPC Closure Ring Weld Inspections - HIB 42

Vendor welding procedures perform a verification of the MPC closure ring to MPC shell fillet weld per the resolution to HIB No. 42.

6.4.2 MPC Heat Loads and Vacuum Drying - HIBs 45R2 and 48R1

Procedure OP 2223 was revised to ensure verification of proper heat generation rates per HIB 45R2 and HIB 48R1.

6.4.3 Mating Device Alignment on HI-STORM - HIB 46

Procedure OP 2224 was revised to ensure for proper alignment prior to placing the mating device onto the HI-STORM per HIB No.46.

6.4.4 MPC Heat Loads and Thermal Calculations - HIB 51

These HIBs address a lack of clarification as to how to determine individual cell heat loads that comply with the cask thermal analysis. This HIB applies to cask loaded under CoC Amendments 5, 6, and 7. Therefore, it does not apply to VY. Note that this issue was also addressed in ECO 5014-196R1.

6.4.5 Potential MPC Overpressurization – HIB 53

This HIB addresses a concern about potentially overpressurizing the MPC during processing when there is no vent path established from the MOPC that is filled with water. VY DFS procedures have been reviewed and revise as appropriate to address this issue.

6.4.6 Lifting Apparatus ANSI N14.6 Qualification – HIB 56

VY has evaluated this HIB as it applies to lifting devices used during cask handling and compliance with applicable commitments to ANSI N14.6 has been confirmed.

6.4.7 Inputs to Dose Analyses – HIB 61

This HIB provides guidance to users for the combinations of burnup and cooling time to be used in site-specific ISFSI dose analyses. This HIB has been reviewed and considered in VY's dose analysis.

6.4.8 Partial Loading of MPCs – HIB 62

This HIB provides guidance for users who must, or wish to partially load an MPC. The information in this HIB was incorporated into the HI-STORM FSAR under ECO 5014-212 and has been reviewed by VY. If and when an MPC is short-loaded, this guidance and the FSAR will be consulted.

6.4.9 MPC Siphoning – HIB 63

This HIB identifies a potential problem with electric pumps not having a discharge check valve siphoning water out of the MPC. The recommendations in the HIB have been implemented at VY.

6.4.10 HI-TRAC Water Jacket Filling – HIB 64

This HIB addresses the inadvertent filling of the HI-TRAC 125D water jacket with spent fuel pool water through quick disconnect fittings. The problem only applied to later-vintage HI-TRAC 125D transfer casks. VY uses the HI-TRAC 100D transfer cask. Thus, this HIB does not apply to VY.

6.4.11 Leaving Mating Device on HI-STORM – HIB 65

This HIB provides guidance for leaving the mating device installed on a HI-STORM overpack after the MPC has been downloaded. This guidance has been considered and incorporated into DFS procedures, as appropriate.

6.5 MPC Leakage Testing

In January, 2006, Holtec International implemented a 72.48 change to the HI-STORM FSAR to remove a fabrication shop helium leak test of the MPC. A number of MPCs were fabricated and delivered to customers untested, including some to VY. Five untested VY MPCs were loaded with fuel and moved to the ISFSI. In August, 2009, the NRC found the change to remove the leak test to be unacceptable and issued a violation to Holtec (EA-09-190). Untested MPCs that were delivered to VY but not loaded with fuel were successfully leak-tested on site before being loaded with fuel and deployed at the ISFSI. The untested MPCs that were loaded with fuel and moved to the ISFSI were found to be acceptable for continued operation as documented in Entergy's letter to the NRC (ENOC-10-00034 dated 11/2/10) and the NRC's response dated January 26, 2011 concurring with Entergy's position.

APPENDIX A – HAZARDS EVALUATION FOR THE VY ISFSI AND HAUL PATH

1.0 Introduction

This summary of the hazards analysis is provided to give a summary of the hazards that were addressed and the resulting administrative controls that are included in plant procedures.
The hazards evaluation demonstrated that the assumptions regarding site-specific hazards affecting the cask are not exceeded at the VY site, and that the VY ISFSI operations do not place the VY plant outside of its licensing basis. A description of site-related hazards to the cask and ISFSI operation-related hazards to the plant are provided, along with results of the analysis of those hazards.

2.0 Hazards Walkdown

In order to systematically develop the appropriate hazard information, a site walkdown of the VY ISFSI and haul path was performed from the Containment Access Building (CAB) to the VY ISFSI Facility Pad.

3.0 Hazards Evaluation

3.1 Falling Objects

During the Haul Path Hazards Walkdown, a number of objects were identified as having the potential of falling onto the ISFSI Facility Pad or onto the travel path of the cask. These objects are discussed in the sections that follow. The following were considered:

- Plant Stack
- North Warehouse
- Wooden Visual Barrier
- Security Towers
- Light Towers
- Liquid Nitrogen Tank No. TK-1001-1
- Containment Access Building
- Construction Office Building
- Light Tower No. 12

As a result of the review of these potential falling objects, it is concluded that the design basis and the accident analysis presented in the Holtec HI-STORM 100 FSAR remain the bounding case. Procedures have established administrative controls that suspend cask transport operations during high wind conditions to ensure cask protection from falling objects or flying debris.

3.2 Explosion/Fire Source

There were numerous items identified during the Haul Path Hazards Walkdown that had potential for explosion and/or fire. There were various transformers, fuel oil tanks, a new wooden visual barrier structure, and Radwaste Storage areas near the ISFSI Facility Pad and along the haul path.

In general, fire hazards can be compared to the design basis fire in Section 11.2.4.2.1 of the CFSAR, which assumes 50 gallons of burning transporter fuel engulfing the HI-STORM overpack. This analysis assumes a fire emissivity of 1, and a surface emissivity of 0.9, a flame temperature of 1475°F, an ambient air temperature of 100°F, and an air temperature into the HI-STORM 100 ventilation path of 300°F. The HI-STORM 100 safety analysis does not, however, evaluate exposure of the HI-STORM 100 overpack to nearby non-engulfing fires. Therefore, a modified version of the Holtec approach was applied to the VY HI-STORM 100s exposed to nearby fire hazards.

A non-engulfing fire causes the cask or overpack surface temperature to rise due to radiative heat transfer. There are two reasons for dismissing this type of fire hazard from further evaluation. The first is if there is no line of sight from the fire to the cask or ISFSI. Any intermediate object that serves as a shade will prevent radiative heat transfer from occurring. The second is if there is no credible ignition source. This approach will be applied to the haul path only. Specifically, it is not credible that buildings adjacent to the haul path will spontaneously ignite while the transport cask is in close proximity. Such ignition events are usually associated with unusual activities or vehicle collisions, and administrative controls on activities near the haul path can be credited with avoiding these types of events. Specific hazards evaluated include:

- North Warehouse
- Diesel Fuel Oil Tank
- North Radwaste Storage Area
- Wooden Visual Barrier
- Transformer No. T-11-1A
- Radwaste Storage In LSA/Box Containers
- Transformer No. T-12-1A
- Construction Office Building
- Fuel Oil Storage Tank (Tank No. TK-40-1A)
- Containment Access Building (CAB)
- Diesel Fire Pump Day Tank
- Transient and Parked Vehicles near the ISFSI, including fuel tankers
- Radwaste Compactor Area
- Bottles of compressed gases
- CO_2 tanks

3.3 Chemical Release Source

During the Haul Path Hazards Walkdown, the only chemical release sources that were identified included Carbon Dioxide Tank No. TK-115-1 and Liquid Nitrogen Tank No. TK-1001-1.

3.4 High Energy Line Break

There were no high energy lines identified during this walkdown. As a result, there was no potential for any damage to the cask or cask transporter from a high energy line break.

3.5 Flooding Sources

The primary flooding source is the Connecticut River. The VY UFSAR contains a flooding analysis that addresses the river. The design of the ISFSI Facility Pad resulted in a pad elevation that provides adequate cask storage system protection from any potential flooding impacts due to the Probable Maximum Flood (PMF). Plant Operations Procedures Number OP 2224, OP 2225 and OP 2229 have a precaution regarding transport during conditions that could result in the flooding of the Connecticut River. Specific hazards evaluated include:

- Liquid Radwaste Tanks
- Surge Tank No. TK-11A
- Condensate Storage Tank
- Hypochlorite Tanker Shipments

3.6 Pressurized Components/Rotating Missile Sources

There were two types of potential missile sources identified during this walkdown. The first type was rotating equipment such as pumps and motors. The second type was compressed gas cylinders. Specific hazards evaluated:

- Gas Cylinders
- Carbon Dioxide Tank No. TK-115-1
- Carbon Dioxide Tank No. TK-115-1 Pumps
- Liquid Nitrogen Tank No. TK-1001-1
- Liquid Nitrogen Tank No. TK-1001-1 Pumps
- A/C Unit Compressors and Fans
- Nitrogen Gas Cylinders

3.7 Transient Obstacles

There were no transient obstacles identified during this walkdown. As a result, there was no potential for any damage to the cask or cask transporter from transient obstacles. Transient vehicles are discussed in the Fire/Explosion section.

3.8 Overhead Power Line Clearance

There were no overhead power lines identified during this walkdown. As a result, there was no potential for any damage to the cask or cask transporter from overhead power lines.

3.9 Radiation Monitors

There were no radiation monitors identified during this walkdown. As a result, there was no potential for any damage to the cask or cask transporter from radiation monitors. However, their radiological readings may increase during cask storage and cask transport operations. Radiological monitoring needs to consider the likelihood of these increased readings. Appropriate responses and actions need to be taken as needed to address these occurrences. Consideration should be given to establishing administrative or procedural controls in recognition of this fact.

3.10 Environmental TLD

There were some environmental TLDs identified during this walkdown, as indicated on SVE Associates Thermo Luminescent Dosimeter Locations for Entergy VY. There is no potential for any damage to the cask or cask transporter from TLDs. Conversely, there are negligible effects on these TLDs as a result of the cask transport. TLD No. DR-45 is the closest to the haul path but during travel past its location, the radiation effects will be negligible. TLD No. DR-46 is over 500 feet away from the pad and therefore, the radiation effects of cask storage on this TLD is also negligible.

3.11 Existing Structures, Systems and Components

There were no other miscellaneous existing structures, systems or components identified during this walkdown. As a result, there was no potential for any damage to the cask or cask transporter from miscellaneous SSCs.

4.0 Required Administrative Controls

Based on the hazards analysis, administrative controls were established to:

- Suspend cask transport operations during high winds.
- Inspect the CAB pre-haul to assure that there is no possibility of the leakage of potentially explosive gases.
- Maintain a fire watch while the cask is inside the CAB.
- Suspend all activities potentially involving compressed flammable gas when the cask is within 27 feet.
- Ensure permanent storage of compressed flammable gas bottles will be at a distance of at least 27 feet from the nearest surface of the HI-Storms on the ISFSI Pad.
- Perform a pre-haul inspection to evaluate transient combustibles that may exist near the haul path that were not previously evaluated.
- Ensure that any forklift in the North Warehouse is stored more than 22.4 feet from the nearest surface of a HI-STORM on the ISFSI Pad.

- Ensure no HI-STORM is placed within 20 feet of the east wall of the North Warehouse, as measure from the wall outside surface to the North/South centerline of the overpack.
- Ensure established safe setback distances (identified in Ref 19) are observed and maintain a distance of at least 27 feet for trucks carrying compressed gas bottles and 10 feet for gasoline powered vehicles.
- Establish a fire watch at the compactor room area during cask transport operations.
- Suspend cask transport operations in the event the Radwaste storage in LSA boxes located east of the Radwaste building ignite and cause a fire.
- Establish a caution regarding transport during conditions that could cause a flood.
- Use EN-IS-109 during handling and storage of compressed gas cylinders located about 50 feet south of the ISFSI Pad.
- Use EN-IS-109 during handling and storage of nitrogen gas cylinders located west of the Reactor Building Railway entrance.
- Establish radiological monitoring controls that recognize the likelihood of increased readings during cask storage and transport operations.
- A pre-haul inspection will be made and any potential compressed flammable gas sources will be removed.
- The CAB sprinkler system shall be operable during HI-STORM transport to mitigate any potential fire within the structure.

5.0 Conclusions

The hazards identified during the ISFSI and haul path hazards walkdown do not exceed the design basis and accident analysis presented in the Holtec HI-STORM 100 FSAR, which remain the bounding case.

In some cases the hazards analysis identifies the need to establish administrative controls to ensure that hazards are controlled to ensure there is no impact on the ISFSI facility or load path. These administrative controls are included in procedures.

TAB 3

STATE OF VERMONT
PUBLIC SERVICE BOARD

Petition of Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. for a Certificate of Public Good, under 30 V.S.A. § 248, to Construct a Dry-Fuel-Storage Facility at Vermont Yankee Nuclear Power Station	)	Docket No. ________

MEMORANDUM OF UNDERSTANDING

This is a Memorandum of Understanding, made as of June 21, 2005 (hereafter the "MOU"), between Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (collectively hereafter the "Company"), and the Vermont Department of Public Service (the "DPS").

Preliminary Statement

The Vermont Yankee Nuclear Power Station (the "Station") is licensed by the Nuclear Regulation Commission ("NRC") to operate until March 21, 2012. The Station will have insufficient capacity to store spent-nuclear fuel ("SNF") in its existing spent-fuel pool, however, sometime prior to March 21, 2012.

Entergy VY proposes to construct a dry-fuel-storage ("DFS") facility at the Station to store SNF. Before it may do so, the Company must petition the Public Service Board (the "PSB") for a certificate of public good ("CPG") authorizing the construction and operation of a DFS facility under 30 V.S.A. § 248.

Chapter 157 of Title 10, Vermont Statutes Annotated, may prevent the PSB from considering the Company's DFS petition. In hearings on legislation proposing to authorize the Company to petition the PSB for a CPG authorizing a DFS facility under 30 V.S.A. § 248, the General Assembly asked questions and raised certain concerns about the proposed DFS facility.

To address these questions and concerns and facilitate the enactment of legislation by the General Assembly authorizing the Company's petition to the Board under 30 V.S.A. § 248, the Company and the Department hereby enter into this MOU and agree as follows:

1. **Line-of-Sight Barriers.** The Company will erect a wall to the extent required to provide line-of-sight protection on the north and east sides of the DFS pad. In addition, the Company will construct a protective structure if and to the extent required by the NRC, whether the result of a site-specific study or otherwise.

2. **Location of DFS Pad.** The Company will construct the DFS pad at a location set back at least 100 feet from the Connecticut River's 500-year floodplain, as depicted in the Flood Insurance Rate Study and on the Flood Insurance Rate Map, both dated September 27, 1991,

prepared by the Federal Emergency Management Agency for the Town of Vernon, Vermont (the "Floodplain"), in a location adjacent to the Company's existing facilities within the Station's Protected Area. If the Company determines that to comply with applicable NRC requirements it must, or the NRC directs the Company to, locate the DFS pad and any related facilities at a different location within the Owner Controlled Area ("OCA") of the Station, the Company will consider and use its best efforts to locate such facilities in a way that will minimize impacts on access to and use of lands within the OCA.

3. **Cask Spacing.** The Company will locate casks on the DFS pad in a manner such that access to individual casks will be maintained to the greatest extent possible. Under cask loading currently anticipated through the end of the Company's existing NRC license, this will result in individual access to each cask.

4. **Access Roads.** The Company will not construct roads providing access to the DFS pad closer than 100 feet from the Floodplain except that existing roads may be maintained (but not enlarged) within 75 feet of the Floodplain.

5. **Monitoring.** The Company will monitor the temperature on each cask located on the DFS pad continuously using an electronic-monitoring system contemplated by the Company's current DFS-system design. Monthly the Company will manually conduct radiation surveillance of each such cask. The DPS and the Company, in consultation with the Department of Health, will develop a protocol for reporting the results of such monitoring and surveillance to the DPS and the Department of Health.

6. **De-icing.** The Company will not use corrosive or flammable chemicals on or within fifteen feet of the DFS pad for purposes of de-icing.

7. **Out-of-State Waste.** The Company will not store waste generated outside of the state of Vermont at the Station.

8. **Off-Site Transfer.** The Company will use its commercial best efforts to ensure that high-level SNF stored at the Station is removed from the site in a reasonable manner and as quickly as possible to an interim or permanent location outside of Vermont.

9. **Pool Density.** The Company will configure the spent-fuel pool so that high-decay-heat assemblies of SNF are surrounded by low-decay-heat assemblies of SNF.

10. **Security.** The Company will be responsible for all costs of security for the Station required under its license from or otherwise by the NRC except that during decommissioning it will have the right to use funds from the Decommissioning Trust Fund to pay for such security.

11. **Clean Energy Development Fund.** The parties acknowledge that the General Assembly is concurrently with this MOU enacting legislation authorizing the Company to file a petition for approval to construct and operate a DFS facility under 30 V.S.A. § 248 based upon this MOU, including specifically the Company's agreement to fund a Clean Energy Development Fund established by the legislation (the "Fund"). The Company hereby agrees that if the Board issues to the Company a CPG authorizing such construction and operation, and if the Company obtains all approvals necessary to uprate the Station, the Company will pay to the

State of Vermont for deposit into the Fund payments calculated to total $15,625,000 during the period commencing January 1, 2006, and ending March 21, 2012. Payments will be made in equal quarterly amounts of $625,000 per quarter commencing as of January 1, 2006, with the initial payment to be due when the Company receives all such approvals and to include all quarterly payments due up to that time and with the last payment to be due on January 1, 2012.

12. **Preemption.** The Company agrees that it will not file an action or petition based on or otherwise seek, claim, defend or rely on the doctrine of federal preemption to prevent enforcement of its express obligations under this MOU.

13. **NRC Requirements.** Nothing in this MOU, including specifically Paragraph 12, shall be interpreted as prohibiting or restricting the Company from complying with any requirements or order of or any of its obligations under its license or otherwise to the NRC.

14. **Governing Law.** This MOU is governed by Vermont law and any disputes hereunder will be decided by the PSB.

15. **DPS Obligations.** The DPS will support the issuance of an order and findings by the PSB approving this MOU, subject to the DPS's obligations under Title 30 of the Vermont Statutes Annotated.

16. **Non-MOU Positions.** This MOU does not limit the DPS's ability to investigate issues or advocate positions that are relevant in the proceeding before the PSB in which the Company seeks a CPG for the DFS facility. Other than the specific conditions included in this MOU, the DPS will advocate for any further conditions on the Company and the physical attributes and operations of the DFS facility that the DPS believes are necessary to support a finding of public good pursuant to 30 V.S.A. § 248.

17. **Precedential Effect.** The parties agree that this MOU will not be construed by any party or tribunal as having precedential impact on any future proceeding involving the parties, except as necessary to implement this MOU or to enforce an order of the PSB resulting from this MOU.

ENTERGY NUCLEAR VERMONT YANKEE, LLC

By: Jay K. Thayer
Name: Jay K. Thayer
Title: Vice-President of Operations and duly authorized agent

ENTERGY NUCLEAR OPERATIONS, INC.

By: Jay K. Thayer
Name: Jay K. Thayer
Title: Vice-President of Operations and duly authorized agent

STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

By:____________________________________
Name: David O'Brien
Title: Commissioner of the Department of Public Service