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HI-STORE CIS Facility Occupational Dose Calculation

FOR

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DOCUMENT ISSUANCE AND REVISION STATUS[1]

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DOCUMENT CATEGORIZATION

In accordance with the Holtec Quality Assurance Manual and associated Holtec Quality Procedures (HQPs), this document is categorized as a:

- ☑ Calculation Package[3] (Per HQP 3.2)
- ☐ Technical Report (Per HQP 3.2) (Such as a Licensing Report)
- ☐ Design Criterion Document (Per HQP 3.4)
- ☐ Design Specification (Per HQP 3.4)
- ☐ Other (Specify):

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Notes

1. This document has been subjected to review, verification and approval process set forth in the Holtec Quality Assurance Procedures Manual. Password controlled signatures of Holtec personnel who participated in the preparation, review, and QA validation of this document are saved on the company"s network. The Validation Identifier Record (VIR) number is a random number that is generated by the computer after the specific revision of this document has undergone the required review and approval process, and the appropriate Holtec personnel have recorded their password-controlled electronic concurrence to the document.

2. A revision to this document will be ordered by the Project Manager and carried out if any of its contents including revisions to references is materially affected during evolution of this project. The determination as to the need for revision will be made by the Project Manager with input from others, as deemed necessary by him.

3. Revisions to this document may be made by adding supplements to the document and replacing the "Table of Contents", this page and the "Revision Log".

Summary of Revisions

Revision 0 – Original issue

Table of Contents

Safety Analysis Summary
(Source Document ID: HI-2177600R0)

Project title	HI-STORE CIS Facility
Report title	HI-STORE CIS Facility Occupational Dose Calculation
Name of the Site	HI-STORE CIS Facility
Name of the SSC (acronym for "system, structure or component")	HI-STORM UMAX Version C System HI-TRAC CS
Applicable NRC docket number	72-1040, 72-1051
Ref Holtec FSAR report number	HI-2167374, Revision 0
Ref. Purchasing Spec. ID (if applicable)	NA
Pertinent technical disciple	Shielding
[Proprietary Information withheld per 10CFR2.390]	

This Safety analysis summary is intended to provide the necessary information to demonstrate that the SSC identified above will render its intended safety function established within the purview of the above- referenced technical discipline under all applicable normal, off- normal and extreme environmental (accident) conditions. This safety summary is limited to demonstrating compliance in the specific area (technical discipline) of evaluation noted above and does not purport to cover other safety considerations that may apply to the subject SSC. The principal objective of this summary document is to provide a concise input to a safety significant Plant document such as a "design modification package" or a multi- disciplinary comprehensive safety evaluation (such as that required under 72.212) needed to implement a planned Plant initiative in accordance with the Plant's established safety confirmation protocol.

The summary information provided below is shared with the client and is archived as a part of the Holtec proprietary document (viz., a "Calculation package" which has imbedded Holtec intellectual property and is hence prohibited from external dissemination) in the Company's configuration control system. The main body of the parent document may make reference to this summary document, as appropriate, to prevent the need to repeat the same information. This document is QA validated along with its parent report and is subject to revision (and re-submittal to the client) if any significant change in the input parameters to the analysis so warrant. To serve its role as an authoritative input to the Plant, it is intended to be self-contained and entirely focused on safety. For this purpose, this summary document is organized in a series of sections to provide a succinct and concise safety assessment, as follows:

1. *Scope of Analysis*

This report provides a shielding evaluation for the HI-STORE CIS Facility Occupational dose rates. Specifically, evaluations and calculations are presented here for the following conditions and configurations:

- Occupational dose rates at the surface and 1 meter from a single HI-STORM UMAX Version C.
- Occupational dose rates at the surface, 0.5 meters, 1 meter, and 2 meters from the HI-TRAC CS.
- Estimated personnel exposures for loading operations of one canister at the HI-STORE CIS Facility.

The dose rates for the UMAX VVM Version C and the HI-TRAC CS are calculated for the source term case which have the burnup, cooling time, and initial enrichment combination shown in Table 7.1.1 of Reference [8].

2. *Acceptance Criteria*

There are no specific acceptance criteria for this analysis. Occupational doses to individuals are administratively controlled to ensure that they are maintained below 10CFR20.1201(a)(1) annual limits [11] i.e. the more limiting of:

i. The total effective dose equivalent being equal to 5 rem (0.05 Sv); or
ii. The sum deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rem (0.5 Sv).

Additionally, qualitatively a determination that the HI-STORM UMAX and HI-TRAC CS provide "suitable shielding" in accordance with 10CFR72.128(a)(2) [9].

3. *Computer codes and their benchmarking status*

Holtec International maintains an active list of QA validated computer codes on the Company's network that are approved for use in Safety significant projects.

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4. *Principal references*

[1] I.C. Gauld, O.W. Hermann, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," ORNL/TM-2005/39, Version 5.1, Vol. I, Book 3, Sect. S2, Oak Ridge National Laboratory, November 2006.

[2] I.C. Gauld, O.W. Hermann, R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," ORNL/TM-2005/39, Version 5.1, Vol. II, Book 1, Sect. F7, Oak Ridge National Laboratory, November 2006.

[3] X-5 Monte Carlo Team, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5", LA-UR-03-1987, Los Alamos National Laboratory April 2003 (Revised in 2008).

[4] USNRC Docket 72-1040, "Final Safety Analysis Report on The HI-STORM UMAX Canister Storage System", Holtec Report No. HI-2115090, Revision 3.

[5] *Shielding Analysis of the HI-STORM UMAX.* HI-2125194 Latest Revision. Holtec International.

[6] USNRC Docket 71-9373, "Safety Analysis Report on the HI-STAR 190 Package", Holtec Report No. 2146214, Revision 0.

[7] Final Safety Analysis Report for the HI-STORM FW Cask MPC Dry Storage System, HI-2114830 Latest Revision. Holtec International.

[8] USNRC Docket 72-1051, "Licensing Report on The HI-STORE CIS Facility" HI-2167374 R0. Holtec International.

[9] 10CFR72, Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater than Class C Waste," USNRC, Washington, DC.

[10] Thermal Analysis of HI-TRAC CS Transfer Cask. HI-2177553 Revision 0. Holtec International.

[11] 10 CFR Part 20 "Standards for Protection Against Radiation," Title 10, of the Code of Federal Regulations – Energy, Office of the Federal Register, Washington, D.C.

4.1 *Drawings*

The following drawings were used to develop the MCNP models in this report.

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5. *Approach and major assumptions to insure conservative results*

The occupational dose shielding analysis of the HI-STORE CIS Facility, can be separated into two distinct parts. The first is the generation of the radiation source terms to represent the spent nuclear fuel at the appropriate burnup and cooling time. The second part is the radiation transport simulation to calculate the dose rates near and far from a single cask.

The source terms are calculated using the computer codes SAS2H and ORIGEN-S from SCALE 5.1 [1], [2]. These codes are a widely accepted means of generating radiation source terms from spent nuclear fuel. The dose rates are calculated using the computer code MCNP5 Version 1.51 [3]. MCNP5 is a state of the art Monte Carlo code that offers coupled neutron-gamma transport using continuous energy cross sections in a full three-dimensional geometry.

The occupational dose rate calculations use the same calculational methodology employed in Reference [4].

The HI-STORM UMAX VVM model includes several key assumptions, as follows:

1. A standard 17x17 PWR design basis fuel assembly as defined in the HI-STORM UMAX FSAR [4], as shown in Table 5.2.1 of [4] is used for both the source term and MCNP calculations.

2. The cobalt-59 impurity level is assumed to be 0.8 g/kg for the hardware above and below the active fuel region. [Proprietary Information withheld per 10CFR2.390].

3. The fuel enrichment is conservatively assumed to be 5.0 wt% in the MCNP models. [Proprietary Information withheld per 10CFR2.390].

4. It is conservatively assumed that each loaded HI-STORM UMAX VVM contains 37 fuel assemblies each with a design basis BPRA present consistent with Section 5.2 of Reference [4].

5. The dose rates as a function of distance for the HI-STORM UMAX casks are calculated for the source term case presented in Table 7.1.1, which has a heat load of approximately 32 kW, consistent with the maximum allowable heat load of the HI-STAR 190 (Table 7.C.7 of Reference [6]).

Additional modeling assumptions, modeling deviations and discussion can be found in references [4] and [5].

The HI-TRAC CS normal conditions model includes several key assumptions, as follows:

1. There are some steel components at the bottom of the HI-TRAC CS that extend beyond the outermost radius of the main overpack. These components are conservatively not credited.
2. Conservatively the walls of the HI-TRAC CS are shorter than the dimensions shown in Section 1.5 Licensing Drawings [8].
3. The HI-TRAC CS optional shield ring is not credited.

6. *Input data & source*

The input data for SAS2H, ORIGEN-S and MCNP models are from references [4] and [5].

The concrete and soil compositions and densities are as described in Table 7.3.1 [8]. Additional material compositions and material properties of the storage system are provided in Subsection 5.3.2 and Table 5.3.2 in [4].

7. *Results and Safety Findings*

Dose rates around a HI-TRAC CS and around a single HI-STORM UMAX storage module, loaded with the MPC-37 and design basis fuel, are presented in Table 7.4.1 and 7.4.2 of Reference [8] respectively.

The estimated personnel exposures for loading operations of one canister at the HI-STORE CIS Facility are presented in Table 11.3.1 of Reference [8].

It is concluded from the shielding analysis and results that the HI-TRAC CS and HI-STORM UMAX provide suitable shielding in accordance with 10CFR72.128(a)(2) [9].

1. Introduction

The Introduction and scope of the analysis is presented in Section 1 of the Safety Analysis Summary.

This report is organized with a very short front section followed by detailed appendices. The actual calculations and evaluations are presented in the appendices.

2. General Methodology

The shielding analysis of the HI-STORE CIS Facility UMAX VVM Version C and HI-TRAC CS can be separated into two distinct parts. The first is the generation of the radiation source terms to represent the spent nuclear fuel at the appropriate burnup and cooling time. The second part is the radiation transport simulation to calculate the dose rates at various locations around a single cask.

The radiation source terms are calculated using the SAS2H and ORIGEN-S modules from the SCALE 5.1 [1] and [2] code system from Oak Ridge National Laboratory. This is a widely accepted means of generating radiation source terms from spent nuclear fuel.

The radiation transport simulation in the HI-STORM UMAX shielding models is performed with MCNP5 1.51 [3] from Los Alamos National Laboratory. MCNP is a Monte Carlo code that offers coupled neutron-gamma transport using continuous energy cross sections in a full three-dimensional geometry.

The reader is also referred to references [4], [5] for additional discussion of the methodology and calculation of the source terms.

3. Acceptance Criteria

The acceptance criteria are presented in Section 2 of the Safety Analysis Summary.

4. Assumptions

The major assumptions are listed in Section 5 of the Safety Analysis Summary.

5. Input Data

Input data are presented in Section 6 of the Safety Analysis Summary, and appropriately referenced within each Appendix.

6. Computer Codes

Computer Codes are listed in Section 3 of the Safety Analysis Summary.

7. Analysis and Results

The steps to determine dose rate as a function of distance from the HI-STORM UMAX VVM are outlined in Chapter 5 of Reference [4]. Results and safety findings are presented in Section 7 of the Safety Analysis Summary.

This section provides a brief description of the contents of each Appendix.

Appendix A. SAS2H/ORIGEN-S Source Terms: This Appendix provides the neutron, fuel gamma, and Cobalt-60 hardware source terms of SAS2H and ORIGEN-S calculations.

Appendix B. MCNP Filenames: This Appendix briefly describes the nomenclature used for MCNP calculations. With that, the content of each file can be derived from the filename.

Appendix C. MCNP Modeling of Casks: The MCNP modeling methodology and tally descriptions are provided. The results for a single cask dose vs. distance, and single cask 1 meter dose rates at the top, bottom, and side mid-height various sources (neutrons, gammas, etc.) are also provided.

8. Computer Files

All files are stored on the Holtec computer server in Camden, NJ[

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9. Summary

A summary of the results and safety findings is provided in Section 7 of the Safety Analysis Summary.

10. References

The references are listed in Section 4 of the Safety Analysis Summary.

10.1 Drawings

The drawings are listed in Section 4.1 of the Safety Analysis Summary.

Appendix A: SAS2H/ORIGEN-S Source Terms (total of 1 page)

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Appendix B: MCNP Filenames and Tally Specifications (total of 3 pages)

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Attachment A

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