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HI-STORE CIS Facility Site Boundary Dose Rates Calculations for HI-STORM UMAX System

FOR

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DOCUMENT CATEGORIZATION

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☐ Technical Report (Per HQP 3.2) (Such as a Licensing Report)

☐ Design Criterion Document (Per HQP 3.4)

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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-2177599R0)

Project title	HI-STORE CIS Facility
Report title	HI-STORE CIS Facility Site Boundary Dose Rates Calculations for the HI-STORM UMAX System
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 calculates the dose rates as a function of distance for 500 loaded UMAX VVMs containing fully loaded MPC-37 canisters at the HI-STORE CIS Facility. The dose rates as a function of distance 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]. Using a single UMAX VVM model, dose rates as a function of distance, with occupancy factors, are calculated at distances in the range of 10 meters to 1000 m to demonstrate compliance with 10CFR72.104 [9].

Additionally a fire accident with the transfer cask, the HI-TRAC CS is considered with the density of the concrete reduced to account for degradation from the fire. The dose is calculated for a 30-day accident duration and a 100 meter distance to demonstrate compliance with 10CFR72.106 [9].

2. *Acceptance Criteria*

The acceptance criteria for this analysis are dictated by 10CFR72.104 and 10CFR72.106 [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.

4.1 *Drawings*

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

[Proprietary Information withheld per 10CFR2.390]

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

The site boundary analysis of the HI-STORE CIS Facility with the ISFSI containing 500 loaded UMAX VVMs, 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 distance dose rate calculations use the same methodology employed in the site boundary MCNP cask model in Reference [4] with tallies at greater distances in the 10 meter to 1000 meter range.

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. Two separate occupancy factors are assumed: 2000 hours (40 hours per week, 50 weeks per year) and 8760 hr, which is full time occupancy for the entire year.

6. 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]).
7. No credit is taken for self-shielding of the UMAX VVM closure lids (i.e. one UMAX VVM closure lid above ground is not credited for shielding another UMAX VVM's dose contribution to the site boundary).

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

The HI-TRAC CS fire accident model includes several key assumptions, as follows:

1. The material composition of the HI-TRAC CS fire accident degraded concrete density is as described in Table 7.3.1 [8] and conservatively bounds the calculated degradation due to the fire accident calculated in Appendix B of Reference [10].
2. The accident duration is assumed to last 30 days, consistent with Reference [4].
3. The distance of the HI-TRAC CS under accident conditions to the Controlled Area Boundary is assumed to be 100 meters, consistent with Reference [4].
4. The height of the tally at 100 meters is 5 feet (1.524 meters) above ground level.

6. *Input data & source*

The input data and material compositions for SAS2H, ORIGEN-S and MCNP models are from references [4] and [5]. The UMAX VVM pitch in both the x and y directions of 17 feet (5.1816 meters) is shown in Table 1.1.1 of [8]. The ISFSI array for 500 loaded UMAX VVMs is modeled as a 20 × 25 array, crediting the additional x and y distances to the site boundary.

The distance from the nearest loaded UMAX VVM to the site boundary is a distance of 400 meters as is shown in Table 1.0.1.

The material 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*

The HI-STORM UMAX VVM ISFSI site boundary shielding analysis at the HI-STORE CIS Facility is presented in this report. The dose rates and annual dose (for occupancy factors of 2000 hrs/yr and 8760 hrs/yr) as a function of distance for the HI-STORM UMAX VVM ISFSI are presented in Table 7.4.3 of Reference [8]. The dose rates and annual doses are calculated at distances ranging from 100 meters to 1000 meters from the casks. Figure 7.4.3 of Reference [8] shows ISFSI dose rates as a function of distance.

The maximum controlled area boundary dose rate (assuming an occupancy of 2,000 hours per year) is below the 25 mrem annual dose limit of 10CFR72.104 [9].

The nearest residence is 1.5 miles from the HI-STORE CIS Facility. The dose calculations conservatively assume a full-time resident (8760 hours/year) is only 1000 meters from the nearest loaded UMAX VVM. In the case of this nearest residence, the dose is calculated to be below the 25 mrem annual dose limit prescribed in 10CFR72.104 [9].

The HI-TRAC CS fire accident results are presented in Table 7.4.4 of Reference [8], with the resulting accident dose (assuming a 30-day accident duration) at 100 m from the cask showing compliance with the requirements of 10CFR72.106 [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 analysis of the HI-STORE CIS Facility UMAX VVM ISFSI 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 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 for this analysis are dictated by 10CFR72.104 and 10CFR72.106, and are summarized here.

Normal condition requirements from 10CFR72.104 [9].

1. During normal operations and anticipated occurrences, the annual dose equivalent to an individual who is located beyond the controlled area, must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ.
2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

Accident condition requirements from 10CFR72.106 [9]

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 Rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem). The lens dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or to any extremity shall not exceed 0.50 Sv (50 rem). The minimum distance from the spent fuel or high level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

As discussed in Subsection 5.1.2 of the HI-STORM UMAX FSAR [4], the dose requirement of 10CFR72.106 is satisfied for the design basis accident, and no additional site specific analyses are required.

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. Please note that the results sections in the appendices might be page numbered separately from the 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

The HI-STORE CIS Facility shielding analysis of the UMAX VVM ISFSI is presented in this report.

The hourly and annual dose rates from the UMAX VVM ISFSI are presented in Table 7.4.3 of Reference [8]. The annual dose rates are specified for occupancy factors of 2000 hrs/yr and 8760 hrs/yr. The doses are calculated at distances ranging from 100 meters to 1000 m from the casks.

The results of this calculation demonstrate compliance with 10CFR72.104 [9].

Additionally, a HI-TRAC CS fire accident condition was considered in which the concrete is considered to be degraded. The dose at 100 meters from a single HI-TRAC CS for the fire accident is presented in Table 7.4.4 of Reference [8]. The results of this calculation demonstrate compliance with 10CFR72.106 [9].

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 2 pages)

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Appendix C: MCNP Modeling of Casks (total of 6 pages)

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Attachment A

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