Final Safety Evaluation Report

for the

HI-STORE Consolidated Interim Storage Facility

Independent Spent Fuel Storage Installation

Specific Materials License No. SNM-2516

Docket No. 72-1051

Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission

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ACRONYMS AND ABBREVIATIONS

AASHTO	American Association of State Highway and Transportation Officials
ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AFB	Air Force Base
AISC	American Institute of Steel Construction
ALARA	as-low-as-reasonably achievable
ALI	annual limits on intake
AMSL	above mean sea level
AMP	aging management programs
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASCE	American Society of Civil Engineers
ASM	Additional Security Measures
ASME	American Society of Mechanical Engineers
AST	aboveground storage tanks
ASNT	American Society for Nondestructive Testing
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BLM	Bureau of Land Management
BPVC	Boiler and Pressure Vessel Code
BECT	burnup, enrichment, and cooling time
BWR	boiling water reactor
C	Celsius
CCTV	closed-circuit television
CEC	canister enclosure container
CEUS-SSC	Central and Eastern United States Seismic Source Characterization
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CG	center of gravity
CISCC	chloride-induced stress corrosion cracking
CIS	Consolidated Interim Storage
CLSM	controlled low-strength material
CoC	Certificate of Compliance
CS	concrete shielded
CTB	canister transfer building
CTF	canister transfer facility
DAC	derived air concentrations
DCE	decommissioning cost estimate
DE	design earthquake
DECE	design extended condition earthquake
DFP	decommissioning funding plan
DOE	Department of Energy
DOT	Department of Transportation
DP	decommissioning plan
EIS	environmental impact statement
ELEA	Eddy-Lea Energy Alliance
EPRI	Electric Power Research Institute
EPA	Environmental Protection Agency

ACRONYMS AND ABBREVIATIONS

ER	environmental report
ERP	emergency response plan
ERO	emergency response organization
F	Fahrenheit
FAA	Federal Aviation Administration
FEMA	Federal Emergency Management Agency
FMCSA	Federal Motor Carrier Safety Administration
FRA	Federal Railroad Administration
FSAR	final safety analysis report
FW	flood and wind
GAO	Government Accountability Office
GED	General Education Development
GNEP	Global Nuclear Energy Partnership
GTCC	Greater than Class C
GMM	ground-motion model
HBF	high burnup fuel
HDRP	High Burnup Dry Storage Cask Research and Development Project
HEC-HMS	Hydrologic Engineering Center Hydrologic Modeling System
HEC-RAS	Hydrologic Engineering Center River Analysis System
HMR	Hydrometeorological Report
IDLH	Immediately Dangerous to Life and Health
IFR	instrument flight rule
ISFSI	independent spent fuel storage installation
ISG	Interim Staff Guidance
ISP	Interim Storage Partners
ISR	independent safety reviewers
ITS	important-to-safety
ITS-B	ITS Category B
ITS-C	ITS Category C
kg	kilogram
kN	kilonewtons
L	liters
LCO	limiting condition for operation
LLEA	local law enforcement agency
LLC	limited liability company
LWR	light water reactor
MAPS	managing aging processes in storage
MCNP	Monte Carlo N-Particle Transport
MMD	Mining and Minerals Division
MOX	mixed oxide
MPC	multi-purpose canister
mrem	millirem
MRS	monitored retrievable storage
MT	metric tons
MTR	military training route
MTU	metric tons uranium
NASEM	National Academies of Sciences, Engineering, and Medicine
NCEER	National Center for Earthquake Engineering Research
NEF	National Enrichment Facility

ACRONYMS AND ABBREVIATIONS

NFPA	National Fire Protection Association
NFHM	non-fuel hardware
NLCD	National Land Cover Database
NOAA	National Oceanic and Atmospheric Administration
NRCS	Natural Resources Conservation Service
NSHMP	National Seismic Hazard Mapping Project
NTSB	National Transportation Safety Board
NWS	National Weather Service
NM	New Mexico
NRC	Nuclear Regulatory Commission
NITS	not important to safety
NWS	National Weather Service
OCA	owner-controlled area
OSHA	Occupational Safety and Health Administration
Pa	pascals
PA	protected area
PCT	peak cladding temperature
PGA	peak ground acceleration
PIR	potential impact radius
PFS	Private Fuel Storage
psi	pound per square inch
psig	pound per square inch gauge
PHMSA	Pipeline and Hazardous Materials Administration
PSHA	probabilistic seismic hazard analysis
PMF	probable maximum flood
PMP	probable maximum precipitation
PSP	physical security plan
PWR	pressurized water reactor
QA	quality assurance
QAP	quality assurance program
RG	Regulatory Guide
RAI	request for additional information
ref-cm	reference cubic centimeters
RNAV	navigation routes
RQD	rock quality designation
SAR	safety analysis report
SCP	safeguards contingency plan
SEI	Structural Engineering Institute
SER	safety evaluation report
SFP	support foundation pad
SFST	Spent Fuel Storage and Transportation
SLD	special lifting devices
SNF	spent nuclear fuel
SLES	Safeguards LAN/Electronic Safe
SNM	special nuclear material
SPT	standard penetration test
SR	surveillance requirement
SRM	Staff Requirements Memorandum
SSC	systems, structures, and components

ACRONYMS AND ABBREVIATIONS

SSHAC	Seismic Hazard Analysis Committee
Sv	sieverts
TLD	thermoluminescent dosimeters
TNT	trinitrotoluene
TQP	training and qualifications plan
TS	technical specifications
USACE	U.S. Army Corps of Engineers
USDA	U.S. Department of Agriculture
10 CFR	Title 10 of the *Code of Federal Regulations*
USDA	U.S. Department of Agriculture
USGS	U.S. Geological Survey
VBS	vehicle barrier system
VCT	vertical cask transporter
VFR	visual flight rule
VVM	vertical ventilated module
WBD	watershed boundary dataset
WCS	Waste Control Specialists, LLC
WIPP	Waste Isolation Pilot Plant
WRCC	Western Regional Climate Center

EXECUTIVE SUMMARY

By letter dated March 30, 2017, Holtec International (the applicant) submitted a license application to the U.S. Nuclear Regulatory Commission (NRC) to construct and operate a consolidated interim storage (CIS) facility, which the applicant referred to as the HI-STORE CIS Facility, for interim storage of commercial spent nuclear fuel (SNF) in southeastern New Mexico. The application specifies a possession limit for SNF of 8,680 metric tons (9,590 short tons) total of uranium in the form of undamaged fuel assemblies, damaged fuel assemblies, and fuel debris in 500 loaded canisters. The applicant updated and supplemented its application in numerous submittals to the NRC, most recently on January 20, 2023.

The applicant prepared the application consistent with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." The applicant also relied on information provided in Regulatory Guide 3.50, "Standard Format and Content for a Specific License Application for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility," Revision 2, issued September 2014, to prepare a draft license. The application consists of the following documents:

- A **safety analysis report (SAR)**, in which the applicant described its plans for designing, constructing, operating, maintaining, and decommissioning the proposed HI-STORE CIS Facility, as required by 10 CFR 72.24, "Contents of application; Technical information." The applicant prepared the SAR using NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000, and applicable interim staff guidance that supplements it.

- A **proposed license** and associated **technical specifications**, as required by 10 CFR 72.26, "Contents of application: Technical specifications," and 10 CFR 72.44, "License conditions."

- An **emergency plan**, in which the applicant described its plan for resolving any emergencies that happen during the HI-STORE CIS Facility's operation, as required by 10 CFR 72.32, "Emergency Plan."

- An **environmental report**, in which the applicant provided information, as required by 10 CFR 72.34, "Environmental report," that the staff used for its environmental review, conducted in parallel with preparation of the staff's safety evaluation report (SER). The staff published a draft environmental impact statement (EIS) in March 2020, a final EIS in July 2022, and a supplement to the final EIS in November 2022.

- A **physical security plan**, in which the applicant described its plans for ensuring that the HI-STORE CIS Facility and nuclear material are appropriately protected. This document contains nonpublic safeguards information. It includes the security training and qualification plan and safeguards contingency plan, as required by 10 CFR 72.180, "Physical protection plan," and 10 CFR 72.184, "Safeguards contingency plan." The staff performs this review in parallel with the safety evaluation and documents the review in a separate security evaluation.

- A report discussing the **financial qualifications** of the applicant, in which the applicant described its plans and qualifications to fund the construction, operation, and decommissioning of the proposed facility, as required by 10 CFR 72.22, “Contents of application; General and financial information,” and 10 CFR 72.30, “Financial assurance and recordkeeping for decommissioning.”

- A **training and qualifications program**, in which the applicant describes the training and qualification requirements that personnel at the facility must meet to engage in the proposed activities authorized by this license, as required by 10 CFR 72.28, “Contents of application; Applicant’s technical qualifications.”
- A description of the **quality assurance program** for the facility, in which the applicant describes the policies and procedures it will implement to ensure that the design, fabrication, construction, testing, operation, modification, and decommissioning of important to safety structures, systems, and components, as well as managerial and administrative controls, at the facility are conducted in accordance with 10 CFR Part 72, Subpart G, “Quality Assurance,” as required by 10 CFR 72.24, “Contents of application; Technical information.”
- A **decommissioning plan**, in which the applicant describes the proposed practices and procedures for decommissioning the facility, as required by 10 CFR 72.30, “Financial assurance and recordkeeping for decommissioning.”

This SER documents the staff’s review and conclusions on the safety-related aspects of the license application. The staff conducted its technical review in accordance with the applicable NRC regulations in 10 CFR Part 20, “Standards for Protection Against Radiation,” and 10 CFR Part 72. The staff reviewed the applicant’s SAR, the supplemental analyses referenced within, and other analyses, reports, and documents incorporated by reference in the SAR. The staff’s review followed the guidance in NUREG-1567[1], applicable regulatory guides, and interim staff guidance.

Unless otherwise stated, this SER references information in SAR Revision 0T, submitted January 20, 2023; documents referenced in or attached to the SAR; the applicant’s responses to the staff’s requests for additional information; and other relevant literature.

As noted above, the license application requests authorization to temporarily store commercial SNF at the HI-STORE CIS Facility, an away-from-reactor independent spent fuel storage installation (ISFSI), for a license period of 40 years. At least 2 years before the end of this license term, the licensee may apply to renew the license. The proposed HI-STORE CIS Facility would provide an option for storing SNF from U.S. commercial nuclear power reactors.

Description of the HI-STORE CIS Facility Site

The applicant proposed to locate the HI-STORE CIS Facility on a 423-hectare (1,045-acre) parcel owned by ELEA, LLC. ELEA, LLC, is an alliance of (in alphabetical order) the city of Carlsbad, the county of Eddy, the city of Hobbs, and the county of Lea, which together surround

[1] After the March 2017 receipt of the license application, NUREG-1567 was superseded by NUREG-2215 (published in April 2020). NUREG-2215 incorporates the guidance in NUREG-1567 and all interim staff guidance documents (ISGs) cited in this SER. The staff’s findings in this SER are not affected by the supersession of NUREG-1567.

the proposed site (ELEA is a composite of Eddy and Lea counties). The site is located approximately 1.6 kilometers (1 mile) north of U.S. Highway 62, 51 kilometers (32 miles) east of Carlsbad, and 55 kilometers (34 miles) west of Hobbs. A portion of the site is currently used for cattle grazing. Land uses in the nearby area include oil and gas exploration and production, oil and gas-related services industries, underground potash mining, livestock grazing, and limited recreational activity. The Pecos River is the closest through-flowing surface water feature to the site. At its nearest approach, the distance from the site to the Pecos River is 42 kilometers (26 miles).

Description of the HI-STORE CIS Facility Storage Systems

The HI-STORE CIS Facility would consist of the HI-STORM UMAX ISFSI pads (where the seal-welded dry storage canisters are stored), a rail spur and cask receiving area, an equipment building to store the HI-TRAC CS, the vertical cask transporter, ancillaries and spare parts, an administrative building to house inspection, security and administrative staff as well as access control facilities, a security building at the entrance to the facility to house security personnel and equipment, including those described in the physical security plan, and health physics staff and equipment, as required. The NRC previously certified the HI-STORM UMAX canister storage system in Certificate of Compliance (CoC) No. 1040. The HI-STORM UMAX system is the subterranean version of the HI-STORM FW storage system, which the NRC previously certified in CoC No. 1032. The SAR incorporates by reference portions of the HI-STORM UMAX Final SAR, Revision 3, and the HI-STORM FW Final SAR, Revision 4.

Safety of the HI-STORE CIS Facility

The staff determined that the proposed HI-STORE CIS Facility and proposed storage system are structurally sound and that the facility, if constructed and operated in accordance with the design and limits specified in the license and technical specifications, can provide safe and secure interim storage of the SNF during all normal, off-normal, and credible accident conditions. The analyses of potential hazards provided in the application include natural and human-made phenomena. The natural phenomena considered and evaluated in the application include potential seismic activity near the site, fires, flooding, lightning, and tornado winds, as well as missiles generated by these. The human-made phenomena considered in the application include fires, explosions, cask drops, cask tipovers, hazards from nearby oil and gas production and exploration activities, potash mining operations, nearby gas pipelines explosions, transportation of hazardous materials on nearby rail lines and highways, and commercial and military aircraft crashes. After a detailed review of the applicant's analyses, the staff concluded that none of these events would pose a credible hazard to the HI-STORE CIS Facility and that the proposed storage system design is structurally safe and will meet all applicable regulatory requirements.

The staff has also determined that the applicant has shown that the SNF within the storage casks will remain subcritical (i.e., unable to sustain a nuclear chain reaction) during all phases of operation under all normal, off-normal, and credible accident conditions. The cask systems are seal welded to prevent leakage of radioactive material. Additional radiation shielding is provided by transportation, transfer, and storage casks during handling and storage.

The applicant provided radiation dose estimates for the surrounding public and the workers at the proposed HI-STORE CIS Facility. The applicant estimated that members of the public nearest to the proposed HI-STORE CIS Facility would receive radiation doses below NRC regulatory requirements, which for normal conditions of operation is 0.25 millisievert (mSv) per year (yr) (25 millirem/yr) and for credible accidents is 0.05 Sv/yr (5 rem/yr). The applicant also calculated radiation dose rates within the vicinity of individual casks to demonstrate that workers at the HI-STORE CIS Facility will not receive doses that exceed 0.05 Sv/yr (5 rem/yr), the NRC annual regulatory limit for workers at nuclear facilities. The NRC has established these radiation dose limits to prevent any undue risk and to ensure the safety of all members of the public and workers at a nuclear facility. The applicant also described its radiation protection program, which employs a radiation protection principle of as low as is reasonably achievable. Radiation doses received by the workers and dose rates within the vicinity of the storage area will be monitored to verify that radiation dose limits are not exceeded. The staff reviewed the analyses provided by the applicant and concluded that the HI-STORE CIS Facility and the proposed cask designs are radiologically safe and will meet regulatory requirements.

As required by 10 CFR Part 72, the applicant demonstrated that all structures, systems, and components (SSCs) of the proposed HI-STORE CIS Facility that are important to safety would continue to perform their design functions during normal and off-normal conditions and during any credible accidents postulated to occur. Based on its review and evaluation of the information provided, the staff concluded that the applicant has provided acceptable analyses of the design and performance of these SSCs important to safety under normal, off-normal, and accident conditions. The staff further concluded that the applicant's analyses related to off-normal and accident events demonstrate that the HI-STORE CIS Facility will be sited, designed, constructed, and operated so that public health and safety will be adequately protected during all credible off-normal and accident events, and that the capability for SNF to be retrieved from storage will be preserved.

The staff evaluated the HI-STORM UMAX canister storage system against the parameters and conditions specific to the HI-STORE CIS Facility site and the SNF to be stored. Based on its review, the staff finds that the use of the HI-STORM UMAX canister storage system as proposed for the HI-STORE CIS Facility is acceptable in accordance with the site-specific provisions of 10 CFR Part 72, subject to the conditions of the license.

SNF would be transported to the facility by rail using the HI-STAR 190 transportation package, which the NRC has previously approved, and SNF shipped to the facility would be subject to the limits for burnup, enrichment, and cooling time specified in Revision 3 of the HI-STAR 190 SAR, dated November 2, 2018.

Other Requirements

To demonstrate its financial qualification, the applicant identified anticipated sources of funds for the HI-STORE CIS Facility. The NRC staff concludes that the applicant provided reasonable assurance of its financial qualifications for construction, operation, and decommissioning of the HI-STORE CIS Facility.

The staff also found that the applicant's emergency plan appropriately described the HI-STORE CIS Facility's program for responding to onsite emergencies and for seeking offsite assistance, if necessary.

Lastly, the staff found that Revision 1 of the applicant's physical security plan meets all the applicable regulatory requirements to ensure that operation of the HI-STORE CIS Facility will be protected from theft, sabotage, diversion, or malicious acts, and provides for the common defense and security. The staff's evaluation of the physical security plan, transmitted to the applicant under separate cover, is not available to the public because it contains sensitive unclassified information designated as safeguards information.

References

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

Holtec International, "Final Safety Analysis Report on the HI-STORM FW MPC Storage System," Holtec Report No. HI-2114830, Revision 4, NRC Docket No. 72-1032, June 24, 2015. Agencywide Documents Access and Management System Accession No. ML15177A338.

Holtec International, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Holtec Report No. HI-2115090, Revision 3, NRC Docket No. 72-1040, June 29, 2016. ML16193A339.

Holtec International, "Holtec International HI-STORE CIS (Consolidated Interim Storage Facility) License Application," NRC Docket No. 72-1051, March 30, 2017. ML17115A431.

Holtec International, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Physical Security Plan," Revision 3 (nonpublic), Holtec Report No. HI-2177559, H-011-20, transmittal dated March 2, 2020. ML20065H155.

Holtec International, "Environmental Report on the HI-STORE CIS Facility," Holtec Report No. HI-2167521, Revision 8, August 31, 2020. ML20295A485.

Holtec International, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground CISF—Financial Assurance & Project Life Cycle Cost Estimates" (proprietary), Holtec Report No. HI-2177593, Revision 2, August 9, 2022. ML22227A162. Public version ML22331A008.

Holtec International, "HI-STORE CIS Facility Decommissioning Cost Estimate and Funding Plan," Holtec Report No. HI-2177565, Revision 1 (proprietary), November 16, 2022. Public version ML22331A012.

Holtec International, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Emergency Response Plan," Revision 5, Holtec Report No. HI-2177535, NRC Docket No. 72-1051, November 17, 2022. ML22331A015.

Holtec International, "Attachment 2 - Proposed HI-STORE License/Technical Specifications," November 23, 2022. ML22331A005.

Holtec International, "Licensing Report on the HI-STORE CIS Facility," Holtec Report No. HI-2167374, Revision 0T, NRC Docket No. 72-1051, January 20, 2023. ML23025A112.

U.S. Nuclear Regulatory Commission, Regulatory Guide 3.50, "Standard Format and Content for a Specific License Application for An Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility," Revision 2, September 2014. ML14043A080.

1 GENERAL DESCRIPTION

On March 30, 2017, Holtec International (Holtec or the applicant) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a specific independent spent fuel storage installation (ISFSI) license to construct and operate the HI-STORE Consolidated Interim Storage (CIS) Facility in Lea County, New Mexico, in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.” The license application seeks NRC approval to store up to 8,680 metric tons of commercial spent nuclear fuel in the HI-STORM UMAX canister storage system for a 40-year license term. The NRC staff conducted an acceptance review of the license application and supporting documents, as supplemented, and, by letter dated February 28, 2018, determined that the application provided sufficient information for the staff to docket the application and begin a detailed safety, security, and environmental review.

The HI-STORE CIS Facility is an ISFSI that uses NRC-certified dry cask storage technology. In accordance with 10 CFR 72.42, “Duration of license; renewal,” the NRC will initially license the HI-STORE CIS Facility for a 40-year license term. At least 2 years before the end of this license term, the licensee may apply to renew the license.

1.1 Scope of Review

The staff reviewed Holtec’s license application for the HI-STORE CIS Facility in accordance with the applicable requirements in 10 CFR Part 72 and consistent with the staff review guidance in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000, including section 1.4, “Acceptance Criteria,” and section 1.5, “Review Procedures,” and Regulatory Guide 3.50, “Standard Format and Content for a Specific License Application for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility,” Revision 2, issued September 2014.

1.2 Regulatory Requirements

The regulations in 10 CFR Part 72 establish the requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel, power-reactor-related Greater that Class C waste, and other radioactive materials associated with spent fuel storage at an ISFSI and the terms and conditions under which the Commission will issue these licenses. The following regulations detail the regulatory requirements relevant to the general description of the proposed HI-STORE CIS Facility:

- 10 CFR 72.22, “Contents of application: General and financial information”
- 10 CFR 72.24, “Content of application: Technical information”

1.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed the information submitted by the applicant in Revision 0T of its safety analysis report (SAR), dated January 20, 2023, as well as other information submitted in response to the staff’s requests for supplemental information, requests for additional information, and information incorporated by reference. The staff evaluated the applicant’s description of the location, principal functions, design features, and construction and operation schedules for the HI-STORE CIS Facility in SAR chapter 1, “General Description.” The staff visited the site on several occasions during its review to confirm the information presented in the application. Additionally, the staff also reviewed the licensing reports, supplemental analyses, and calculation packages provided in the license application to confirm that the application contained the specific information required by 10 CFR 72.22(a) through (d).

1.3.1 General Description of Installation and General Systems Description

The staff reviewed SAR section 1.0, "Introduction," section 1.1, "General Description of Installation," and section 1.2 "General Systems Description," which provide an overview of the proposed HI-STORE CIS Facility. The applicant described the location of the proposed facility in Lea County, New Mexico; the materials to be stored at this facility; the spent fuel storage systems to be used; and the principal design criteria for these systems. In addition, in SAR table 1.0.3, "Systems and Documents Incorporated by Reference for HI-STORE," table 1.0.4, "Canisters Allowed for Storage in HI-STORM UMAX at HI-STORE," and table 1.0.5, "Transport Casks Allowed for Receipt of Canisters at HI-STORE," the applicant identified the specific storage and transportation systems it proposes to use at the site. The dry cask storage system incorporated by reference is based on the two types of canisters for spent fuel storage that are certified for use in Certificate of Compliance No. 1040 for the HI-STORM UMAX canister storage system. Because the final SAR for the HI-STORM UMAX system references the final SAR for Certificate of Compliance No. 1032 for the HI-STORM FW canister storage system, the applicant also incorporated by reference specific portions of the HI-STORM FW system in the HI-STORE CIS Facility SAR. The applicant described how all spent fuel transfer, handling, and retrieval operations will be conducted in seal-welded canisters, which confirms that the HI-STORE CIS Facility will not employ a spent fuel pool or individual fuel handling capability, and that no gaseous, liquid, or solid radioactive waste treatment systems are associated with the proposed operations at the site. In addition, the applicant described the major structures and components at the HI-STORE CIS Facility, including the canister transfer building and UMAX ISFSI pads, and explained how spent fuel would be received, offloaded, and stored.

As part of the proposed license conditions, the applicant stated in Proposed Conditions 6A, 7A, and 8A that it will store up to 8,680 metric tons (9,590 short tons) of uranium in the form of undamaged fuel assemblies, damaged fuel assemblies, and fuel debris. This quantity of spent fuel would be stored in a maximum of 500 multipurpose canisters (MPC-37 or MPC-89) that will be emplaced in two HI-STORM UMAX ISFSI pads with 250 cavity enclosures each.

In addition to the description of the facility provided in SAR section 1.1, the staff also reviewed the layout of the HI-STORE CIS Facility given in SAR section 1.5, "Drawings," including Drawing No. 10940, "HI-STORE Site Plan and General Arrangement" (proprietary), which shows a general illustration of the proposed locations of important structures and buildings at the site, including the canister transfer building, security building, ISFSI pads, equipment storage building, administration building, vehicle barrier system, security fence, and controlled area boundary. The staff determined that this site layout and its respective buildings, structures, and systems are consistent with the descriptions in SAR sections 1.1 and 1.2.

Based on its review, the staff determined that the applicant provided a broad, nonproprietary overview of the HI-STORE CIS Facility that can be used to familiarize interested parties with the features of the facility, including its principal design characteristics. The staff compared the drawings provided in SAR section 1.5 with the detailed drawings presented elsewhere in the SAR and determined that the drawings are consistent. The staff also determined the drawings in SAR chapter 1 provide sufficient detail to describe the operational concept and safety-related features of the HI-STORE CIS Facility. The staff determined that the facility description is consistent with use of the facility for interim storage and not permanent disposal, and that the applicant provided sufficient information about the spent fuel it proposes to store to familiarize interested parties with the layout and operations at the HI-STORE CIS Facility.

The staff also determined the applicant provided sufficient information for interested parties to understand the operating systems used at the site. The staff found the information acceptable because it meets the requirements of 10 CFR 72.24(b), (f), and (l) regarding descriptions of structures, features, and operational modes to reduce to the extent practicable radioactive waste volumes generated at the facility and to maintain control over radioactive materials in gaseous and liquid effluents.

1.3.2 Identification of Agents and Contractors

The staff reviewed the description of agents and contractors in SAR section 1.3, "Identification of Agents and Contractors." The applicant stated that, in addition to its role as applicant, Holtec will also retain responsibility for engineering, design, and construction of all aspects of the facility structures, buildings, and equipment. The applicant described its capabilities for design, fabrication, and operation, including its experience providing engineering, design, and fabrication services for other dry cask storage facility licensees in the United States and internationally. The applicant stated that any external fabricators for construction or fabrication of equipment or structures at the HI-STORE CIS Facility will be evaluated and audited in accordance with its NRC-approved quality assurance program. In SAR table 1.0.2, "Projected Milestone Dates for HI-STORE CIS," the applicant laid out a proposed schedule for construction and operation of the HI-STORE CIS Facility.

Based on its review, the staff determined that the applicant described the agents involved in design, construction, and operation of the HI-STORE CIS Facility, including principal consultants, service organizations, and quality assurance service providers. The staff verified that the application clearly defined the division and assignment of responsibilities among the participants on the HI-STORE CIS Facility project.

1.3.3 Material Incorporated by Reference

The staff reviewed SAR section 1.0, "Introduction," which references SAR table 1.0.3, "Systems and Documents Incorporated by Reference for HI-STORE." SAR table 1.0.3 identifies the HI-STORM UMAX System, NRC docket no. 72-1040, and the HI-STORM FW multipurpose canisters (MPCs) MPC-37 and MPC-89, NRC docket no. 72-1032, as sources of information incorporated by reference in the application. SAR table 1.0.3 also identifies the Holtec International quality assurance manual under docket no. 72-1084. As discussed in chapter 12 of this safety evaluation report, the latest approved version of the quality assurance manual is Revision 15, dated June 10, 2022.

In each SAR chapter where the applicant incorporated by reference specific information, the applicant included a table entitled, "Material Incorporated by Reference in this Chapter." These tables, which identify the specific information incorporated by reference, the source of the information, and other pertinent information, appear in SAR chapters 5, 6, 7, 8, 9, 11, 15, and 16. The tables identify the following sources of the information incorporated by reference:

- HI-STORM UMAX FSAR, Revision 3, dated June 30, 2016
- HI-STORM FW FSAR, Revision 4, dated June 24, 2015
- HI-STAR 190 transportation package SAR, Revision 3, dated November 2, 2018

- HI-STORM UMAX certificate of compliance, initial issuance and amendments 1 and 2, dated April 6, 2015, September 8, 2015, and January 9, 2017

Additionally, section 2.1 of the applicant's proposed technical specifications, "Approved Contents, Fuel Specifications and Loading Conditions," identifies specific information incorporated by reference from the HI-STORM UMAX certificate of compliance and HI-STAR 190 SAR listed above.

The staff confirmed that the applicant identified all documents incorporated by reference in the license application and summarized this information in its respective chapter, section, or both, of the SAR.

1.4 Evaluation Findings

Based on its review of the information provided in the application, the staff determines the following:

- The information presented in the license application and SAR chapter 1 satisfies the requirements for general information in 10 CFR 72.22 and for design and operational information in 10 CFR 72.24(b), (f), and (l). This finding is based on a review that considered the regulations themselves; Regulatory Guide 3.48, Revision 1, NUREG-1567, and accepted practices.

- The applicant identified the agents and contractors responsible for the design, construction, and operation of the installation.

- The applicant tabulated all technical reports, supplemental analyses, and docketed material incorporated by reference in the SAR.

1.5 References

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

Holtec International, Inc., "Final Safety Analysis Report on the HI-STORM FW MPC Storage System," Holtec Report No. HI-2114830, Revision 4, NRC Docket No. 72-1032, June 24, 2015. Agencywide Documents Access and Management System Accession No. ML15177A338.

Holtec International, Inc., "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Holtec Report No. HI-2115090, Revision 3, NRC Docket No. 72-1040, June 29, 2016. ML16193A339.

Holtec International, Inc., "Holtec International HI-STORE CIS (Consolidated Interim Storage Facility) License Application", March 30, 2017. ML17115A418.

Holtec International, Holtec Report No. HI-2146214, "Safety Analysis Report on the HI-STAR 190 Package," Revision 3, November 2, 2018. ML18306A911.

Holtec International, Inc., “Licensing Report on the HI-STORE CIS Facility,” Holtec Report No. HI-2167374, Revision 0T, NRC Docket No. 72-1051, January 20, 2023. ML23025A112.

U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage),” Revision 1, August 1989. ML003739163.

NRC, NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” March 2000. ML003686776.

NRC, “Certificate of Compliance for Spent Fuel Storage Casks, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040,” April 6, 2015. ML15093A498.

NRC, “Certificate of Compliance for Spent Fuel Storage Casks, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040, Amendment No. 1,” September 8, 2015. ML15252A426.

NRC, “Certificate of Compliance for Spent Fuel Storage Casks, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040, Amendment No. 2,” January 9, 2017. ML16341B061.

NRC, “Letter to Kimberly Manzione, Holtec International re: Holtec International’s Application for Specific Independent Spent Fuel Storage Installation License for the HI-STORE Consolidated Interim Storage Facility for Spent Nuclear Fuel – Accepted for Review.” February 28, 2018. ML18059A251.

2 SITE CHARACTERISTICS

In chapter 2, “Site Characteristics,” of Revision 0T of the safety analysis report (SAR), dated January 20, 2023, the applicant described the site of the proposed Holtec Consolidated Interim Storage (CIS) Facility, the population distribution around the site, land and water uses, and associated site activities. The applicant also evaluated the site characteristics with respect to safety and identified assumptions applied when evaluating the design for safety, establishing installation criteria for the design, and providing the design bases in other evaluations in the SAR.

2.1 Scope of Review

The staff evaluated the site characteristics in chapter 2 of the SAR by reviewing the documents cited in or attached to the SAR, the applicant’s responses to the staff’s requests for additional information (RAIs), and other relevant literature. The staff reviewed the application information on site characteristics to determine whether the applicant (1) identified the site characteristics, including natural and human-induced phenomena, and addressed the adequacy of the design basis with respect to these phenomena, (2) characterized local land and water use and population to determine the critical populations likely to be affected, and (3) characterized the transport processes that could move any released contamination from the facility to the maximally exposed individuals and populations. To make its determination, the staff reviewed the information in the application on area geography and demography; nearby industrial, transportation, and military facilities; and the meteorological, hydrological, geological, and seismological characteristics of the site and its surrounding area.

2.2 Regulatory Requirements

The regulatory requirements relevant to the siting evaluation of the proposed Holtec HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.40, “Issuance of license”
- 10 CFR 72.90, “General considerations”
- 10 CFR 72.92, “Design basis external natural events”
- 10 CFR 72.94, “Design basis external man-induced events”
- 10 CFR 72.96, “Siting limitations”
- 10 CFR 72.98, “Identifying regions around an ISFSI [independent spent fuel storage installation] or MRS [monitored retrievable storage] site”
- 10 CFR 72.100, “Defining potential effects of the ISFSI or MRS on the region”
- 10 CFR 72.103, “Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003”

- 10 CFR 72.122, “Overall requirements”

Where appropriate, the staff makes findings of regulatory compliance with the requirements of 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” which are fully addressed in chapter 2 of the SAR. Because compliance with some regulations can only be determined by the integrated review of several sections in SAR chapter 2, other chapters in the SAR, or both, the staff did not make a finding of regulatory compliance in each major section of this SER chapter unless the applicant fully addressed the specific regulatory requirement. However, the staff made findings of the technical adequacy and acceptability regarding information reviewed in the application for each section in chapter 2 of the SAR. The staff’s findings on regulatory compliance appear in this chapter at the end of its complete review of a subject area and are summarized in Section 2.4.

2.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed and evaluated the site characteristics of the proposed site discussed in chapter 2 of the SAR, documents cited in or attached to the SAR, the applicant’s responses to the staff’s RAIs, and other relevant literature. The review covered site geography and demography; nearby industrial, transportation, and military facilities; meteorology; surface hydrology; subsurface hydrology; and geology and seismology.

2.3.1 Geography and Demography

The staff reviewed the information in SAR section 2.1, “Geography and Demography,” that discussed site location, site description, population distribution and trends, and land and water use.

2.3.1.1 Site Location

The staff reviewed the applicant’s description of the proposed site location in the introduction to section 2.1 of the SAR and the applicant’s proposed technical specification 4.1, “Site.” SAR section 2.1.1, “Site Location,” describes the site location. The proposed facility will be located in Lea County, New Mexico, and within the property currently owned by Holtec. The property includes the entire Section 13 of Township 20 South, Range 32 East, and portions of adjacent Section 17 and Section 18 of Township 20 South, Range 33 East, in the New Mexico Public Land Survey system, as shown in SAR figure 2.1.14, “Surface Land Ownership in the Vicinity of the Site.” The proposed facility will be located within Section 13 and is approximately 55 kilometers (km) (34 miles (mi)) west of Hobbs, New Mexico, and approximately 51 km (32 mi) east of Carlsbad, New Mexico. Lea and Eddy Counties of New Mexico and Andrews and Gaines Counties of Texas surround the proposed site.

As described in SAR table 1.0.1, “Overview of the HI-STORE Facility,” the site consists of 423 hectares (1,045 acres) of land. The SAR provides the latitude and longitude of the site with appropriate maps and aerial photographs.

Based on its review of the description of the site location given in the SAR, the staff finds it acceptable because it clearly describes the geographic location of the site, including the site’s relationship to political boundaries and natural and anthropogenic features. The maps provided in the SAR are acceptable because they provide sufficient detail to review the geographical, geological, and engineering features of the proposed facility. The staff finds that the information presented in this section is acceptable for use in other sections of the SAR to develop the design bases for the HI-STORE CIS Facility, perform safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.24(a), 10 CFR 72.90(a), 10 CFR 72.90(e), and 10 CFR 72.98(a) with respect to this topic.

2.3.1.2 Site Description

The staff reviewed SAR section 2.1.2, “Site Description,” which describes the site using maps to delineate the site boundary and the controlled area. The proposed facility will be located within the land currently owned by Holtec, purchased from the Eddy-Lea Energy Alliance (ELEA), a limited liability company owned by Eddy and Lea Counties and the cities of Carlsbad and Hobbs, New Mexico. The ISFSI pad will be located within the site boundaries in the western half of the site.

SAR figures 2.1.1, 2.1.2, and 2.1.5 show the location and orientation of the HI-STORE CIS Facility structures with respect to nearby roads. There is no obvious way in which the traffic on adjacent roads can interfere with ISFSI operations. The site is mostly undeveloped land currently used for cattle grazing. SAR figures 2.1.9 through 2.1.11 show the topography in the vicinity of the ISFSI site. The high point of the topography is on the southern border of the site. The topography slopes gently toward two drainage areas, Laguna Plata and Laguna Gatuna. There is no wetland on the site. SAR figure 2.1.5 shows an aerial view of the site with full buildout along with the site vicinity. The site includes existing structures such as the following:

- a producing well of gas and distillate (Hanson State #001), located between the communication tower and the facility boundary on the southern edge of Section 13,
- a small water drinker for livestock, located along the water pipeline in the northeast corner of Section 13 outside the facility boundaries,
- a communication (cell) tower in the southwest corner of Section 13,
- an abandoned oil recovery facility with tanks and associated hardware in the northeast corner of the property outside Section 13,
- an oil recovery facility with tanks and associated hardware beyond the northeast corner of the property outside Section 13,
- an abandoned oil recovery facility in the far southeast corner of the property,
- a temporary flexible pipeline for natural gas aboveground diagonally through the center of the site and,
- existing underground pipelines of natural gas at the east of the site running in the north-south direction and a section of a pipeline along the east-west edge of Section 13 south of the proposed facility.

SAR section 2.1.2 states that the water pipeline and temporary flexible pipeline for natural gas will be relocated and the water drinker will be removed before construction at the site begins. During the licensed term of the proposed facility, temporary pipelines will not be allowed to traverse the site unless they can be shown to not pose a hazard to the safety-related structures. There are no water wells at the site.

At least 18 plugged and abandoned oil and gas wells have been located within the property boundaries; however, none of these wells are within the area where the proposed ISFSI would be located or where any land would be disturbed. There are no plans to activate any of these 18 plugged and abandoned oil and gas wells. SAR figure 2.1.25, “Oil and Gas Wells within 1 mile of HI-STORE Facility,” shows the oil and gas wells within 1.6 km (1 mi) of the facility, and SAR table 2.1.4, “Oil and Gas Wells within 1 mile of HI-STORE Facility,” gives basic information about these wells. The only producing gas and distillate well, Hanson State #001, is outside the HI-STORE CIS Facility protected area boundary, as shown in SAR figure 2.1.25 and discussed SAR section 2.2.2, “Pipelines.” There are seven aboveground storage tanks (ASTs) associated with past brine disposal activities. The capacity of these tanks varies between 24,000 and 40,000 liters (L) (150 to 250 barrels). These holding tanks are used for separating residual oil from oilfield brine and storing brine. None of these ASTs are currently in use, and no containers of hazardous substances are in use, as stated in SAR section 2.1.2. In addition, 40,000 L (250-barrel) capacity ASTs are present at the location of the Hanson State #001 well at the southwest corner of Section 13 outside the facility protected area.

As described in Holtec Report No. HI-2210487, “HI-STORE Gas Pipeline Risk Evaluation” (proprietary), Revision 2, dated August 12, 2022, three natural gas pipelines lie east of the proposed site boundary running north-south: (1) a DCP Mainstream 508-millimeter (mm) (20-inch)-diameter pipeline approximately 0.26 km (0.16 mi) away, (2) a DCP 203 mm (8-inch)-diameter pipeline approximately 0.27 km (0.17 mi) away, and (3) a Lucid Energy 152 mm (6-inch)-diameter pipeline approximately 0.26 km (0.16 mi) away. An extension of the Lucid Energy pipeline turns west and runs approximately 12 meters (m) (40 feet) south of the southern edge of the site boundary to the Hanson State #001 well pad. Another pipeline is planned approximately 1.3 km (0.8 mi) west of the western site boundary.

As described in SAR section 2.1.2, the site does not support any vegetation with any economic significance or public interest. Vegetation at the site and in the immediate surrounding area is primarily desert grasslands, which are characterized by a significant amount of grasses and less than 10 percent of total cover being shrubs and forbs. A biological survey conducted in 2016, summarized in appendix B of “Environmental Report on the HI-STORE CIS Facility” (ER), Revision 9, dated November 20, 2020, found that the vegetation of the area has changed from desert grassland to mesquite scrubland due to overgrazing.

Based on its review of the site description given in the SAR, the staff finds that the site description is adequate because the descriptive information and maps delineate the site boundary, controlled area, general natural and human-made features, topography, and surface hydrologic features. The staff also finds that this information is acceptable for use in other sections of the SAR to develop the design bases for the facility, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.90(a)–(e) and 10 CFR 72.98(a) with respect to this topic.

2.3.1.3 Population Distribution and Trends

The staff reviewed SAR section 2.1.3, “Population Distribution and Trends.” This section describes the population distribution and trends near the proposed facility. The population data used in the SAR were derived from the U.S. Census Bureau and from estimates of future population provided by the State of New Mexico and the State of Texas. Employers within 8 km (5 mi) of the proposed site provided information on the transient population.

As shown in SAR figure 2.1.12, “Region of Influence with a 50-Mile Radius of the Site,” the area within a radius of 80 km (50 mi) of the site includes Eddy, Chaves, and Lea Counties of New

Mexico and Andrews, Loving, Culberson, Reeves, Winkler, Yoakum, and Gaines Counties of Texas. This area within all the counties is primarily rural. Based on the data from the U.S. Census Bureau, the population within a radius of 80 km (50 mi) of the site was 166,914 in 2015. Between 2010 and 2015, the population within this area grew at a slower rate compared to that of the rest of New Mexico.

There are five ranches within a 10 km (6 mi) radius of the site. Fewer than 20 people reside in these five ranches, giving a population density of five residents per square mile. The nearest residence is the Salt Lake Ranch approximately 2.4 km (1.5 mi) from the proposed site. SAR figure 2.1.13, “Sector Population Map,” shows the distribution of the resident population in concentric circles with radii of 1.5, 3, 5, 6.5, and 8 km (approximately 1, 2, 3, 4, and 5 mi) and in 22.5-degree segments. Only nine residents live within 8 km (5 mi) of the proposed site.

Between 2015 and 2025, the population within a 10 km (6 mi) radius of the site is projected to grow at a slower rate, approximately 10 percent, compared to the population in the other parts of New Mexico, which is projected to grow at a rate of approximately 19 percent. With the projected growth rate (10 percent), the population within an 8 km (5 mi) radius of the site is estimated to increase to 10 persons, as stated in SAR section 2.1.3.

Approximately 303 persons are currently employed within 8 km (5 mi) of the proposed facility in farming and mining professions and comprise the entire transient population. The nearest schools, colleges, daycares, nursing homes, and hospitals are located in Hobbs, New Mexico. None of these types of facilities are projected to open during the 40-year licensing period within 8 km (5 mi) of the proposed facility. Holtec could not reasonably foresee additional projects within this area in the near future. Additionally, employers in this area have not forecasted any change in the existing transient population. Consequently, Holtec has concluded that the transient population near the proposed site would remain steady at the current level in the near future, as stated in SAR section 2.1.3.

Based on its review of the information presented in the SAR, the staff determined that the population distribution and trends in the region have been adequately described and assessed. The sources of the population data used in the SAR are the U.S. Census Bureau and employers for nearby projects. The staff determined these sources to be appropriate and the basis for population projections to be reasonable. The staff determined that the applicant has appropriately investigated the region with respect to the present and future character and distribution of the population; therefore, the staff finds that the requirements of 10 CFR 72.98(c)(1) have been met. This information is also acceptable for use in other sections of the SAR to develop the design bases for the proposed facility, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.90(e), 10 CFR 72.98(a), and 10 CFR 72.100(a), with respect to this topic.

2.3.1.4 Land and Water Use

The staff reviewed SAR section 2.1.4, “Land and Water Use,” which describes land and water uses in the region surrounding the proposed site. Lands within 10 km (6 mi) surrounding the proposed site are either privately owned, State lands, or Bureau of Land Management (BLM) lands, as shown in SAR figures 2.1.14 and 2.1.15, “Land Ownership near the CIS Facility Site.” The BLM owns and manages the majority of the land immediately surrounding the site, as shown in SAR figure 2.1.14. The land surrounding the proposed site is used for livestock grazing; mineral extraction; oil and gas exploration, production, and related services; and limited recreational activities. Agriculture is a major activity within 80 km (50 mi) of the proposed site, along with mining, oil- and gas-related activities, and livestock grazing. Land ownership and use in this extended area are similar to those within the region 10 km (6 mi) from the proposed site.

Extraction of natural resources near the proposed site includes underground potash mining, caliche mining, and oil and gas production. SAR figure 2.1.16, “Mineral Resources Near the Site,” shows the mineral resources near the proposed site. Both the potash mining and the oil and gas industries use major surface facilities, although mining-related surface facilities are confined to a fairly small area. As stated in SAR section 2.1.4, based on information from the U.S. Geological Survey (USGS) and New Mexico Mining and Minerals Division (NM MMD) (2023), the proposed site is not co-located with any existing mining facilities. The staff verified that the NM MMD “Registered Mines in New Mexico” (NM MMD, 2023) show that the there are no mines co-located with the proposed site.

The proposed site is within the Secretary of the Interior’s Designated Potash Area, according to U.S. Department of the Interior Order No. 3324, “The Secretary of the Interior: Oil, Gas, and Potash Leasing and Development within the Designated Potash Area of Eddy and Lea Counties, New Mexico,” published in the *Federal Register* (FR) in 2012 (77 FR 71814, December 4, 2012) in accordance with the Mineral Leasing Act of 1920, the Mineral Leasing Act for Acquired Lands of 1947, and the Federal Land Policy and Management Act of 1976. Two mining companies currently hold the mining leases for extracting potash near or under the proposed site: Intrepid Potash—New Mexico, LLC, and Mosaic Potash Carlsbad Inc. The staff’s review focuses on the Intrepid Potash mines because the Mosaic mine is located more than 16 km (10 mi) from the site.

Intrepid Potash operates two potash mines near the proposed site. The Intrepid North Mine (previously called the National Potash Mine) is located approximately 6.7 km (4.2 mi) west of the proposed site. According to NM MMD (2021), active underground mining ceased in 1982, and the mine is permanently closed. However, the surface facilities are currently used to manufacture potash products. The Intrepid East Mine, located approximately 7.8 km (4.9 mi) southwest of the proposed site, is currently mining potash ore from its underground workings. SAR figure 2.1.17, “Mined Potash near the CIS Facility Site,” and figure 2.1.23, “Potash Mines in the Vicinity of the Facility,” show that the nearest mine workings from the Intrepid East Mine are approximately 3.4 km (2.1 mi) from the southwest boundary of the site. Potash is mined at a depth of 550 to 915 m (1,800 to 3,000 feet) near the proposed site, mostly using the room-and-pillar mining method with conventional underground mining machineries, as stated in SAR section 2.1.4. Intrepid Potash also uses a solution mining method to extract potash in the HB Solution Mine, located over 24 km (15 mi) from the proposed site.

Near the proposed site, caliche occurs close to the surface or at depths between 3.3 and 6.7 m (10 and 20 feet) from the surface. It is mined using traditional excavation machinery at the Caliche Mine, located approximately 6.4 km (4.0 mi) southwest of the proposed facility.

Exploration and extraction of oil and gas are well established in the area surrounding the proposed site. Numerous wells have been drilled surrounding the proposed facility through the potash-bearing strata to explore for and produce oil and gas, as illustrated in SAR figure 2.1.20, “Oil and Gas Activity near the CIS Facility Site.” Oil and gas are extracted from the relatively shallow Yates Formation at a depth of 930 m (3,050 feet) to deep gas targets in middle Paleozoic formations at depths lower than 4,880 m (16,000 feet). One producing gas and distillate well (Hanson State #001) is currently present between the proposed site boundary and the communication tower, as described in SAR section 2.1.2. In addition, at least 18 plugged and abandoned wells are located within the site. Extraction of oil and gas below the potash mine workings is possible. The BLM and the New Mexico Oil Conservation Division control how oil and gas operations are to be conducted following U.S. Department of the Interior Order

No. 3324 and State of New Mexico Oil Conservation Commission Order No. R-111-P, "Application of the Oil Conservation Division upon Its Own Motion to Revise Order R-111, as Amended, Pertaining to the Potash Areas of Eddy and Lea Counties, New Mexico," dated April 21, 1988 (State of New Mexico Oil Conservation Commission, 1988), in the potash enclave in New Mexico, according to "Environmental Geology Assessment Report for Issues Related to the Proposed Ochoa Mine, Lea County, New Mexico," issued in 2012 by AECOM. Mines historically leave protective pillars surrounding oil and gas wells as a precaution. Order No. 3324 establishes a buffer zone of 0.8 km (0.5 mi) around gas wells and drill islands and 0.4 km (0.25 mi) around oil wells and drill islands (SAR figure 2.1.25). No potash mining operation can be conducted within the BLM buffer zones, which surrounds much of the site as illustrated in SAR figure 2.1.14. Well drilling and completion operations are required to be conducted in a manner that prevents salt dissolution and protects the well through its productive life, often 20 to 30 years or more. Decommissioning a well has similar requirements.

As stated in Order No. 3324, new wells on Federal land must be drilled from a drill island. The Belco Tetris and Tetris Anise drill islands are located near the northern edge of Sections 22, 23, and 24. Section 24 is adjacent to and immediately south of Section 13, where the proposed facility would be located. The BLM has approved the Tetris Anise drill island as part of the Paske West Development Area. Additionally, Titus Oil & Gas Production has proposed the Egg Roll Development Area, which would use the Tetris Anise Drill Island to develop multiple zones in the Bone Spring formation (approximately 2,400 m (7,800 feet) below the ground surface) and the Wolfcamp formation, which is below the Bone Spring formation (SAR figure 2.6.2, "Geologic Cross Section through the Capitan Reef Area, Eddy and Lea Counties, NM"). All active or plugged wells within 1.6 km (1 mi) of the site, as shown in SAR figure 2.1.25 and listed in SAR table 2.1.4, are more than 305 m (1,000 feet) from the storage location at the proposed facility.

Salt from the subsurface is extracted in the Delaware Basin using injection wells, called brine wells. Fresh water is injected into the underground through these wells to dissolve the salt formations and extract saturated brine. A few brine wells in Eddy County suffered catastrophic collapse. These wells were 61 to 91 m (200 to 300 feet) in diameter and 30 to 60 m (100 to 200 feet) deep. This brine is used for oil and gas well drilling.

Catastrophic subsidence has occurred due to suspected oil well casing corrosion and dissolution of salt formations, such as Wink Sinks I and II and the Jal Sink. Similar incidents have occurred in oil fields having salt deposits in Texas and Kansas. The Wink sinks developed in the Hendrick oilfield in Winkler County, Texas. The Jal Sink is located near Jal, New Mexico. It is suspected that a failure of the well casing allowed unsaturated water to come in contact with and dissolve salt layers.

Nonmunicipal irrigation consumes the largest amount of water in Lea County. Municipalities are the next largest water user group, with water rights of 5,921 hectare-meters (48,000 acre-feet). The City of Hobbs holds the largest water rights of 2,479 hectare-meters per year (20,100 acre-feet per year). Over the next 40 years, water use in Lea County is projected to increase in all water-use categories to approximately 44,405 hectare-meters (360,000 acre-feet). A major increase of more than 90 percent of the water used in 1995 is anticipated from the irrigated agriculture for animal feed to be used by the dairy industry in Lea County (35,771 hectare-meters (290,000 acre-feet) in 2040). SAR section 2.1.2 states that there are no water wells at the proposed site.

Based on its review of the description of the land and water use near the proposed site in SAR section 2.1.4, the staff finds that the applicant has adequately identified and assessed the land

and water use for the site of the proposed facility. The staff finds that the applicant has appropriately investigated the region of the proposed site with respect to present and projected future uses of land and water within the region. The staff finds that this information is acceptable for use in estimating the concentrations of radionuclides to populations from any potential airborne or liquid effluents released from the proposed facility, developing the design bases for the facility, performing additional safety analyses, and demonstrating compliance with regulatory requirements in 10 CFR 72.98(a)–(c) with respect to this topic.

2.3.2 Nearby Industrial, Transportation, and Military Facilities

The staff reviewed the information in SAR section 2.2, "Nearby Industrial, Transportation, and Military Facilities," for human-made hazards from facilities that might affect the proposed HI-STORE CIS Facility. This review identifies the credible possible hazards from these nearby facilities that might endanger the radiological safety of the proposed facility, including potential onsite and offsite hazards. The applicant categorized the nearby facilities, located within 8 km (5 mi) of the proposed site, into five broad groups: industrial facilities (SAR section 2.2.1), pipelines (SAR section 2.2.2), air transportation (SAR section 2.2.3), ground transportation (SAR section 2.2.4), and nuclear facilities (SAR section 2.2.5). No military facilities are located within 8 km (5 mi) of the proposed site.

2.3.2.1 Nearby Industrial Facilities

The industrial facilities within 8 km (5 mi) of the proposed site include the Land Firm, Caliche Mine, Intrepid Potash Mines, and Transwestern Compressor Station.

The Land Firm is located 3 km (1.9 mi) southwest of the proposed facility. This facility remediates contaminated soil from oil and gas operations using the microbial degradation technique. Contaminated soil is trucked to the facility. Consequently, it is unlikely that this facility would present a credible hazard for operation at the proposed facility.

Caliche generally occurs near the surface or at depths of 3 to 6 m (10 to 20 feet) and is mined using conventional excavation machinery approximately 6.4 km (4 mi) southwest of the proposed facility. Consequently, the staff finds that caliche mining would not pose a credible hazard for operation at the proposed facility because of their large distance from the proposed facility and their use of excavation machineries only to mine caliche.

The Transwestern gas compressor station, which is located approximately 8.4 km (5.2 mi) southwest of the proposed site, compresses the natural gas flowing through the gas pipelines to the desired pressure, as stated in SAR section 2.2.1, "Industrial Facilities." Because of the large distance separating them, the staff finds it unlikely that the compressor station would pose a credible hazard for operation at the proposed facility.

2.3.2.1.1 Nearby Potash Mines

SAR section 2.1.4 describes the activities and hazards associated with mining potash in several underground mines surrounding the proposed site. The applicant provided additional information in responses dated September 16, 2020, and August 16, 2021, to the staff's RAIs 2-9, 2-10, 2-11, 2-12, 2-12-S, and 2-13. The staff also consulted the documents referenced in the SAR and additional literature to assess the subsidence-related hazards associated with the mining activities to extract potash near the proposed site. The U.S. Nuclear

Regulatory Commission (NRC) staff review focused on whether past, present, and future potash extraction at the nearby mines would pose a hazard to safe operation of the proposed facility.

As noted in section 2.3.1.4 of this safety evaluation report (SER), the proposed site is located within the Designated Potash Area defined by U.S. Department of the Interior Order No. 3324. The order ensures coordinated co-development of the oil, gas, and potash deposits within the Designated Potash Area. The State of New Mexico Oil Conservation Commission, by Order No. R-111-P, developed a similar coordinated co-development agreement to assure maximum conservation of the oil, gas, and potash resources of New Mexico within the Potash Area in non-federally owned land. These orders impose restrictions on drilling for oil and gas wells and mining of potash ore within the proposed site and its surroundings.

As described in SAR section 2.1.4, potash is mined at a depth of approximately 550 to 915 m (1,800 to 3,000 feet) from the Salado Formation near the proposed site. The Salado Formation is nearly flat near the proposed site. Potash has been mined nearby using the room-and-pillar mining method. In southeastern New Mexico, the typical height of the rooms is 1.8 to 3.0 m (6 to 10 feet), and width typically varies from 7.6 to 9.1 m (25 to 30 feet). The typical length of a mine room varies between 6.1 and 9.1 m (20 and 30 feet), as described in the applicant's response to RAI 2-11, dated September 16, 2020. In the first mining or pillar development phase, 60 to 70 percent of ore is mined out of the potash seam. During the "pillar robbing" or retreating phase, the extraction of potash could reach 90 percent or higher. Generally, a single ore zone is worked in a mine.

Only the room-and-pillar mining method is used to extract potash near the proposed site. Any subsidence hazard associated with potash mining would be from room-and-pillar mining. The HB Solution Mine, operated by Intrepid Potash, uses the solution mining technique to extract the portion of the ore left in the remnant pillars in the old room-and-pillar workings; however, the mine is over 24 km (15 mi) west of the site. No solution mine workings are located near the site, as stated in the applicant's September 16, 2020, response to RAI 2-10. Consequently, the staff accepts that solution mining at the HB Solution Mine is not expected to affect the proposed facility, as the solution mine is a significant distance away from the proposed facility.

Past and Current Potash Mining at Nearby Mines

As described in SAR section 2.1.4, two mining companies currently hold mining leases near or under the proposed site for extracting potash: Intrepid Potash—New Mexico, LLC and Mosaic Potash Carlsbad, Inc. Intrepid Potash—New Mexico, LLC holds a lease to mine potash beneath the proposed site.

Subsidence is the response of the overlying strata to construction of an underground opening. The overlying strata become unsupported as the potash ore is mined out and, consequently, the strata deform, resulting in both vertical and horizontal ground movements that may eventually reach the surface. If this ground movement is not controlled or minimized, it can cause damage to both surface and underground structures. Damage visible at the surface includes surface depressions, cracking or collapse, sinkholes, and modifications of surface drainage pathways.

Relatively little deformation has been observed during the first mining phase or pillar development phase when the extraction ratio is less than 60 to 70 percent. However, during the retreat mining or pillar robbing phase, the extraction ratio can reach 90 percent or more, the remnant pillars start failing, and roof fracturing occurs. The disturbance propagates upward and manifests as surface cracks and depressions. Following AECOM's 2012 environmental geology assessment report, SAR section 2.1.4 states that, based on historical data and anecdotal evidence in the potash mines of southeastern New Mexico, surface subsidence is virtually complete within 5 to 7 years after completion of the retreat mining. However, potash, being a viscoelastic material, will creep, and minor protracted subsidence may continue beyond the 5 to 7 years after completion of the retreat mining. Literature supports the conclusion on time to complete subsidence in southeastern New Mexico. Based on “Geomechanical Analyses to Investigate Wellbore/Mine Interactions in the Potash Enclave of Southeastern New Mexico,” by J.G. Argüello, et al., published in 2009, major movement at the surface occurred within 175 to 200 days after pillar extraction at the U.S. Potash mine (currently called Intrepid Potash West mine). Subsidence was nearly complete in 240 days. The surface movement was minimal after 500 days. Based on the foregoing discussion, the staff finds it reasonable to assume that the surface subsidence profile in the nearby potash mines would be virtually complete in 5 to 7 years following completion of retreat mining.

As described in SAR section 2.1.4, the maximum depth of subsidence is generally limited by the mining height; however, the maximum subsidence is rarely achieved, and a typical 1.2 m (4 feet) of subsidence may occur in rooms with a height of 1.8 m (6 feet) (67 percent of the excavation height). Argüello, et al. (2009) reported the measured maximum subsidence at different potash mines near the proposed site based on the information available in the literature. Approximately 1.9 m (6.4 feet) of surface subsidence was measured at the U.S. Potash mine (currently called Intrepid Potash West mine); with a mining height of 2.7 m (9 feet), the maximum subsidence is approximately 71 percent of the excavation height. At the MS Chemical Leo mine (currently called Intrepid North Mine), maximum subsidence of 1.2 m (4 feet) was observed in a panel with 1.5 m (5-foot)-high rooms; 80 percent of the excavation height. AECOM (2012) also reported subsidence ranging from 74 to 77 percent of the excavation height 5 to 7 years after completion of the retreat mining. Further subsidence with time was minimal. Measurements over both conventional (room and pillar) and longwall mining panels at the U.S. Potash mine suggest a rule of thumb that the maximum subsidence is approximately two-thirds of the excavation height (Argüello, et al., 2009). Based on the above discussion, the staff finds it reasonable to assume a maximum depth of surface subsidence at the nearby potash mines of approximately 67 percent of the room (excavation) height. Consequently, the expected maximum subsidence near the proposed site after the retreat phase of room-and-pillar mining would be approximately 1.2 m (4 feet) because the room height is typically 1.9 m (6 feet).

The extent of surface subsidence, the subsidence trough, is generally larger than the mined-out area and is primarily controlled by the depth, width, and height of the mine workings, the extraction ratio, and the time elapsed since the opening was created. The extent of surface subsidence, called the angle of draw, is expressed as the angle of inclination between the vertical direction at the edge of the mine openings and the point on the surface undergoing zero displacement at the edge of the subsidence trough. Based on a November 1979 report by Golder Associates, the SAR has taken the upper limit of the angle of draw to be 45 degrees to define the extent of subsidence beyond the outermost mine workings. The NRC staff independently searched the literature to find actual observations of the angle of draw in potash mines in southeastern New Mexico. Givens et al. (1985) observed an angle of draw 35 degrees within the Carlsbad Potash District from the results of the National Geodetic Survey. Historic subsidence studies at several potash mines near the proposed site are reviewed by Argüello, et al. (2009), and AECOM (2012). At the U.S. Potash mine (currently called Intrepid West mine and closed since 2016), which is approximately 23 km (14 mi) from the proposed site, the estimated angle of draw varied from 30 to 51 degrees (Argüello, et al., 2009). It was believed that the angle of draw increased with weaker strata. Subsidence measurements at the Southwest Potash mine (also known as the AMAX mine) showed a maximum angle of draw of 36 degrees (Argüello, et al., 2009). Based on the historic information in nearby potash mines, the NRC staff finds that an angle of draw of 45 degrees would be reasonable to assess the potential hazard from mining-induced subsidence to the proposed facility. The State of New Mexico Oil Conservation Commission, in its Order No. R-111-P, also supports a 45-degree angle of draw for the extension of the subsidence trough from the outer mine working). As the potash is mined at a depth of 550 to 915 m (1,800 to 3,000 feet), the subsidence trough would be expected to be less than 1.0 km (0.6 mi) from the outermost mine workings. Consequently, mining subsidence would not be a hazard to any surface facility at least 1.0 km (0.6 mi or 3,200 feet) away from the outermost mine working near the proposed site.

As stated in SAR section 2.1.4 and shown in SAR figures 2.1.17 and 2.1.23, the past and present mine workings of both the Intrepid Potash North and Intrepid Potash East mines are more than approximately 3.4 km (2.1 mi) away from the proposed site. Considering the area of influence from subsidence, the estimate based on the depth of the mining and a 45-degree area of influence, that mining operations ceased at the Intrepid Potash North mine in 1982, that no mining has taken place at the nearest mine workings of the Intrepid East Mine, and that most of the surface subsidence ends within 5 to 7 years after completion of mining, appreciable future surface subsidence is not expected from nearby existing mine workings. Based upon the preceding discussion, the staff concludes that the proposed site would not be affected by the subsidence from past and current potash extraction at the nearby mines.

<u>Future Potash Mining in Section 13 and Surroundings</u>

There is no current proposal to mine potash around or beneath the proposed site, as per the letters from Intrepid Potash—New Mexico, LLC, dated August 12, 2021, and Mosaic Potash Carlsbad Inc., dated August 13, 2021 to Mr. Ed Mayer, Program Director, Holtec International. Additionally, the applicant does not expect that mining of potash around and beneath the proposed site in Section 13 would occur for the duration of the license for the proposed facility because of several factors, as discussed in SAR section 2.1.4:

- the complexity of mine development due to significant oil and gas operations nearby
- the distance from the current mine workings in nearby potash mines
- the economic viability of potash extraction in Section 13

The BLM and the State of New Mexico’s Energy, Minerals and Natural Resources Department coordinate development and extraction of potash mineral and oil and gas reserves with the local lease holders following U.S. Department of the Interior Order No. 3324 and the State of New Mexico Oil Conservation Commission Order No. R-111-P. Order No. 3324 establishes a buffer zone around oil and gas wells. In accordance with the Order, a potash mine will maintain a buffer zone with a radius of 0.8 km (0.5 mi) for a gas well and a buffer zone with a radius of 0.4 km (0.25 mi) from an oil well for protection on Federal land, as shown in SAR figure 2.1.24, “Order 3324 Oil & Gas Drill Island Buffer Zones.”

Intrepid Potash—New Mexico, LLC, can mine the potash ore around and beneath the proposed site in Section 13 by extending the current mine workings or constructing new mine shafts with associated surface facilities to access the orebody. Intrepid Potash—New Mexico, LLC, can approach the site from the existing mine workings of the Intrepid East Mine workings approximately 3.4 km (2.1 mi) southwest of the proposed site, as stated in SAR section 2.1.4 and in the applicant's August 16, 2021, response to RAI 2-12-S. These mine workings would have to be extended through a narrow region between the buffer zones of the "Snoddy Federal #021H" (API 30-025-40338) well and the "Felmont Federal Com #002" (API 30-025-26302) well, under Laguna Gatuna, and into Section 13, as shown in SAR figure 2.1.24. The new extension of the existing mine workings to reach and mine the potash ore in Section 13 would be over 4.8 km (3 mi) in length, as stated in the response to RAI 2-12-S. Alternatively, Intrepid Potash—New Mexico, LLC, could construct new shafts with ancillary surface facilities to extract the potash ore in Section 13. Again, based on SAR figure 2.1.14 "Surface Land Ownership in the Vicinity of the Site," the new shafts would be on Federal land if they are located south of Section 13. If located north of Section 13, the shafts could either be on Federal land or on private land, depending on the location.

Mosaic Potash Carlsbad Inc. does not own mineral rights close to the proposed site, including in Section 13. Consequently, Mosaic Potash Carlsbad Inc. would have to acquire the mineral rights and develop an entirely new mining facility, which would include sinking a new mine shaft, establishing support facilities on the surface, and procuring mining machinery, as stated in the Holtec responses to RAIs dated August 16, 2021.

Based on the above discussion, the applicant concluded that mining potash ore around or beneath the site of the proposed facility would not take place during the licensing period because of the complexity of mining in the oil and gas fields, the distance from the existing mine workings, and the significant capital investment for such an operation in Section 13.

The staff reviewed the information presented in the SAR and in response to RAIs 2-8, 2-9, 2-10, 2-11, 2-12, and 2-12-S. The staff also reviewed the information available in the literature to assess whether future mining of potash ore from beneath and around the proposed site in Section 13 would be feasible. Intrepid Potash—New Mexico, LLC, holds the mining lease for potash mineral underneath the proposed site. The staff assessed whether potash mining could reasonably occur during the licensing period of the proposed facility beneath Section 13 and its surroundings.

Based on its review, the staff finds that the potash ore beneath Section 13 and its surroundings can only be mined by constructing new mine workings from the existing workings of the Intrepid Potash East mine or sinking new shafts and developing a new mining facility with associated surface support infrastructure. The staff has not considered the Intrepid Potash North mine because it is currently not registered as a mine (NM MMD, 2023). To extract the potash ore from beneath Section 13, at least two headings from the existing mine workings or two shafts from the surface would be needed for proper ventilation and safe mining operations.

The staff review finds that the new extension of the existing mine workings to access Section 13 would likely be entirely on Federal land, as shown in SAR figures 2.1.14 and 2.1.23. New headings from existing mine workings would traverse the region between the buffer zones of the "Snoddy Federal #21H" (API 30-025-40838) well and the "Felmont Federal Com #002" (API 30-025-26302) well, as stated in the applicant's response to RAI 2-12-S and shown in SAR figure 2.1.24. As shown in SAR figure 2.1.24, both of these wells are gas wells. In accordance with Order No. 3324, on Federal land a 0.8 km (0.5 mi) radius buffer zone is imposed precluding new potash mining. The region between these two buffer zones is less than approximately 300 m (1,000 feet) wide, as stated in the response to RAI 2-12-S. As discussed in the State of New Mexico Oil Conservation Commission Order No. R-111-P, "[r]elease of methane into potash mine workings would endanger the lives of miners and would render further mining activities uneconomic because of the additional, and more expensive safety requirements which would be imposed by the Mine Safety and Health Administration . . . of the U.S. Department of Labor." Consequently, planning and operation to develop these headings need to be well controlled to avoid any flow of methane into the mine workings. In response to RAI 2-12-S, the applicant estimated that a heading to reach the potash ore beneath Section 13 would be over 4.8 km (3 mi) long. From the Intrepid Mine East, two potential new excavation headings would be more than 9.6 km (6 mi) long before reaching an area for potash extraction in Section 13. Excavation of such long headings to reach the ore underneath Section 13 would require significant capital outlay, on the order of several tens of million dollars, before any ore extraction starts, based on the cost estimate provided by the applicant dated August 31, 2022 (Attachment 3 to Holtec Letter 5025069 (proprietary), 2022). In addition, the ore haulage and ventilation costs would increase significantly, thereby increasing the production cost of potash mined from this area.

Alternatively, Intrepid Potash—New Mexico, LLC, could access the potash ore in Section 13 by constructing shafts (two would be needed) with new surface support facilities. A portion of the ore surrounding the shafts needs to be left in place (shaft protection pillar) for shaft stability. It is noted that the size of the shaft protection pillar needs to be larger than the depth of potash-bearing strata, which is 550 to 915 m (1,800 to 3,000 feet), to avoid any impact on the shaft stability from the subsidence caused by any nearby mine workings, assuming the angle of draw of 45 degrees. Sinking new shafts to extract potash in Section 13 would also be an expensive investment (on the order of several tens of millions of dollars before any potash extraction), based on the cost estimate provided by the applicant.

The staff independently conducted a preliminary estimate of the cost of mining the potash beneath Section 13 using the pre-feasibility cost models given in Camm and Stebbins (2020). Depending on the production capacity varying from 360,000 to 600,000 ton/year, the capital cost before commencing the potash extraction could be tens of millions of dollars or more in 2019 dollars for either approach discussed above to access the potash beneath the site. Therefore, the staff concludes that the cost estimate given by the applicant is reasonable. Additionally, if accessed via extensions of the existing mine workings, the staff expects that it would cost significantly more to extract potash in Section 13 than the pre-feasibility cost estimate because this cost estimate does not account for extremely long excavations to access and mine the ore, significantly longer than in a typical room and pillar mine.

The "Mineral Commodity Summaries 2023," issued in 2023 by the USGS, reports that most of the U.S. potash reserves are in Montana, North Dakota, Utah, Arizona, and Michigan. In addition, the staff finds that the mines in the United States produce only a small fraction, approximately 11 percent, of the domestic consumption of potash. The rest (89 percent) is imported from Canada (79 percent), Belarus (7 percent), Russia (9 percent), and other countries (5 percent). World potash production capacity was 64 million tons in 2022, compared to the world consumption of 40 million tons. Several new mines and expansion projects in Belarus, Canada, and Russia are projected to increase the world production capacity in the near future. New mining projects in Australia, Canada, Eritrea, Brazil, Ethiopia, Spain, and US are planned to begin production in 2025.

Based on the preceding discussion, the staff concludes with reasonable assurance that it is highly unlikely that potash ore beneath Section 13 and its immediate vicinity would be mined during the licensed life of the proposed facility, given the significant complexity to mine only a small portion of the potash reserve and the economic viability of such a project, particularly given the abundance of alternative mining sites and excess potash production capacity worldwide. In addition, the mine management of both the Intrepid Potash and Mosaic informed Holtec in letters dated August 12, 2021, and August 13, 2021, respectively, that they do not have any plans to extract the potash beneath Section 13 . Therefore, for all these reasons, the staff concludes that past, present, and future potash mining near the proposed facility would not pose a hazard to its safe operation.

In summary, the applicant examined the proposed site and its surroundings for activities associated with potash mining. The staff finds that the requirements of 10 CFR 72.90(a)−(c) have been satisfied. The staff finds that the applicant appropriately assessed the past, present, and future potash mining at nearby potash mines and its effects on safe operations at the proposed facility. The assessment is based on prevalent mining practices in southeastern New Mexico, past and present mining locations, the Federal land surrounding the site, Order No. 3324, and the presence of nearby oil and gas operations. Therefore, the staff finds that the requirements of 10 CFR 72.94(a)−(c) have been satisfied. The applicant appropriately identified the region surrounding the proposed site where potential mining of potash may affect safe operations at the proposed facility. Therefore, the staff finds that the requirements of 10 CFR 72.98(a) have been satisfied.

2.3.2.1.2 Transport of Hazardous Materials by Highway 62/180

The applicant assessed the potential hazards to the proposed facility from hazardous materials being transported on the nearby U.S. Highway 62/180. The assessment is described in SAR section 2.2.4, “Ground Transportation,” and is documented in Holtec Report No. HI-2210620, “HI-STORE Highway 62/180 Hazardous Chemicals Risk Evaluation” (proprietary), Revision 2, dated August 12, 2022. The staff reviewed the assessment as well as responses to RAIs 2-14, 2-14-S, and 2-14-S-1, dated September 16, 2020; August 16, 2021; and April 15, 2022, respectively; pertinent literature; and databases maintained by different governmental agencies.

As described in SAR section 2.2.4, U.S. Highway 62/180 runs more than 762 m (2,500 feet) south of the proposed site. It is the closest and most-trafficked public road near the proposed site. It is a divided highway with a maximum speed limit of 113 km per hour (70 miles per hour (mph)) near the proposed site, as shown in SAR figures 2.2.1 and 2.2.4. The applicant stated that commercial shipments of hazardous materials transported over U.S. Highway 62/180 include a wide range of hazardous materials, including, but not limited to gasoline, diesel fuel, acids, carbon dioxide, nitrogen, liquid nitrogen, chlorine gas, refrigerants, fuel gases, oxygen, explosives, and low-level radioactive materials.

U.S. Highway 62/180 is on the National Hazardous Materials Route Registry, updated in the *Federal Register* on July 14, 2014 (79 FR 40844), and can be used for the transportation of radioactive waste materials to the Waste Isolation Pilot Plant (WIPP) facility. The route to the WIPP facility is approximately 8 km (5 mi) southwest of the proposed site, as shown in SAR figure 2.2.3, “WIPP Transportation Route.” Although radioactive materials destined to the WIPP facility typically do not use U.S. Highway 62/180, there have been instances where transuranic wastes associated with the WIPP facility were transported along this highway.

The applicant assessed the potential for a highway incident involving a release of hazardous materials with very low Immediately Dangerous to Life and Health (IDLH) limits to affect the operations at the proposed facility. The applicant selected chlorine to be the most hazardous chemical because its IDLH concentration is 10 parts per million. The staff agrees with the selection as other hazardous chemicals have higher IDLH concentrations (i.e., are less dangerous). Additionally, as transuranic radioactive waste is not typically transported to the WIPP facility using U.S. Highway 62/180, the staff agrees with the applicant's selection of chlorine as the most hazardous chemical that can be transported near the proposed facility. Holtec Report No. HI-2210620 states that chlorine is generally transported in tanker trucks as "bulk packaged" shipment for road transport. Tanker trucks have a typical capacity between 7,570 and 37,855 L (2,000 and 10,000 gallons (gal)).

In 49 CFR 107.1, "Definitions," an incident is defined as "an event resulting in the unintended and unanticipated release of a hazardous material." This includes the release of reportable quantities of hazardous materials during transport. The applicant used the Hazmat Incident Database maintained by the Pipeline and Hazardous Materials Administration (PHMSA) of the U.S. Department of Transportation (DOT). Based on this database, 168,680 highway incidents with hazardous materials took place between January 1, 2010, and January 1, 2021. Out of these, more than 80 percent (135,512) of the incidents involved 38 hazardous materials listed in table 5-1, "Highway Incident Hazmat Release Report Data for US," of Holtec Report No. HI-2210620. The chemical most frequently released in a large truck crash is paint (PHMSA Id. No. UN1263), and chlorine (PHMSA Id. No. UN1017) is not one of these 38 hazardous materials. Only 13 incidents (0.0077 percent of all releases) involved a chlorine release within this time period.

Additionally, the applicant used large truck incident data from the DOT Federal Motor Carrier Safety Administration (FMCSA). The FMCSA tracks fatal, injury, and property-damage-only incidents involving large trucks (both single-unit and combination trucks). In addition, the FMCSA tracks the release of hazardous cargoes in large truck crashes. The FMCSA defines a single-unit truck as a medium or heavy truck with engine, cab, drive train, and cargo area in one chassis. A combination truck is a truck tractor pulling any number of trailers. The staff finds the selection of the PHMSA Hazmat Incident Database and information on large truck crashes from the FMCSA is appropriate to assess the potential hazard from an accidental release of hazardous materials near the proposed site.

The applicant stated, based on information from the FMCSA, that large trucks in the United States traveled 1,432,880 million kilometers (890,351 million miles) from fiscal years 2016 through 2018. In Holtec Report No. HI-2210620, the applicant also stated, using the FMCSA data on large trucks from fiscal years 2016 through 2018, that on average 1.64 large truck incidents happened per million miles of travel. The applicant used the incident information from fiscal years 2016 through 2018 because the National Highway Traffic Safety Administration changed the crash reporting system. The applicant considered both single and combination (two or more trailers) trucks involved in fatal, injury, or property-damage-only crashes in fiscal years 2016 through 2018 to determine this incident rate.

The staff reviewed the estimated incident rate for large trucks in the United States. The data used to determine this incident rate are from the FMCSA. This government agency is entrusted with collecting and disseminating incident statistics for large trucks. Therefore, the staff finds it acceptable to derive an incident rate using data collected by the FMCSA. The staff independently verified the estimated incident rate for large trucks in the United States from 1998 through 2018. The FMCSA has cautioned against combining incident information in the years before and after 2016 because of the change in the reporting system. As the staff is using the incident information of all crashes (fatal, injury, and property damage) in a given year and the estimated incident rate will be used to compare with the applicant's derived rate, the staff extended the reporting period. During 1998 through 2018, large trucks (both single unit and combination) traveled 8,630,411 million km (5,362,689 million miles) and had 8,505,477 crashes (all fatal, injury, and property damage only). Therefore, the incident rate for large trucks in the United States during these 21 years (1998 through 2018) would be 1.59 crashes per million miles traveled. The incident rate for 5 years (2014 through 2018) is 1.61 crashes per million miles for large trucks. These incident rates compare quite well with the incident rate derived by the applicant. Based on the preceding discussion, the staff accepts the incident rate of 1.64 crashes per million miles for large trucks as appropriate to use in the hazard analysis.

Holtec Report No. HI-2210620 assumes that 16 km (10 mi) of U.S. Highway 62/180 are within 1.6 km (1 mi) of the proposed site location such that a hazardous material release could possibly affect safe operation at the proposed facility. Based on the map shown in figure 5-1, "Highways near facility," of Holtec Report No. HI-2210620, the staff finds a 16 km (10 mi) length of the highway would be acceptable and conservative.

The applicant estimated the annual frequency of large truck accidents in the 16 km (10 mi) length of highway near the proposed site to be 1.64×10^{-5} accidents. The applicant presented the statistics for large truck incidents with hazardous cargoes from the FMCSA for fiscal years 2016 through 2018 in attachment A of Holtec Report No. HI-2210620. Based on the information presented in attachment A, 11,652 (0.796 percent) large trucks involved in crashes in this period of 3 years were carrying hazardous cargoes in the United States. The applicant estimated that the annual frequency of large trucks involved in crashes within the 16 km (10 mi) stretch near the proposed site carrying hazardous cargoes would be 1.31×10^{-7} (= $1.64 \times 10^{-5} \times 0.796$) incidents.

The applicant stated that the probability that the wind is blowing toward the proposed site from U.S. Highway 62/180 is approximately 8 percent of the time by combining the values for southwest to northwest, south-southeast to north-northwest, and east-southeast to west-northwest, as shown in the wind rose in figure 5-2, "Lea County Regional Airport Station All Wind Rose (12/01/1948 – 12/31/2014)," of Holtec Report No. HI-2210620. In addition, as chlorine is released in only 0.0077 percent of the incidents, the applicant estimated that the frequency of released chlorine blowing toward the proposed facility in a release from a large truck involved in an incident on U.S. Highway 62/180 would be 8.08×10^{-13} (= $1.31 \times 10^{-7} \times 0.000077 \times 0.08$) per year.

Based on figure 5-2 of Holtec Report No. HI-2210620 and SAR table 2.3.2, "Lea County Regional Airport Station All Wind Data (12/01/1948–12/31/2014)," the staff finds that the proportion of time wind blowing toward the proposed site from the highway could be significantly more than the 8 percent value cited by the applicant. Based on SAR table 2.3.2, the staff estimates that the proportion of time that wind would blow toward the site to be approximately 46 percent, by combining values in southeast to northwest, south-southeast to north-northwest, south to north, south-southwest to north-northeast, and southwest to northeast. This increase in the proportion of the time would increase the estimated annual frequency by eight times. However, based on the preceding discussion, the staff concludes that a potential release of chlorine from a large truck crash near the proposed site would nevertheless not be a credible hazard to the proposed facility since the annual frequency is too low (significantly lower than the threshold of 1×10^{-6} per year, which is based on the Commission's Staff Requirements Memorandum (SRM) for SECY-01-0180, dated November 14, 2001).

In summary, the staff finds that the applicant collected and assessed large truck incident data in the United States and in New Mexico, where available, from authoritative sources. The applicant assumed that incidents involving large trucks would occur at the same rate near the proposed site on U.S. Highway 62/180, which is a divided highway. The applicant estimated the annual frequency of chlorine blowing toward the proposed facility from a release while being transported on this highway. The assessment is based on the proposed design of the facility and the estimated incident rate. Therefore, the staff finds that the requirements of 10 CFR 72.90(a)–(c) have been satisfied. The applicant collected incident information of large trucks carrying hazardous chemicals from reliable and authoritative sources and evaluated the hazard to the proposed facility using appropriate methods. Therefore, the staff finds that the requirements of 10 CFR 72.94(a)–(c) have been satisfied. The applicant also identified the region in which a hazardous material release from a large truck on U.S. Highway 62/180 would reasonably be able to affect the proposed facility. Therefore, the staff finds that the requirements of 10 CFR 72.98(a) have been satisfied.

2.3.2.1.3 Nearby Railway Transport of Hazardous Materials

The applicant assessed the potential hazards to the proposed facility from hazardous materials being transported on nearby railroads. The assessment is described in SAR section 2.2.4 and is documented in a 2022 railway hazardous chemical risk evaluation report, Holtec Report No. HI-2210619, "HI-STORE Railway Hazardous Chemicals Risk Evaluation" (proprietary), Revision 2, dated March 17, 2022. The staff reviewed the assessment, including the responses to RAIs 2-15 and 2-15-S, dated September 16, 2020, and April 15, 2022, respectively, pertinent literature, and databases maintained by different governmental agencies.

The nearest operating railroad, the BNSF Railroad, as shown in attachment B of Holtec Report No. HI-2210619, serves the local potash mines to transport potash ore to the processing plant(s). This industrial railroad is approximately 6 km (3.8 mi) west of the proposed site and runs from the Intrepid Potash North and the Intrepid Potash South sites to Carlsbad, New Mexico. The applicant assumed 16 km (10 mi) in railroad length for conservatism. Figure 5-1, "Railways Near the HI-STORE CIS Facility," of Holtec Report No. HI-2210619 shows the actual length of railway line running mostly north to south near the proposed site to be shorter than the applicant's 16 km (10 mi) assumption. The staff finds this conservative assumption acceptable. Additionally, the applicant did not consider a potential incident involving rail cars on the spur line dedicated to the proposed facility, as the rail cars on this spur would not be transporting hazardous chemicals, as stated in section 1.0, "Introduction," of Holtec Report No. HI-2210619. The staff accepts this assumption, as the applicant would have control of the materials being transported on railcars on this spur line.

Holtec Report No. HI-2210619 states that, although potash is mainly transported on this railroad, a wide range of hazardous materials may be transported, including, but not limited to, gasoline, diesel fuel, acids, carbon dioxide, nitrogen, liquid nitrogen, chlorine gas, refrigerants, fuel gases, oxygen, and explosives.

In this assessment, the applicant considered all incidents with rail transport and the release of hazardous materials from the database maintained by the DOT Federal Railroad Administration (FRA). The applicant stated, in its April 15, 2022, response to RAI 2-15-S, that the FRA defines a train accident as “a safety-related event involving on-track rail equipment (both standing and moving), causing monetary damage to the rail equipment and track above a prescribed amount.” Additionally, the FRA defines a train incident as “any impact between a rail and highway user (both motor vehicles and other users) of the crossings at a designated crossing site, including walkways, sidewalks, etc., associated with the crossing.” As a transportation-related incident includes railway accidents and is defined as “any unintentional release of a hazardous material during transportation, loading or unloading, or temporary storage related to transportation, and is not reliant on a dollar threshold” value, the applicant has used the incident data from the FRA database in this assessment. The staff agrees with this assumption, as the incident data of a particular time period include all accident data during the same period.

The applicant assessed the potential for a railway incident involving the release of hazardous materials with very low IDLH limits that could affect the operations at the proposed facility. The applicant selected chlorine to be the most hazardous chemical because its IDLH concentration is 10 parts per million. The staff agrees with the selection, as other hazardous chemicals have higher IDLH concentrations.

In Holtec Report No. HI-2210619, the applicant stated that chlorine is generally transported in tanker trucks as “bulk packaged” shipment for rail transport. Railroad tank cars have a typical capacity between 45,461 and 156,840 L (10,000 and 34,500 gal). The regulation 49 CFR 107.1, “Definitions,” defines an incident as “an event resulting in the unintended and unanticipated release of a hazardous material.” This includes the release of reportable quantities of hazardous materials during transport.

The applicant used the Hazmat Incident Database maintained by the DOT PHMSA to assess the type of hazardous chemical released in an incident. The database shows that 6,539 railway incidents in which hazardous materials were released took place between January 1, 2010, and January 1, 2021 (a period of 11 years). Of these, approximately 80 percent (5,257) of incidents involved 40 hazardous materials listed in table 5-1, “Railroad Incident Hazmat Release Report Data for US,” of Holtec Report No. HI-2210619. The chemical most frequently released in a rail incident is alcohol, not otherwise specified (PHMSA Id. No. UN1987), while chlorine (PHMSA Id. No. UN1017) was released in 38 incidents, that is, in 38 out of 5,257, or 0.58 percent of the incidents. Hydrochloric acid (PHMSA Id. No. UN1789) was released in 4.8 percent of the incidents, and sulfuric acid was released in 3.1 percent of the incidents.

The applicant stated that, based on the FRA database shown in attachment A of Holtec Report No. HI-2210619, 114,064 railway incidents took place between January 1, 2011, and January 1, 2021 (a period of 10 years). During this period, 7,056,342,933 miles were traveled by trains, resulting in an average rate of 16.16 incidents per million miles of train travel over the same 10-year period. As the railroad near the proposed site has been assumed to be 16 km (10 mi) long, and hazardous materials were released in only 173 incidents over the same period, the applicant estimated the annual frequency of a hazardous materials release near the proposed site to be 2.45×10^{-7} incidents (= $1.61\times10^{-4} \times 173/114{,}064$). In addition, the applicant estimated that the wind will blow unfavorably from the railroad in the west toward the proposed facility approximately 15 percent of the time. Consequently, the applicant estimated that the annual frequency of any hazardous material release incident near the proposed facility with unfavorable winds would be 3.68×10^{-8}.

As chlorine (PHMSA Id. No. UN1017) was released 0.58 percent of the time, the applicant estimated that the annual frequency of a potential release incident near the proposed facility would be 2.14×10^{-10}. Similarly, the applicant estimated the annual frequency of a release incident of sulfuric acid with more than 51 percent acid (PHMSA Id. No. UN1830) would be 1.11×10^{-8}.

The staff reviewed the estimated incident rate for rail accidents/incidents in the United States using the data from the FRA. This governmental agency is entrusted with collecting and disseminating incident statistics of railway accidents/incidents. Therefore, the staff finds it is appropriate to derive an accident/incident rate using data collected by the FRA. Based on the FRA, the term accident/incident is used to describe the entire list of reportable events. The term includes both accidents and incidents, and the FRA reports the accident/incident information in the database (e.g., FRA 1.12 Accident/Incident Overview), as presented in attachment A of Holtec Report No. HI-2210619.

Based on figure 5-1 of Holtec Report No. HI-2210619, the staff finds that unfavorable wind conditions would be when wind is blowing from the railroads toward the proposed facility. The unfavorable wind directions could include wind blowing from west to east, west of southwest to east of northeast, southwest to northeast, and south of southwest to north of northeast. Based on SAR table 2.3.2, wind could be blowing in unfavorable directions approximately 25 percent of the time. Based on this information, the staff estimates the annual frequency of release of hazardous materials near the proposed facility with unfavorable winds would be 6.1×10^{-8}. As chlorine is released in only 0.58 percent of all hazardous material releases in calendar years 2011 through 2021, the staff estimates that the annual frequency of chlorine release near the proposed site from an accident/incident on nearby railroads would be 3.5×10^{-10}. Similarly, the annual frequency would be 1.8×10^{-8} accidents/incidents of sulfuric acid with more than 51 percent acid and 2.9×10^{-9} accidents/incidents of hydrochloric acid based on 312 accidents/incidents that released hydrochloric acid out of 6,539 accidents/incidents releasing hazardous materials, as described in table 5-1 of Holtec Report No. HI-2210619. The staff finds that the annual frequency of releases of chlorine and two acids are roughly two to four orders of magnitude lower than 1×10^{-6} and, therefore, the staff concludes that a potential release of hazardous materials from a rail car due to an accident/incident would not pose a credible hazard to the proposed facility.

The applicant used the data from the PHMSA for a period of 11 years (calendar years 2011 through 2020) for estimating the probability of a specific chemical release, given a release has occurred, and used data from calendar years 2011 through 2020 (a 10-year period) to estimate the annual release frequency of a specific chemical. The staff independently checked the annual release frequency using PHMSA data for the same 10-year period. The final conclusions remain unchanged.

Based on the review of the applicant's assessment of the potential hazards from an accidental release of hazardous materials while the materials are being transported on rail cars near the proposed site, the staff finds the hazard assessment is acceptable because the applicant used appropriate data from the FRA and an appropriate statistical method to determine that the hazard would not be a design-basis event for the design and operation of the proposed facility. The staff finds that the applicant has thoroughly investigated the hazard, and the assessment shows that the potential hazard is not a credible hazard to the proposed facility. The staff notes that the railroad near the proposed site is used mainly to transport potash ore to the processing plants. The applicant assumed that hazardous materials would be transported in a similar proportion to that being transported nationwide or in New Mexico. It is unlikely that hazardous materials would be transported on this rail line and in the same proportion of total cargo. Therefore, the staff finds this a conservative assumption.

In summary, the staff finds that the applicant has appropriately assessed the potential release of hazardous chemicals while the materials are being transported on rail cars near the proposed site, and the associated consequences. The assessment is based on nearby railway lines. Therefore, the staff finds that the requirements of 10 CFR 72.90 (a)–(c) have been satisfied. The applicant examined the annual frequency of a potential release of hazardous materials near the proposed site using the information in the FRA database for the entire country and for the State of New Mexico. The applicant used appropriate data from the authoritative source and an appropriate statistical analysis method to determine the annual release frequency of each potential hazardous chemical near the proposed site. Therefore, the staff finds that the requirements of 10 CFR 72.94(a)–(c) have been satisfied. The applicant appropriately identified the region surrounding the proposed site where a potential release of hazardous chemicals while the chemicals being transported on railcars may affect safe operations at the proposed facility. Therefore, the staff finds that the requirements of 10 CFR 72.98(a) have been satisfied.

2.3.2.2 Nearby Pipeline Hazards

The applicant assessed the hazards to the proposed facility from a rupture of one or more pipelines near the proposed site carrying natural gas, and the associated hazards. The assessment is documented in the HI-STORE CIS Facility gas pipeline risk evaluation, Holtec Report No. HI-2210487, and summarized in SAR section 2.2.2. The NRC staff reviewed the assessment described in Holtec Report No. HI-2210487. In addition, the staff reviewed the applicant's responses to RAIs 2-6, 2-7, 2-7-S, and 2-7-S-1, dated September 16, 2020, August 16, 2021, and April 15, 2022; pertinent literature on the rupture of natural gas pipelines; and associated fire and explosion hazards. The staff also reviewed other licensing proceedings dealing with similar hazards.

SAR section 2.2.2 and Holtec Report No. HI-2210487 identified three natural gas pipelines currently near the proposed site:

(1) A DCP Midstream 51-centimeter (cm) (20-inch)-diameter pipeline is located approximately 0.25 km (0.16 mi) east of the eastern boundary of the proposed facility and carries natural gas at a pressure of 4.7 megapascals (MPa) (680 pounds per square inch gauge (psig)), although the pipeline is rated at a pressure of 8.2 MPa (1,180 psig).

(2) A second DCP natural gas pipeline with a 20 cm (8-inch) diameter is located 0.25 km (0.16 mi) east of the eastern boundary of the proposed facility and operates at 0.4 MPa (60 psig).

(3) A Lucid Energy 15 cm (6 inch)-diameter natural gas pipeline is located approximately 0.25 km (0.16 mi) east of the eastern boundary of the proposed facility and then turns west and runs approximately 12 m (40 feet) south of the southern edge of the proposed facility. The pipeline ends at the well pad of the currently operating well west of the proposed facility. It operates at 0.42 MPa (60 psig).

In addition, the applicant identified a planned Transwestern 51 cm (20-inch)-diameter natural gas pipeline that would be located approximately 1.3 km (0.8 mi) from the western boundary of the proposed site. This pipeline would normally be operated at 4.7 MPa (680 psig), although it would be rated at 8.2 MPa (1,180 psig).

Figure 1, "Lucid 6" Pipeline Including Location of Isolation Valves," of Holtec Report No. HI-2210487 shows the locations of the isolation valves of the 15 cm (6-inch)-diameter Lucid Energy pipeline near the proposed site. Based on this figure, one isolation valve near the proposed site would be at the well pad of Hanson State #001 gas and distillate well near the southwestern corner of the proposed facility in Section 13. The other isolation valve is located just north of the proposed facility.

Figure 2, "DCP Midwestern 20" and 8" Pipelines Showing Location of Isolation Valves," of Holtec Report No. HI-2210487 shows the two isolation valves of the DCP Midstream 51 cm (20-inch) and 20 cm (8-inch) pipelines nearest the proposed site. The nearest isolation valves for the 51 cm (20-inch) pipeline are located on Route 243 south of U.S. Highway 62/180 and at a significant distance northwest of the proposed site. The isolation valves for the 15 cm (6-inch) DCP Midstream pipelines are further south and northwest of the proposed site.

In section 3.0, "Risks Posed by Natural Gas Releases," of Holtec Report No. HI-2210487, the applicant stated that ignition of a flammable vapor cloud from a natural gas pipeline rupture might result in the following hazards, which are discussed below:

(1) missile generation from rupture of a pipeline

(2) a cloud or flash fire, or a fireball, if the ignition is delayed

(3) a jet fire

(4) a vapor cloud explosion, either from detonation or deflagration transitioning to detonation in favorable site conditions

The staff agrees with the possible events that may take place after ignition of the released natural gas from a ruptured pipeline, as they are the potential events widely accepted in the industry (e.g., Stephens, 2000; McGillivray, 2018; Sluder et al., 2021).

(1) Missile Generation

In section 3.0 of Holtec Report No. HI-2210487, the applicant discussed the potential for large pipeline fragments to be thrown a considerable distance and to act as missiles impacting the safety-related structures. For example, three pieces of 8 m (26-foot)-long pipeline segments were thrown 40 to 110 m (130 to 360 feet) in a 1965 incident in Natchitoches, Louisiana. Pipeline fragments, rocks, and debris were thrown more than 244 m (800 feet) in a pipeline rupture event in New Jersey. Based on these historical records and observations from other similar events in the PHMSA database, the applicant concluded that a potential rupture of the 51 cm (20-inch) pipeline would not endanger the critical areas of the proposed facility, as shown in figure 4, "DCP Midwestern 20" and 8" Pipelines Showing Distances to Critical Facilities," of

Holtec Report No. HI-2210487, as the pipelines are too far from the critical areas of the proposed facility.

The staff reviewed the information presented by the applicant on potential missiles that may be generated in a pipeline rupture at the HI-STORE CIS Facility. In addition, the staff reviewed pertinent literature on missiles generated in pipeline rupture events. The staff concludes that the potential consequences of missiles generated by the rupture of the 51 cm (20-inch) pipeline as bounding because this pipeline not only has the largest diameter but also the highest pressure among the current pipelines. The future pipeline will be located too far away to have any detrimental consequences from a potential missile strike. The applicant also cited several examples of the rupture of a high-pressure gas pipeline in which a pipeline fragment has been ejected a significant distance away. However, in those cases, the applicant did not present the potential impact radius (PIR) distance, as defined in 49 CFR 192.903, "What definitions apply to this subpart?" Sluder et al. (2021) discussed the rupture of a 76 cm (30-inch) natural gas transmission pipeline in Carlsbad, New Mexico. The largest piece of the pipeline resulting from the rupture was 7.9 m (26 feet) in length and was ejected 71 m (234 feet) from the rupture crater, approximately 48 percent of the PIR. Stephens (2000) reviewed several pipeline accidents in the National Transportation Safety Board (NTSB) database and concluded that damage is generally confined to the PIR. The PIR of the 51 cm (20-inch) pipeline is 144 m (474 feet) and of the 15 cm (6-inch) pipeline is 10 m (32 feet). The staff finds that the safety-related structures and systems at the proposed facility would be a significant distance away (at least three times the PIR) from the nearest pipelines, so it is unlikely that any potential missile generated in rupture of a pipeline would be able to damage these structures and systems.

Additionally, the proposed facility will be located in tornado Region III, as specified in figure 1, "Tornado intensity regions for the contiguous United States for exceedance probabilities of 10^{-7} per year," of NRC Regulatory Guide (RG) 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007. SAR table 2.7.2, "Tornado Generated Missiles," indicates that the cask transfer building (CTB) is designed for the design-basis tornado missiles appropriate for tornado Region II, as specified in table 2, "Design-Basis Tornado Missile Spectrum and Maximum Horizontal Speeds," of RG 1.76. In addition, the HI-STORM UMAX system and the vertical cask transporter and HI-PORT conveyance are designed against the tornado missile spectrum, as stated in SAR table 8.3.1, "Comparison of Accident Events between the HI-STORM MAX FSAR and the HI-STORE SAR." Backfill soil used in pipeline installation, which may also contain sand, gravel, and rocks, could be ejected from the crater, in addition to fragments of the ruptured pipe. Sluder et al. (2021) has concluded that the tornado missile spectrum specified in RG 1.76 generally bounds the missiles that could be potentially generated by a ruptured pipeline. Specifically, the airborne rock particles, used in construction of a pipeline, would have characteristics similar to the design-basis missile represented by a 2.54 cm (1-inch)-diameter steel sphere traveling at 6 m per second (m/s) (13 mph). In addition, it is very unlikely that damage to the safety-related structures from a fractured section of the 51 cm (20-inch) pipeline would be more severe than the damage from the design-basis tornado missiles represented by a 4.58 m (15-foot)-long, 0.168 m (6.625-inch)-diameter Schedule 40 steel pipe weighing 130 kilograms (kg) (287 pounds (lb)) and a 1,178 kg (2,593 lb) automobile. Both missiles are assumed to travel at 24 m/s (54 mph), as in RG 1.76. Consequently, based on the preceding discussion, the staff agrees that it is highly unlikely that potential missiles generated in a rupture of the 51 cm (20-inch) pipeline would damage the critical areas of the proposed facility. Additionally, the potential damage would be significantly less severe from fragments generated from the rupture of the 20 cm (8-inch) and the 15 cm (6-inch) diameter pipelines with only 0.42 MPa (60 psig) operating pressure. Therefore, the NRC staff agrees with the assessment that the potential damages from missiles generated from the rupture of a natural gas transmission pipeline would not be a potential hazard because of the design features and the intervening distance of the safety-related structures at the proposed facility.

(2) Flash Fire or Cloud Fire or Fireball

Holtec Report No. HI-2210487 states that a vapor cloud fire (which includes both flash fire and cloud fire) would be of very short duration, a few tens of seconds only. Consequently, the thermal radiation experienced by an object near a flash fire would be bounded by the jet fire, which is generally of significantly longer duration. In addition, the buoyant nature of the natural gas precludes formation of a persistent flammable vapor cloud at ground level. The applicant, therefore, concluded that vapor cloud fires pose little hazard to the proposed facility and need not be considered further. Fireballs radiate intense heat in a very short time; however, the overpressure associated with fireballs is negligible. The applicant did not assess the effects of a fireball, as they are generally associated with the release of pressurized liquids rather than compressed gas.

The staff reviewed the applicant’s assessment of a flash or a cloud fire resulting from a pipeline rupture or a potential fireball. In addition, the staff reviewed the pertinent literature on flash and cloud fires and fireballs from natural gas releases. The staff finds that the quantity of natural gas involved in a flash fire is dependent on the time of ignition of the plume after rupture of the pipeline carrying natural gas. Sluder et al. (2021) has reported that 74 percent of ignitions of released natural gas plumes occur within 2 minutes of pipe rupture. As the time required to close the isolation valves of the 51 cm (20-inch) pipeline would be 1 hour after a major rupture, as stated in section 8.1.1, “Modeling Assumptions and Conservatisms,” of Holtec Report No. HI-2210487, the ignition of the natural gas released from a rupture of a pipeline near the proposed facility would not depend on the isolation valves’ closure time. In addition, the radiant heat emitted by a flash fire would be intense but short lived (duration of a few seconds). The safety-related structures at the proposed facility are not expected to be damaged from an exposure of such short duration. Additionally, the staff agrees that the fireballs are typically associated with bursting vessels containing pressurized liquids. Therefore, the staff finds that the effects of short-duration fireballs would be bounded by jet fires of much longer duration.

(3) Jet Fire

A jet fire can pose a hazard to the safety-related structures, systems, and components (SSCs) because of intense thermal radiation. Holtec Report No. HI-2210487 assesses the potential consequences of a jet fire that may result from the rupture of a natural gas transmission pipeline near the proposed facility using the BREEZE Incident Analyst 4.0 software. BREEZE Incident Analyst uses the jet flame model developed by the Gas Research Institute and is used widely in the industry. The applicant considered the 51 cm (20-inch) and 15 cm (6-inch) pipelines for this assessment. The staff agrees with the selection, as these pipelines are closer to the proposed facility boundaries. Therefore, potential thermal flux from a jet fire that resulted from the rupture of these pipelines would bound the thermal flux of the 20 cm (8-inch) pipeline. In addition, the staff finds that the 51 cm (20-inch)-diameter pipeline is closest to the boundary of the proposed facility. In addition to having the largest diameter, this is a high-pressure pipeline rated at 8.2 MPa (1,180 psig), although the normal operating pressure is approximately 4.7 MPa

(680 psig). The other two pipelines are further away from the proposed facility and operate at a lower pressure (0.4 MPa (60 psig)). Therefore, the staff finds that the thermal flux of the potential jet fire resulting from the rupture of the nearest 51 cm (20-inch) diameter pipeline would bound the thermal flux from either the 20 cm (8-inch) or the 15 cm (6-inch) pipelines, provided the crater formed in a rupture of the 51 cm (20-inch) pipeline does not overlap and rupture the other pipelines and, subsequently, release additional pressurized natural gas. In the section "Assessment of Domino Effects" below, the staff reviews the possibility of a single pipeline rupture causing additional ruptures in other pipelines and found that the rupture of one natural gas pipeline at the proposed site would not affect other nearby pipelines. Additionally, the selected rupture points of the pipelines are closest to the storage pads and the CTB, where the applicant has planned to locate the safety-related structures and systems of the proposed facility.

Immediate ignition of the released natural gas would result in a jet fire anchored at the ruptured pipeline. Pipe fragments or rock pieces rubbing together could ignite the released natural gas to initiate the jet fire. Additionally, delayed ignition of a vapor cloud may burn back to the ruptured pipeline and develop a jet fire. High heat flux from a jet fire presents a thermal hazard to exposed structures and systems. The applicant selected the threshold thermal radiation level of 12.6 kilowatts per square meter (kW/m^2) for exposure of safety-related structures and systems at the proposed facility. At this level of thermal radiation, plastic would melt and wood would ignite with a flame (Sluder et al., 2021). The staff finds that the PIR, as defined in 49 CFR 192.903, is based on this threshold value for protecting people and property from thermal radiation in a jet fire (Stephens, 2000). At this thermal radiation level, an exposed person would experience pain within 3 seconds and blister within 6 seconds. Reinforced concrete structures can withstand much longer duration and higher intensity thermal fluxes without sustaining any damage (NUREG/CR-3330, "Vulnerability of Nuclear Power Plant Structures to Large External Fires," 1983). Therefore, based on the preceding discussion, the staff agrees that the thermal radiation level of 12.6 kW/m^2 is a reasonable threshold for assessing thermal damage to structures and systems at the proposed facility.

The applicant assumed that the pipelines would suffer a guillotine rupture, which results in a double-ended release of natural gas (i.e., release from both ends of the rupture), with the resulting releases merging. The staff finds this assumption conservative as it ignores pipe ruptures with lesser opening sizes than the full-bore rupture considered in this assessment. In addition, the same flow rate from both ends of the ruptured pipeline is assumed. In reality, pressure and, consequently, the flow rate at the downstream side would continue to decrease as time progresses. This assumption of a high discharge rate extends the predicted damage contours further out from the rupture point.

To account for the double-sided flow, the pipeline diameter is assumed to be an equivalent diameter corresponding to double the cross section of the original pipeline (i.e., the original pipeline diameter is increased by a factor of $\sqrt{2}$ or 1.414). Following U.S. Environmental Protection Agency (EPA)-550-B-99-009, "Risk Management Program Guidance for Offsite Consequence Analysis," issued 2009, the applicant's gas pipeline risk evaluation assumes a wind speed of 1.5 m/s (3.3 mph) and an F stability class. The applicant assumed that the jet emerges from the ruptured pipeline section at an angle of 60 degrees to the horizontal and directed along with the wind toward the proposed facility. The staff finds this assumption to be conservative because higher radiant heat would be experienced in the direction of flame tilt. Additionally, the applicant assumed that the natural gas goes directly up in the air without any contact with the crater walls. Any contact with the crater walls is likely to reduce the momentum and associated jet height as well as the amount of material in the jet.

Using the BREEZE Incident Analyst software, the applicant calculated the thermal radiation intensity level from the point of rupture. Holtec Report No. HI-2210487, table 8-3, “Consequences of Jet Fire Scenarios,” shows the calculated distances at different thermal radiation levels. Based on the calculated distance to receive the thermal load associated with the damage threshold (12.6 kW/m^2) and the actual distance from the potential rupture locations of the pipelines, the applicant concluded, as stated in table 8-4, “Potential Damage at Closest Distances from Pipeline in the Event of Pipeline Rupture and Jet Fire,” of Holtec Report No. HI-2210487, that the potential jet fires resulting from pipeline ruptures would not affect safety-related structures and systems at the proposed facility.

The staff reviewed the results of the BREEZE Incident Analyst software analysis given in table 8-2, “Consequences of Exposure to Thermal Radiation,” and table 8-3 of Holtec Report No. HI-2210487 to determine whether safety-related structures and systems at the proposed facility would be impacted by the thermal loads from jet fires developed from the ruptured pipeline(s). The staff finds that the safety-related structures and systems at the proposed facility would be located a significant distance away from the postulated points of rupture of the pipelines. The expected heat exposure, as tabulated in table 8-3 of Holtec Report No. HI-2210487, would be too small to cause any thermal damage to the safety-related structures and systems at the proposed facility.

The staff also notes that the applicant made several conservative assumptions to estimate the thermal damage distance contours from the ruptured pipelines, as described in the preceding discussion. Consequently, the staff concludes that the potential jet fires from the nearby ruptured natural gas pipeline would not pose a credible hazard to the proposed facility.

(4) Vapor Cloud Explosion

For a vapor cloud explosion of the natural gas released from a pipeline rupture to occur, three preconditions need to be satisfied, according to “Guidelines for Vapor Cloud Explosion, Pressure Vessel Burst, BLEVE, and Flash Fire Hazards,” by the Center for Chemical Process Safety, issued in 2010, and Holtec Report No. HI-2210487:

(1) Flammable material must be released into a confined and congested area (area of high turbulence).

(2) Ignition must be sufficiently delayed to allow formation of the methane-air mixture with methane concentration within the flammable rage (i.e., between the lower flammable limit of 5 percent and the upper flammable limit of 15 percent of the natural gas).

(3) An ignition source of sufficient energy must be present to ignite the methane-air mixture.

A vapor cloud explosion may occur from both deflagration and detonation of the natural gas released in a rupture of the transmission pipelines. The flame propagates through the unburned methane-air mixture in a deflagration mode at a burning speed of less than the speed of sound. A detonation may result from ignition of a plume of natural gas if the ignition is strong enough to cause a detonation so that the flame propagation speed is the speed of sound. A deflagration event may transition into a detonation event if suitable confinement for the released natural gas is available and sufficient congestion (presence of turbulence) is present in the flame propagation path. A detonation event generates significant air overpressure. Air overpressure developed in a deflagration event is dependent on the rate of combustion of the natural gas-air mixture and other factors, and lower than that in a similar detonation event.

Methane is the main component of the natural gas. The natural gas plume is buoyant because methane has a lower molecular weight than either nitrogen or oxygen. The buoyant nature of the plume generally precludes a persistent flammable plume of natural gas at ground level. As the plume rises, the availability of structures capable of providing sufficient confinement becomes less likely. Lack of confining structures combining with the relatively low flame speed of methane makes detonation of unconfined natural gas extremely unlikely. The methane concentration in the plume rapidly dilutes below the lower flammable limit, and the buoyant plume dissipates into the surrounding atmosphere. If the ignition occurs immediately after the pipeline rupture, a fireball, or a jet/trench fire, would make detonation of the released natural gas very unlikely. If the ignition of the plume occurs late, a flash fire may occur that may revert to a jet or trench fire at the rupture location.

Upon rupture of the pipeline, the methane would be released as choked flow from the pipeline in a high-pressure momentum jet. This jet would entrain air, and the methane concentration would be diluted. Holtec Report No. HI-2210487 states that intense turbulent mixing and air entrainment would limit the flammable region of the gas plume within a horizontal distance of approximately 100 m (328 feet) from the rupture point. Results of a computational fluid dynamics analysis conducted by the Oak Ridge National Laboratory for a potential rupture for a 107 cm (42-inch) natural gas pipeline at the Indian Point Energy Center, as described in Sluder et al. (2021), show that the lower flammable limit of the natural gas plume would be close to the release point. Consequently, an ignition to initiate a vapor cloud explosion must occur close to the rupture point and, therefore, an ignition close to or inside the proposed facility boundaries could not generate a vapor cloud explosion due to the dispersion of natural gas in the air.

The NRC staff also notes that the availability of an ignition source with sufficient energy to initiate a direct detonation is highly unlikely, especially near the locations of the pipelines. The ignition energy needed to deflagrate a methane-air mixture is in the order of 1×10^{-4} joules (J); however, direct initiation of detonation requires approximately 1×10^{8} J of energy, an increase of 12 orders of magnitude, based on “Vapour Cloud Explosions,” by W.P.M. Mercx and A.C. van den Berg, published in 2005. Common ignition sources, such as sparks from ejected pipeline fragments or an electrical apparatus, hot steam lines, and moving parts in a machine, do not generally possess an energy concentration that would directly initiate a detonation event. Such concentrations of energy may be available from high explosives. Therefore, a direct detonation of natural gas vapor cloud even near the release point is an extremely unlikely phenomenon. A deflagration of the vapor cloud is more likely to occur with associated low air overpressure.

The applicant stated that there are no obstacles in the vicinity of the proposed site that would help in accumulating a sufficient quantity of flammable natural gas. The staff agrees with the assessment. Based on satellite images available on Google Maps, the staff notes that the proposed site and surrounding areas lack any structures that may confine the natural gas plume in sufficient volume. In addition, the path from the pipelines to the proposed facility does not have any features that would provide congestion of the natural gas plume necessary to accelerate the flame front to transition from deflagration to detonation. Due to buoyancy, the plume rises as it travels further away from the release point. Because of the buoyancy, the plume may be above the structures when it traverses the proposed facility, reducing the likelihood of finding confinement for a natural gas-air mixture. In addition, sufficient congestion may not be available at that elevation, an important precondition for initiation of a detonation. Therefore, due to a lack of sufficient confinement and congestion near and inside the proposed facility boundaries, any potential transition of a deflagration started near the release point to detonation would be extremely unlikely.

A review of major vapor cloud incidents in the world in “Review of Vapour Cloud Explosion Incidents,” by Atkinson et al., commissioned by the PHMSA and the United Kingdom Health and Safety Executive, issued in 2017, did not identify any historical records of vapor cloud explosions of liquefied natural gas or methane in open areas with sufficient severity to cause damage. The NRC contacted the PHMSA pipeline accident investigators in connection with potential rupture and associated consequences of natural gas pipelines near the Indian Point Energy Center nuclear power plants, as part of its preparation of the “Report of the U.S. Nuclear Regulatory Commission Expert Evaluation Team on Concerns Pertaining to Gas Transmission Lines Near the Indian Point Nuclear Power Plant,” dated April 8, 2020. The NRC noted that these accident investigators expressed a similar opinion to that in Atkinson et al. (2017), in that they were unaware of any large natural gas (methane) delayed vapor cloud explosions from the rupture of a pipeline. In addition, as stated in the NRC report regarding gas transmission lines near the Indian Point nuclear facilities, the NRC did not find “any record of dense methane gas clouds…igniting or exploding at a location away from the initial pipe rupture.”

Nevertheless, Holtec Report No. HI-2210487 considered a hypothetical detonation of the natural gas that may occur within the turbulent methane jet entrained with air and analyzed the resulting overpressure. Such a detonation, although extremely unlikely, may occur as a result of transitioning from a deflagration to a detonation event or a high-energy density source igniting the methane within the flammable range. The applicant assumed that the hypothetical detonation of the turbulent jet would be at the center of rupture, and the entire flammable methane mass would participate in the detonation. The staff finds the assumptions are reasonable and conservative. The applicant used the U.S. Army TNT Equivalent explosion model, as implemented in the BREEZE Incident Analyst, to determine the overpressure generated. The U.S. Army TNT Equivalent explosion model is consistent with RG 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,” Revision 3, issued November 2021, and is acceptable to use in estimating overpressure. The applicant’s analysis compared the distance at which an overpressure of 6.9 kPa (1 pound per square inch (psi)) occurs as calculated, using this model with the shortest distance between the 51 cm (20-inch) and 15 cm (6-inch) pipelines and the critical structures at the proposed HI-STORE CIS Facility to determine their safety in an explosion (detonation) of the natural gas jet. The applicant’s analysis used a yield factor of 5 percent, following table 1, “Assumption for Determining Mass of TNT,” of RG 1.91. The mass of methane within the flammable limits in the jet developed as a result of the pipeline rupture was calculated using the analytical solutions given in Lees (1996). Appendix A to Holtec Report No. HI-2210487 gives the results for both the 51 cm (20-inch) and 15 cm (6-inch) pipelines. Results of the estimated safe distance (the distance beyond which the overpressure falls below 6.9 kPa (1 psi) appear in Holtec Report No. HI-2210487, table 8-6, “Consequence of Postulated Vapor Cloud Explosions in Turbulent Jet on Pipeline,” and figure 5, “Consequences of a Hypothetical Vapor Cloud Explosion on the 20″ DCP Midwestern Pipeline and the 6″ Lucid Pipeline.” Based on the preceding discussion, the staff concludes that an accidental rupture of a natural gas transmission pipeline near the proposed facility would not pose a credible explosion event.

Assessment of Domino Effects

The applicant also assessed whether the crater developed from a rupture of the 51 cm (20-inch) pipeline would also rupture the nearest pipeline of 15 cm (6-inch) diameter at a distance of approximately 15 m (50 feet). The applicant used the Gasunie model described in “Underground parallel pipelines domino effect: An analysis based on pipeline crater models and historical accidents,” by Silva et al., published in 2016 in the *Journal of Loss Prevention in the Process Industries*. The staff finds that the Gasunie model is widely used and that this empirical model is appropriate to determine the crater width from a pipeline rupture. It uses empirical correlations to predict the dimensions of the crater developed from the rupture of a pipeline, taking into account characteristics of the soil and the pipeline.

In Holtec Report No. HI-2210487, the applicant used this model to estimate a crater radius at the surface from a rupture of the 51 cm (20-inch) pipeline of 7 m (23 feet). The applicant concluded that rupture of the largest diameter pipeline would not induce any rupture of the nearest pipeline. Consequently, no modification of the quantity of natural gas within the flammable range is necessary in calculations of the thermal flux from a jet fire and overpressure from the vapor cloud explosion. The staff finds that the applicant has used a common method to estimate the crater radius from a pipeline rupture. The estimated crater radius for the largest pipeline is less than half of the distance to the nearest pipeline. Consequently, the staff finds that a rupture of a natural gas pipeline at the proposed site would not affect other nearby pipelines. Therefore, the rupture of a single pipeline assumed in the preceding pipeline-related hazard analyses is appropriate.

Summary of Nearby Pipeline Hazards Analyses

The applicant assessed the potential hazards to the proposed HI-STORE CIS Facility from a rupture and subsequent release of natural gas of a nearby transmission pipeline. The staff reviewed the assessment and finds several conservative assumptions.

Instantaneous Guillotine Rupture: The applicant assumed an instantaneous guillotine rupture of the natural gas pipeline. This assumption is conservative because it ignores all possible pipeline ruptures with smaller size openings and assumes the maximum natural gas release rate in a choked pipe flow. In addition, the applicant assumed that a guillotine rupture allows two full-bore releases from the two pipe ends. Additionally, the applicant assumed the same flow rate from both ends of the break. The assumed total flow rate is double the flow rate from a side of the pipeline, assuming the flow from both ends combines into a single jet. In reality, the natural gas pressure and, consequently, the flow rate from the downstream side would be lower than that from the upstream side and would continue to fall as time passes after rupture. If the releasees from both ends do not combine into one resultant jet, the total flammable mass in the resultant jets would be lower than in the single jet assumed in the analysis (double the flow rate from individual ends). Because of these assumptions, the damage contours determined in the analysis extend further out than in reality.

Detonation of Natural Gas in Turbulent Jet Stream: The applicant assumed that the flammable portion of the natural gas in the turbulent flow may detonate and generate overpressure. The staff agrees with the applicant that it would be extremely unlikely to find sufficient confinement near the pipelines at the proposed site to initiate a detonation event. In addition, the buoyancy of the natural gas would make the natural gas plume float over the proposed facility, thereby precluding a confinement. Additionally, common ignition sources do not possess sufficient energy density to directly initiate a detonation event.

Released Jet Not Striking Crater Walls or Floor: Although many releases from buried pipelines would strike the walls or floor of the crater formed, reducing momentum, turbulence, and flame temperature so that the effects of a potential jet flame or hypothetical detonation would be reduced, the applicant assumed that the released jet does not strike the walls or floor of the crater. Therefore, the analysis results would be conservative.

Jet Angle of 60 Degrees Assumed: The applicant assumed that the natural gas jet is tilted 60 degrees from horizontal toward the proposed facility. In addition, the applicant assumed the wind is blowing toward the proposed facility when the pipeline rupture occurs. Consequently, the analysis predicts higher heat fluxes toward the proposed facility.

Peak Release Rate for Jet Flame and Turbulent Jet: The applicant assumed that the peak release rate would be maintained during the release event. In reality, the gas release rate from the ruptured pipe would fall rapidly, resulting in lower heat fluxes from the jet flame and a smaller mass of natural gas within the jet than predicted by the applicant's analysis. In addition, the mass of natural gas participating in the hypothetical detonation would be lower than that used in the analysis.

Pipeline Operating at Maximum Allowable Operating Pressure: The applicant assumed that the 51 cm (20-inch) pipeline is operating at its rated pressure of 8.2 MPa (1,180 psig) instead of the normal operating pressure of 0.4 MPa (680 psig) in the hazard assessment. Assuming a higher gas pressure produces a higher natural gas release rate and larger estimated damage potential.

Based on the preceding discussion, the staff concludes that the applicant has used well-established methods and practices to assess the potential rupture of nearby natural gas transmission pipelines. The staff also finds that the analyses presented show that the natural gas released in such a rupture would not become a credible hazard to the proposed facility.

The staff finds that the applicant has assessed the potential rupture of natural gas transmission pipelines near the proposed site to determine the consequences and severity of the hazards. The assessment is based on the proposed design of the facility and the pipelines nearby. Therefore, the staff finds that the requirements of 10 CFR 72.90(a)–(c) have been satisfied. The applicant has collected pertinent information on nearby natural gas transmission pipelines from reliable sources and evaluated it using appropriate methods. Therefore, the staff finds that the requirements of 10 CFR 72.94(a)–(c) have been satisfied. The pipeline hazard assessment is based on information from the region surrounding the proposed site that may affect safe operation. Therefore, the staff finds that the requirements of 10 CFR 72.98(a) have been satisfied.

2.3.2.3 Aircraft Hazards

The applicant assessed the potential crash hazard from aircraft flying in the vicinity of the proposed site and documented the assessment in SAR section 2.2.3, "Air Transportation," and the calculation package in Holtec Report No. HI-2188201, "HI-STORE CIS Aircraft Crash Assessment" (proprietary), dated May 5, 2021. The staff reviewed the information presented in the SAR, the calculation package, and the applicant's May 24, 2018, response to the staff's RAIs. The staff also reviewed pertinent information available in the literature. This staff review determined whether the hazards to the proposed spent fuel storage facility from aircraft flying nearby have been appropriately characterized and considered in the design of the proposed facility. The staff reviewed the applicant's aircraft crash hazard assessment in accordance with section 3.5.1.6, "Aircraft Hazards," of NUREG-0800, "Standard Review Plan for the Review of Safety Analyses Reports for Nuclear Power Plants: LWR Edition," issued March 2010. The staff accepts the methodology in NUREG-0800 as applicable for reviewing the aircraft crash hazard to the proposed facility. The staff also accepts U.S. Department of Energy (DOE)-STD-3014-2006, "DOE Standard: Accident Analysis for Aircraft Crash into Hazardous Facilities," issued May 2006, as an acceptable methodology to assess the aircraft crash hazard of a facility. In addition, the threshold frequency for a design-basis aircraft crash hazard is taken as 1×10^{-6} per year, based on the Commission's Staff Requirements Memorandum (SRM) for SECY-01-0180, dated November 14, 2001.

The airspace surrounding the proposed facility is unrestricted. Consequently, at any given time, commercial aircraft, general aviation aircraft, or military aircraft may be flying in the vicinity of the proposed site. Section 3.5.1.6 of NUREG-0800 provides three screening criteria that must be satisfied to conclude, by inspection, that the aircraft hazard at a facility or a site is less than 1×10^{-7} per year for accidents that could result in radiological consequences greater than those outlined in the exposure guidelines in 10 CFR Part 100, "Reactor Site Criteria."

The staff review indicates that the site of the proposed facility does not satisfy proximity criterion B, which states that the site should be at least 8 km (5 mi) from the edge of military training routes (MTRs), including low-level training routes, as MTR IR-128/IR-180 crosses over the site. Consequently, based on the review guidance of section 3.5.1.6 of NUREG-0800, a detailed analysis is needed to assess the aircraft crash hazards to the site, taking into consideration flight activities at nearby airports and airways.

In Holtec Report No. HI-2188201, the applicant documented a detailed analysis of aircraft crash hazards at the proposed facility. The applicant examined the flight activities in connection with potential hazards from the crash of a civilian or military aircraft flying in the vicinity of the proposed site. The activities examined include (1) aircraft taking off and landing at nearby airports, (2) aircraft flying in nearby Federal airways, (3) aircraft flying on nearby MTRs, and (4) aircraft flying in nearby holding patterns. The staff reviewed the applicant's data, information, and analyses, along with some of the referenced documents. In Holtec Report No. HI-2188201, table 5.1, "HI-STORE SSC Dimensions," the applicant identified three structures as critical areas to be considered in an assessment of radiological consequences from an aircraft crash and gave their characteristic dimensions. These structures are the (1) UMAX ISFSI storage pads, (2) CTB, and (3) security building.

2.3.2.3.1 Aircraft Taking Off and Landing at Nearby Airports

As discussed in Holtec Report No. HI-2188201, although several commercial and private airports are located near the proposed site, none are within 16 km (10 mi) of the site. The applicant evaluated the following six public airports within 160 km (100 mi) of the proposed site:

(1) Artesia Municipal Airport (ATS) at Artesia, New Mexico: Distance to the proposed site is 76 km (47 mi) with annual average operations of 14,050.

(2) Cavern City Air Terminal (CNM) at Carlsbad, New Mexico: Distance to the proposed site is 55 km (34 mi) with annual average operations of 6,900.

(3) Lea County Regional Airport (HOB) at Hobbs, New Mexico: Distance to the proposed site is 48 km (30 mi) with annual average operations of 12,745.

(4) Lea County Zip Franklin Memorial Airport (E06) at Lovington, New Mexico: Distance to the proposed site is 51 km (32 mi) with annual average operations of 2,200.

(5) Roswell International Airport (ROW) at Roswell, New Mexico: Distance to the proposed site is 109 km (68 mi) with annual average operations of 49,045.

(6) Midland International Air and Space Port Airport (MAF) at Midland, Texas: Distance to the proposed site is 158 km (98 mi) with annual average operations of 76,412.

The applicant used the Air Traffic Activity Data System of the Federal Aviation Administration (FAA) and GRC Inc.'s AirportIQ5010 (https://www.airportIQ5010.com) to collect the number of annual operations at these airports. The staff considers the use of these data sources acceptable because they are both authoritative.

Using the AirNav website (https://www.airnav.com), the staff found four other general aviation airports beyond 16 km (10 mi) but within 160 km (100 mi) of the proposed site:

(1) Seven Rivers (62NM) at Carlsbad, New Mexico: Distance to the proposed site is 69 km (43 mi) with annual average operations of 300 by general aviation aircraft.

(2) Lea County (Jal) Airport (E26) at Jal, New Mexico: Distance to the proposed site is 72 km (45 mi) with annual average operations of 3,000 by general aviation aircraft.

(3) Tatum Airport (18T) at Tatum, New Mexico: Distance to proposed site is 87 km (54 mi) with annual average operations of 500 by general aviation aircraft.

(4) Denver City Airport (E57) at Denver City, Texas: Distance to the proposed site is 93 km (58 mi) with annual average operations of 1,500 by general aviation aircraft.

2.3.2.3.2 Enroute Flights on Federal Victor Airways and High-Altitude Routes

The applicant used the Albuquerque visual flight rule (VFR) sectional chart and the high-altitude enroute chart H-6 from the FAA to identify Federal Victor Airways and high-altitude airways near the proposed site, as given in table 5.3, "Nearby Federal Airways – Width and Distances," of Holtec Report No. HI-2188201. Aircraft fly the Victor airways using either instrument flight rule (IFR) or VFR while remaining below 5,500 m (18,000 feet) above mean sea level (AMSL). Aircraft fly high-altitude J routes at altitudes between 5,500 m (18,000 feet) AMSL and flight level 450 (13,500 m (45,000 feet) AMSL). Q routes are area navigation (RNAV) routes between 5,500 m (18,000 feet) AMSL and flight level 450. Aircraft with an appropriate global positioning system (GPS) can fly the J and RNAV routes.

In table 5.3 of Holtec Report No. HI-2188201, the applicant identified four Victor airways (V-68, V-83, V-102, and V-291), four high-altitude J routes (J-15, J-65, J-66, and J-108), and two Q routes (Q-20 and Q-37) near the proposed site using the Albuquerque VFR sectional chart. In table 5.4, "Usage of Federal Airways Near the CIS Facility," of Holtec Report No. HI-2188201, the applicant summarized information on usage of these airways near the proposed site that it received from the FAA for 2018 in terms of commercial, general, and military aviation. As discussed in section 2.3.2.3.6, "Crash Rate of Aircraft in Inflight Mode," of this SER, the applicant assumed that all military aircraft flying these Federal airways are large military aircraft.

For assessing the hazard to the proposed facility, the applicant selected four airways with the nearest edge less than 16 km (10 mi) from the proposed site:

(1) Victor route V-102: The centerline of this Victor route traverses approximately 10.5 km (6.5 mi) north of the proposed site. The width of the airway is 7.4 km (4.6 mi) on either side of the centerline. The nearest edge of this airway from the proposed site is 3.1 km (1.9 mi).

(2) Victor route V-291: Airway V-291 is a low-altitude Victor route. The centerline of this route crosses approximately 19 km (11.8 mi) north of the proposed site. The width of the airway would be 7.4 km (4.6 mi) on either side of the centerline. The nearest edge of this airway from the proposed site is 11.6 km (7.2 mi).

(3) Jet Route J-15: Jet route J-15 originates at Seattle-Tacoma International Airport and goes to Houston International Airport. The nearest edge of this route is 4.8 km (3 mi) from the proposed site.

(4) RNAV route Q-20: The nearest edge of this RNAV airway from the proposed site is 12.8 km (8.0 mi). Q-20 branches out of jet route J-15 near Albuquerque and joins with it near Junction, Texas.

The staff finds that the process of identifying Federal airways with a major contribution to the overall aircraft crash hazard to the proposed facility using information from the FAA is appropriate. Airways more than 16 km (10 mi) away would have negligible contribution to the overall aircraft crash hazard based on screening criteria given in NUREG-0800, section 3.5.1.6, “Aircraft Hazards).”

2.3.2.3.3 Nearby Holding Patterns and Missed Approaches

In Holtec Report No. HI-2188201, based on Jeppesen instrument approach charts and approach charts provided at https://airnav.com for nearby airports, the applicant identified two holding or missed approach patterns that come close to the proposed site:

(1) Holding pattern for Cavern City Air Terminal (CNM) runway RNAV (GPS) RWY 21. This holding pattern is 11 km (6 nautical mi or 6.8 mi) long and almost 22 km (14 mi) northeast of the airport. The missed approach pattern for Cavern City runway RNAV (GPS) RWY 3 matches this pattern. The proposed site is 19.8 km (12.3 mi) from the nearest edge of the pattern.

(2) Missed approach holding pattern for Lea County Regional (HOB) runway LOC RWY 3. This holding pattern is 11 km (6 nautical mi or 6.8 mi) long and approximately 31 km (19 mi) southwest of the airport. Missed approach pattern for Lea County Regional runway LOC BC RWY 21 and VOR or TACAN RWY 2 match this pattern. The nearest edge of this pattern is 5 km (3 mi) from the proposed site.

2.3.2.3.4 Military Aviation Along Nearby Military Training Routes

Holtec Report No. HI-2188201 identifies two MTRs using the FAA’s Albuquerque sectional chart near the proposed site: (1) IR-128/IR-180 and (2) IR-192/IR-194. Aircraft follow IFR while traversing these routes. The U.S. Department of Defense’s “Area Planning: Military Training Routes. North and South America. AP/1B,” issued 2018, gives the route fixes for each MTR as well as the route width. The route width varies between fixes.

In addition, a route can have different distances to the edges from the route "center." Based on "Area Planning: Military Training Routes. North and South America. AP/1B," the IR-128/IR-180 route is not symmetrical about the centerline near the proposed site. The airway extends 7.4 km (4 nautical mi) toward the proposed site and 5.6 km (3 nautical mi) away from the proposed site from its centerline. Similarly, the IR-192/IR-194 route also is not symmetrical near the proposed site. It extends approximately 5.6 to 7.4 km (3 to 4 nautical mi) from its centerline toward the proposed site and 13.0 km (7 nautical mi) from its centerline away from the proposed site.

Both IR-128/IR-180 routes are operated by Dyess Air Force Base (AFB). These routes pass directly above the proposed site. The centerline of the route runs less than 2.6 km (1.6 mi) east of the proposed site. The route is designated as IR-128 when the aircraft flies south to north and IR-180 when the aircraft flies north to south.

Both IR-192/IR-194 routes are operated by Holloman AFB. The distance of the nearest edge of these routes to the proposed site is 15.8 km (9.8 mi). The route is designated as IR-192 when the aircraft flies north to south and IR-194 when the aircraft flies south to north.

Based on the information obtained from Dyess AFB and Holloman AFB, the applicant provided the annual number of flights by each type of aircraft on IR-128/IR-180 and IR-192/IR-194 in table 5.7, "Military Training Route IR-128/180 Annual Flight Operations," and table 5.8, "Military Training Route IR-192/194 Annual Flight Operations," of Holtec Report No. HI-2188201, as well as additional details about aircraft traffic on those routes. Dyess AFB provided information on MTR IR-128/IR-180 from 2016 through 2019, and Holloman AFB provided traffic information on IR-192/IR-194 for 2019 only. Dyess AFB also stated that it does not fly the portion of IR-128/IR-180 near the proposed site. In addition, none of the aircraft using IR-128/IR-180 carry any weapons or external fuel tanks.

Tables 5.7 and 5.8 of Holtec Report No. HI-2188201 indicate that major portions of the sorties flown on IR-192/IR-194 are by F-16 aircraft, and the majority of the sorties on IR-128/IR-180 are flown by B-1B aircraft. In its analysis, the applicant assumed that all flights on IR-192/IR-194 are by F-16 aircraft and all flights on IR-128/IR-180 are by B-1B aircraft. Additionally, the applicant assumed that all aircraft flying these MTRs will be flying in special flight mode. The staff finds these assumptions acceptable, as only a few sorties on these MTRs are flown by aircraft other than either B-1Bs or F-16s. In addition, assuming the aircraft would be flying in special flight mode is conservative, as the crash rate in special flight mode for any aircraft is higher compared to that in normal flight mode.

2.3.2.3.5 Effective Area of the Proposed Facility

The applicant calculated the effective area of the proposed facility as the summation of the effective areas of the three structures discussed before (the UMAX ISFSI storage pads, the CTB, and the security building). The effective area of a facility is dependent not only on the facility dimensions (length, width, and height) but also on the type of aircraft and is the summation of the aircraft fly-in area and the skid area. The fly-in area is the sum of the facility footprint area and the facility shadow area for a particular type of aircraft. The applicant used the methodology given in DOE-STD-3014-2006 to estimate the effective area of the proposed facility for each type of commercial, general, and military aviation aircraft flying nearby. The characteristics of an aircraft (length of wingspan, cotangent of the aircraft impact angle, and skid distance) used in estimating the effective area are taken from tables B-16, B-17, and B-18 of DOE-STD-3014-2006 and, therefore, are acceptable. The estimated effective areas of the proposed facility for the types of aircraft flying nearby are given in table 6.5, “Federal Airway Effective Area Determination,” table 6.7, “IR-128/180 Probability Determination (B-1B Aircraft),” and table 6.8, “IR-192/194 Probability Determination (F-16 Aircraft),” of Holtec Report No. HI-2188201.

Holtec Report No. HI-2188201 implicitly assumes that the effective area of each structure is independent of other structures; that is, the effective areas of nearby structures do not overlap. This is a conservative assumption, as the effective area of one structure may partially overlap with the effective area of another structure, thereby making the actual effective area of the combined structures slightly smaller than estimated.

2.3.2.3.6 Crash Rate of Aircraft in Inflight Mode

The crash rate of an aircraft is dependent on the aircraft type and the phase of flight (taking off, inflight, and landing) of the aircraft.

<u>Commercial Aviation</u>

The applicant used an inflight crash rate for commercial aircraft flying on the Federal airways of 4×10^{-10} crashes per aircraft flight mile, as given in NUREG-0800, section 3.5.1.6. This crash rate for commercial aircraft is acceptable.

<u>General Aviation</u>

The applicant updated the crash rate of all general aviation types of aircraft of 8.21×10^{-5} crashes/flight hour using information from 1986 through 1993, as given in table 3.28, “General Aviation Accident Rates by Aircraft Subcategories,” of Kimura et al. (1996). The applicant reported an average crash rate of 6.285×10^{-5} crashes/flight hour based on all crashes of general aviation aircraft (both fatal and nonfatal crashes) in 2009 through 2018, as given in the NTSB database. This database shows a reduction of the general aviation crash rate in recent years by 23.4 percent of the crash rate given by Kimura et al. (1996). The applicant revised the inflight (enroute) crash rate of 1.55×10^{-7} crashes per flight mile, given in table 3.37, “General Aviation Total Powered Aircraft Crash Rates by Flight Phase,” of Kimura et al. (1996), in the same proportion to determine an updated general aviation crash rate of 1.19×10^{-7} crashes per flight mile. The staff accepts the revised inflight crash rate of general aviation aircraft as appropriate, as the NTSB data show a reduction in recent years (2009 through 2018) from that in Kimura et al. (1996).

In addition, it is expected that any general aviation crashes where the aircraft sustained only partial damage, or the pilot was only injured, would not have sufficient impact forces to cause any substantial damage to the reinforced concrete structures, leading to a radioactive release. Consequently, it can be argued that only those crashes of general aviation aircraft in which a fatality occurred might conceivably have sufficient energy to cause any significant damage to these structures (Kimura et al., 1996). The applicant used data on all crashes, both fatal and nonfatal, to determine the general aviation crash rate. Using the NTSB data, the fatal crash rate would be substantially lower, less than one-fifth of the rate for all crashes. Therefore, the staff concludes that the revised crash rate of general aviation aircraft used by the applicant is acceptable.

Military Aviation in Nearby Federal Airways

The applicant assumed that all military aircraft flying Federal airways near the proposed facility are large military aircraft operating in normal flight mode. Large military aircraft include bombers and cargo aircraft, typically multiengine aircraft (Kimura et al., 1996). The applicant used a crash rate of all large military aircraft of 1.90×10^{-9} crashes per flight mile, as given in table 4.8, "Crash Data and Estimates of Crash Frequencies Based on the Minuteman III Mishap Database," of Kimura et al. (1996). The staff finds that the assumption of large military aircraft in normal flight mode while transiting the Federal airways is reasonable because these aircraft are flying outside the military operations areas and low-level flight ranges and are not expected to conduct any special maneuvering in unrestricted airspace (Federal routes) shared by both commercial and general aviation aircraft. Additionally, as given in table 5.4 of Holtec Report No. HI-2188201, military aircraft are only a small fraction of all aircraft flying these airways.

Military Aircraft in Nearby Military Training Routes

The applicant took the crash rate of B-1B bombers while using the IR-128/IR-180 route as 1.20×10^{-8} crashes per mile from table 4.8 of Kimura et al (1996), assuming they fly in special flight mode. The staff finds the crash rate acceptable, as it is from an authoritative source. In addition, it omits the reduction in the crash rate observed in recent years. The crash rate of an aircraft in special flight mode is greater than that in normal flight mode (Kimura et al, 1996). Although Dyess AFB has informed the applicant that it does not fly B-1Bs near the proposed site, the applicant assumed that the B-1Bs fly in special inflight mode (i.e., B-1Bs conduct special maneuvering or low-level flight operations) while transiting this MTR. The staff finds that the applicant's assumption of special flight mode is conservative. Similarly, the applicant has assumed that F-16s fly in special flight mode while using IR-192/IR-194. The applicant took the special inflight crash rate of F-16s, as given in table 4.8 in Kimura et al. (1996), of 1.12×10^{-7} crashes per mile in assessing the crash hazard to the proposed facility. The staff finds the crash rate appropriate, as it came from an authoritative source and omits the reduction in the crash rate observed in recent years.

2.3.2.3.7 Annual Aircraft Crash Hazard to the Proposed Facility from Nearby Sources

The applicant analyzed the annual frequency of aircraft crash hazards from different sources present near the proposed facility.

Aircraft Taking off and Landing at Nearby Airports

As shown in table 6.1, "Screening Criteria Check for Airports," of Holtec Report No. HI-2188201, the number of annual operations (take offs and landings) at nearby airports (within 160 km (100 mi) of the proposed facility) are significantly lower than 1,000 times the distance (in miles) squared, D^2. Consequently, the applicant excluded aircraft operations at nearby airports from further consideration in assessing an aircraft crash hazard to the proposed facility. The staff agrees with the assessment that operations at these airports will have negligible effects on operations at the proposed facility because it is consistent with the proximity criteria and table showing fatal crash probabilities for general aviation in section 3.5.1.6 of NUREG-0800. The airports are far away from the proposed site, and the number of operations at these airports is far too low to pose any credible hazard to the safe operation of the proposed facility. In addition, an aircraft flying near the proposed site would be far away from the near-airport environment of these airports and will be in the inflight or enroute phase.

As discussed in SER section 2.3.2.3.1, the staff identified four other general aviation airports within 160 km (100 mi) of the proposed facility. However, because of the large distance from the proposed site and few annual operations at each of these airports (significantly less than 1,000 D^2), the staff concludes that operations at these four airports also do not pose any credible hazard to the proposed facility.

Flights Along Federal Airways

The applicant assumed all commercial flights near the proposed facility are by air carriers and has used the formula given in section III.2, "Airways," of section 3.5.1.6 of NUREG-0800 to determine the annual crash frequency of aircraft transiting a Federal airway near the proposed facility. Using this formula, the applicant determined the annual crash frequency of each type of aircraft (general, commercial, military aviation) flying one of these four airways. Table 6.6, "Federal Airway Probability Determination," of Holtec Report No. HI-2188201 gives the annual crash frequency of each airway. The total estimated crash frequency on Federal airways near the proposed facility is 4.6×10^{-7} per year, as given in the table.

The applicant used the method given in section 3.5.1.6 of NUREG-0800 to estimate the annual crash hazard to the proposed facility from aircraft flying in nearby airways. Information on usage of these airways came from the FAA's Mission Support Services. The applicant used DOE-STD-3014-2006 to calculate the effective areas of the proposed facility. Based on the preceding discussion, the staff concludes that the estimated annual crash frequency of 4.6×10^{-7} per year onto the proposed facility from aircraft in nearby Federal airways is appropriate.

Flights in Holding Patterns and Missed Approaches

The applicant concluded that the annual crash frequency of aircraft flying these approaches and patterns is negligible because these holding and missed approach patterns are more than 3.2 km (2 mi) from the proposed site. The staff agrees with the conclusion because the nearest edges of these holding patterns and missed approaches are a sufficient distance away from the proposed site, following section II, "Acceptance Criteria," of section 3.5.1.6 of NUREG-0800.

Flights Along IR-128/IR-180 and IR-192/IR-194

The applicant determined the effective area of the proposed facility for B-1B and F-16 aircraft flying the IR-128/IR-180 and IR-192/IR-194 MTRs, respectively, and presented them in tables 6.7 and 6.8, respectively, of Holtec Report No. HI-2188201. Using the flight information presented in table 5.6 for B-1B aircraft on IR-128/IR-180 in 2018 and in table 5.8 for F-16 aircraft on IR-192/IR-194 in 2019, the applicant estimated the annual crash frequency onto the proposed facility to be 2.48×10^{-8} per year for B-1B aircraft and 4.07×10^{-9} per year for F-16 aircraft and presented them in tables 6.7 and 6.8, respectively, of Holtec Report No. HI-2188201.

The staff reviewed the estimated crash frequencies of both B-1B and F-16 aircraft while flying IR-128/IR-180 and IR-192/IR-194 MTRs and finds that the applicant used a conservative number of flights and appropriate crash rate information in determining the crash frequencies. For example, the applicant used the flight information for IR-128/IR-180 for 2018. The staff finds, from table 5.7 of Holtec Report No. HI-2188201, that the maximum number of flights took place in 2018, based on information from Dyess AFB on flights using IR-128/IR-180 in 2016 through 2019. In addition, the applicant used MTR information from the U.S. Department of Defense's "Area Planning: Military Training Routes. North and South America. AP/1B." The crash rates of B-1Bs and F-16s are taken from table 4.8 in Kimura et al. (1996). Based on the preceding discussion, the staff finds that the estimated crash frequencies are acceptable.

The applicant also determined the annual frequency of onboard ordnance released in flight striking a safety-related structure at the proposed facility. Dyess AFB informed the applicant that flights using IR-128/IR-180 do not carry any ordnance on board. The applicant did not receive any such information from Holloman AFB for F-16 flights on IR-192/IR-194. Consequently, the applicant assumed that all F-16 flights on IR-192/IR-194 carry ordnance onboard. In addition, the applicant assumed that the onboard ordnance would not be armed while transiting IR-192/IR-194. The staff finds this a reasonable assumption, as the proposed facility does not lie beneath a military operations area and the flight path near the proposed facility does not lead to a nearby bombing run. Additionally, the applicant assumed that an F-16 will not carry an exceptionally large bomb on board. This is a reasonable assumption as F-16s are fighter aircraft.

The applicant also assumed that ordnance jettisoned from an aircraft would not skid on the ground before striking a structure, and the effective area of the proposed facility has been determined taking zero skid distance. The staff finds this is a reasonable assumption as the jettisoned ordnance would likely impact the ground at a very steep angle.

The applicant calculated the frequency of ordnance jettisoned from a F-16 flying the IR-192/IR-194 route striking the proposed facility to be 2.74×10^{-8} per year. The staff finds the estimated strike frequency is acceptable, as it has been determined using conservative assumptions. For example, all F-16 flights generally do not carry ordnance onboard. In addition, not all inflight emergencies experienced by an F-16 would lead to jettisoning the ordnance.

2.3.2.3.8 Cumulative Annual Frequency of Aircraft Crash Hazard

The applicant determined that the frequency of an aircraft crash onto the proposed facility while flying nearby is 5.15×10^{-7} per year by summing the annual crash frequency from individual sources. These sources include aircraft taking off and landing at nearby airports, flying on nearby Federal airways, flying on nearby holding patterns and missed approaches, and military flights through nearby MTRs IR-128/IR-180 and IR-192/IR-194. The estimated annual crash frequency also includes a potential strike by jettisoned ordnance from F-16s flying the route IR-192/IR194. Because the nearby airports are a significant distance away from the proposed site with a relatively small number of annual operations (landing and taking off), the contribution to the total facility hazard by these operations is negligible. Similarly, the holding patterns and missed approaches near the proposed site are still too far away to constitute an appreciable crash hazard to the proposed facility. The staff finds the estimated annual crash frequency to be reasonable.

2.3.2.3.9 Annual Frequency Acceptance Criterion for Aircraft Crash Hazard

Section 3.5.1.6 of NUREG-0800 provides the methodology to estimate the probability of aircraft crashing onto a nuclear power plant. An operating nuclear power plant requires active systems to control the dynamic nuclear and thermal processes that occur in the conversion of nuclear reactions into thermal power. In the event of a mishap, there are large amounts of thermal energy within the reactor core. Emergency cooling systems are provided as part of a reactor facility design to avoid core damage or meltdown and the release of radioactive material into the environment.

Hazards that can initiate onsite accidents leading to a loss of coolant at a reactor facility should have a sufficiently low probability of occurrence. NUREG-0800, section 2.2.3, "Evaluation of Potential Accidents," Revision 3, issued March 2007, states a frequency of occurrence of approximately 1×10^{-7} per year as the NRC staff objective, to screen out external events that may affect the nuclear reactor and have consequences for the safety of the facility and the potential for significant radiological impacts to public health and safety. However, data are often not available to permit an accurate estimation of the probabilities of occurrence of the postulated events. Accordingly, pursuant to NUREG-0800, a probability of occurrence of potential radiation exposures in excess of the 10 CFR Part 100 dose guidelines of approximately 1×10^{-6} per year is acceptable for a nuclear power plant, provided, when combined with qualitative arguments, the realistic probability can be shown to be lower. In the Policy Statement on Safety Goals, published in the *Federal Register* (51 FR 30028) on August 21, 1986, the NRC noted, "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation." This translates to a probability of occurrence of 1×10^{-6} per year.

Compared to a nuclear reactor facility, an ISFSI is a relatively passive system that does not have complex control requirements and has contents with relatively low thermal energy. Consequently, potential fuel damage and the associated radioactive source terms from a potential accident are significantly less than those expected from a nuclear reactor facility. As a result, the estimated consequences from a potential accident at an ISFSI are less severe than those for one at a nuclear reactor facility. The staff, therefore, concludes that a frequency of 1×10^{-6} crashes per year is an acceptable threshold frequency criterion for evaluating aircraft crash hazards at the proposed facility, consistent with the Commission's decision on SECY-01-0180 in SRM M011114A, dated November 14, 2001.

2.3.2.3.10 Summary of Aircraft Hazards Review

The applicant examined activities in connection with potential hazards from the crash of civilian and military aircraft flying in the vicinity of the proposed facility. The activities examined include aircraft taking off and landing at nearby airports; flying in nearby Federal airways, including missed approaches and holding patterns associated with nearby airports; and flying nearby MTRs while jettisoning ordnance.

The staff reviewed the scenarios, data, information, and analyses presented by the applicant in connection with the proposed facility. SER section 2.3.2.3.7 discusses the crash frequency estimates. These frequencies are estimated on the basis of several elements that determine the overall likelihood that each specific type of aircraft operation may lead to an impact at the proposed facility. Typically, these include measures that reflect traffic density (e.g., flights per year), crash rate (e.g., crashes per mile, crashes per unit area per unit time), and the effective target area. The applicant used information from authoritative sources on the crash rates of aircraft engaged in different types of operations near the proposed facility. Additionally, information on the number of flights involved in nearby airports and Federal airways are from the FAA. The Air Force has provided the usage information of nearby MTRs. The staff reviewed the FAA forecast of civil aviation growth for fiscal years 2021 through 2041 in "FAA Aerospace Forecasts Fiscal Years 2021–2041," issued 2022. Commercial and general aviation are expected to grow at a pace of 3.4 percent and 0.75 percent every year, respectively. As shown in table 6.6 of Holtec Report No. HI-2188201, civilian aircraft only become a potential aircraft crash hazard onto the proposed facility when flying the nearby Federal airways. Operations at the nearby (within 160 km (100 mi)) airports and missed approaches and holding patterns do not contribute significantly. Approximately 90 percent of the hazard to the proposed facility from flights on the nearby Federal airways is from general aviation aircraft. Therefore, based on the FAA projection, it is expected that at least until 2041 (the last year the current projection is available), the aircraft crash hazard at the proposed facility will increase minimally. Based on the estimated 5.15×10^{-7} crashes per year and the threshold criterion of 1×10^{-6} crashes per year, the staff concludes that the annual frequency of crashes for both civilian and military aircraft at the proposed facility is acceptable.

Based on the preceding discussion, the staff finds that the applicant has investigated and assessed the flights and flight-related operations near the proposed site to determine the frequency and severity of aircraft crash hazards. The assessment is based on the proposed design of the facility and aircraft operations at nearby airports and flights using the Federal airways and the MTRs. Therefore, the staff finds that the requirements of 10 CFR 72.90(a)–(c) have been satisfied. The applicant has collected pertinent information on aircraft-related activities from reliable and authoritative sources and evaluated them using appropriate methods. Therefore, the staff finds that the requirements of 10 CFR 72.94(a)–(c) have been satisfied. The aircraft hazard assessment is based on information from the potential region surrounding the proposed site on factors that may affect safe operation. Therefore, the staff finds that the requirements of 10 CFR 72.98(a) have been satisfied.

2.3.3 Meteorology

This section describes the staff's review of the information presented in SAR section 2.3, "Meteorology." The topics reviewed are regional climatology, local meteorology, and the onsite meteorological measurement program.

2.3.3.1 Regional Climatology

In SAR section 2.3.1, "Regional Climatology," the applicant described the climate as typically semiarid with mild temperatures, low humidity, and low precipitation with infrequent storms. The applicant described the winter season as typically affected by high-pressure systems located in the central part of the western United States and low-pressure systems located over north-central Mexico. For summer seasons, the applicant described the region as typically affected by low-pressure systems over Arizona. The NRC staff notes that seasonal changes in weather patterns associated with the changes in regional pressure systems alter the source and amount of moisture in the air either from the southeast or the west and subsequently alter the precipitation amount and pattern in eastern New Mexico. For example, thunderstorms during the rainy season from summer to early fall are an inherent part of the monsoonal climate of eastern New Mexico and are driven by moisture-laden winds from the Gulf of Mexico.

The applicant summarized regional meteorological information on temperature, winds, mixing heights, tornadoes, hurricanes, thunderstorms, and precipitation and snowfall from various sources but primarily the Western Regional Climate Center (WRCC). The WRCC is a collaboration of the Desert Research Institute and the National Oceanic and Atmospheric Administration (NOAA) under the auspices of NOAA's program to disseminate meteorological information through regional centers. The applicant selected the Lea County Regional Airport station, which is 37 km (23 mi) to the northeast of the proposed site, to represent regional and local site conditions. The applicant described average and severe weather based on meteorological inputs as follows:

- Temperature: The applicant summarized data for the period 1941–2016 from the Lea County Regional Airport meteorological station. SAR table 2.3.1 and figure 2.3.1, both entitled “Lea County Regional Airport Station Temperature Data (09/01/1941-06/09/2016),” include monthly and annual average values for the minimum, average, and maximum temperatures and monthly and annual extreme values for minimum and maximum temperatures. The average temperature was 16 degrees Celsius (°C) (61 degrees Fahrenheit (°F)). The temperature extremes for minimum and maximum recorded temperatures were -24°C and 42°C (-11°F and 108°F). In addition, the applicant also provided a maximum 3-day average temperature of 33°C (90.7°F) based on 1980–2017 data from the Lea County Regional Airport Station.

- Winds: Prevailing wind directions and average wind speeds provided in SAR table 2.3.2 and in the corresponding wind rose diagram of figure 2.3.2, “Lea County Regional Airport Station All Wind Rose (12/01/1948-12/31/2014),” were based on data for the period 1948–2014 from the Lea County Regional Airport. The applicant stated that calm winds (less than 2.1 kilometers per hour (km/h) (1.3 mph)) occurred approximately 8.4 percent of the time, moderate winds (2.1 to 31 km/h (1.3 to 19 mph)) 84 percent of the time, and strong breezes and higher 7.8 percent of the time, calculated from table 2.3.2 values. South is the direction with the highest percentage of winds registering above calm, though west-southwest is the direction with the largest average wind speed and largest percentage of winds greater than 40 km/h (25 mph). Wind gusts reaching 51 to 76 km/h (32 to 47 mph) occurred in 13 percent of the gust observations, and those greater than 76 km/h (47 mph) occurred in less than 1 percent of the gust observations.

- Mixing Height: Average annual and seasonal mixing heights for morning and afternoon appear in SAR table 2.3.3, “Average Morning and Average Afternoon Mixing Heights.” The applicant estimated these atmospheric stability parameters based on the Holzworth calculations for mixing heights across the contiguous United States from “Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States,” by G.C. Holzworth, issued January 1972, using information from Midlands-Odessa, Texas, which is the closest NOAA weather station to the proposed site with relevant meteorological information.

- Tornadoes: The applicant stated that 92 tornadoes were recorded in Lea and Eddy Counties from 1954 to 2016 for an average of 1.5 tornadoes a year. The most tornadoes in a year occurred in 1991 with 15, although 14 of those occurred in a 2-day period. The applicant accessed the tornadohistoryproject.com webpage in 2016, which is no longer accessible. For the contiguous United States, SAR figure 2.3.3, “Tornado Probability Map,” shows New Mexico in the second lowest probability category (1×10^{-6} to 1×10^{-4}) for tornadoes, which the applicant indicated as applicable for the proposed site. Tornado probabilities for the site were cited to the ELEA “Eddy-Lea Global Nuclear Energy Partnership [GNEP] Siting Study,” issued 2007 (2007 GNEP siting study), but were not further traceable.

- Hurricanes: The applicant stated that impacts from hurricanes were unlikely at the proposed site due to the large distance from an oceanic coast.

- Precipitation: The applicant stated that monthly average, monthly maximum, and daily extreme precipitation amounts in SAR table 2.3.4, “Lea County Regional Airport Station Precipitation Data (09/01/1941-06/09/2016),” were based on records for the period 1941–2016 from the weather station at Lea County Regional Airport. Approximately 80 percent of the precipitation falls from May to October. The largest recorded daily precipitation of 9.1 cm (3.6 inches) occurred in December 2016.

- Thunderstorms and lightning: The applicant stated that thunderstorms occur an average of 39 days per year in nearby Carlsbad, New Mexico. The applicant provided no information on lightning for the site.

- Snow: The annual average of 13.0 cm (5.13 inches), monthly maximum of 54.8 cm (21.2 inches), and daily maximum of 25.4 cm (10.0 inches) of snowfall appear in SAR section 2.3.1, based on records from the Lea County Regional Airport weather station for the period 1941–2016. Table 3.6.5, “Lea County Regional Airport Station Snowfall Data (09/01/1941-06/09/2016),” of the ER provides values for monthly and annual minimum, maximum, average, and extreme daily maximum snowfall.

The staff reviewed publicly available tornado data for Lea and Eddy Counties because cited data in the SAR was not traceable. The staff used NOAA/National Weather Service (NWS) historical data from 1954 to the present in the “New Mexico Tornado History,” available at https://www.weather.gov/abq/cli_torns (accessed March 11, 2022), to find information for Lea and Eddy Counties. During that time period, Lea County had 93 tornadoes and Eddy County had 61. Pre- and post-2007 data are reported separately because the inclusion of damage-causing variables in 2007 shifted the wind speeds in each category from the Fujita Scale (F0 to F5) to the Enhanced Fujita Scale (EF0 to EF5). Before 2007, Lea County had 18 F1, 7 F2, and 1 F3 tornadoes, and Eddy County had 6 F1, 5 F2, and no F3 tornadoes. Beginning in 2007, each county had 1 EF2 tornado. The remainder of the tornadoes recorded in the pre- and post-2007 periods were in the F0 or EF0 categories, respectively. Neither county had any F4, F5, EF4, or EF5 tornadoes. F3-classified tornados—the strongest tornadoes recorded in Lea and Eddy Counties—have winds from 254 to 332 km/h (158 to 206 mph). Because the NOAA/NWS tornado data are consistent with the applicant’s description in SAR section 2.3.1, the NRC staff finds the applicant’s description of tornado data adequate for use in the safety analysis.

In SAR section 2.8, “Safety Related Environmental Determinations,” and section 15.3, “Accidents,” the applicant stated that the strength of tornadoes used in the analysis of the HI-STORM FW [flood and wind] and HI-STORM UMAX systems is bounded by the national meteorological tornadic data. In SAR section 3.2.2.1, Safety Features,” and section 3.8, “Regulatory Compliance,” the applicant stated that there is no risk of uncontrolled load movement under a tornado or extreme wind event for the HI-STORE system because there is no freestanding structure. The NRC staff reviewed tornado loading from SAR chapter 4, “Design Criteria for the HI-STORE CIS Systems, Structures, and Components,” and chapter 15, “Accident Analysis,” in SER sections 4.3.3 and 5.3.2 through 5.3.5.

In SAR table 2.7.1, “Site Specific Data for Thermal and Structural Analysis,” the applicant summarized the ambient site temperature values calculated for comparison with the HI-STORM UMAX criteria for thermal analyses for normal and off-normal events, accidents, and short-term operations. SAR table 4.3.2, “Environmental Data for the Licensing Basis in the HI-STORM UMAX Docket and the HI-STORE Site for Different Service Conditions,” and table 6.3.1, “Thermally Significant Parameters for the HI-STORM UMAX ISFSI at HI-STORE and Corresponding Certified Value in the System FSAR,” list the HI-STORM UMAX criteria defined as follows:

- normal condition for storage: average annual temperature
- off-normal condition for storage: maximum 72-hour temperature
- accident condition of storage: maximum 24-hour temperature
- short-term operations: maximum and minimum 3-day average

The conditions in SAR tables 4.3.2 and 6.3.1 are consistent with entries in table 2.3.6 of Revision 3 of the HI-STORM UMAX final safety analysis report (FSAR), dated June 29, 2016. SAR tables 4.3.2 and 6.3.1 also list calculated site-specific values for the average annual temperature, maximum 72-hour (and 3-day average) temperature, and maximum 24-hour temperature.

The NRC staff evaluated several considerations for the adequacy of temperature data statistics presented in SAR section 2.3 and summarized in SAR table 2.7.1. The staff confirmed entries in SAR table 2.3.1 using summaries from the WRCC, which present data from the NOAA/NWS. The staff also confirmed the reasonableness of the 72-hour (and 3-day) average temperatures using 15-minute or 1-hour temperature data provided in SAR section 2.3.1 for the Lea County Regional Airport meteorological station from the NOAA/NWS “New Mexico Tornado History” for several days when historical maximum temperatures were recorded, including a day cited in the footnote of SAR table 2.3.1. The NRC staff also compared site-specific ambient temperature values to the temperature criteria in SAR tables 4.3.2 and 6.3.1 for the HI-STORM UMAX system. The staff notes that site-specific ambient temperatures from the Lea County Regional Airport weather station fall within the bounds of the HI-STORM UMAX temperature criteria for each of the conditions, except for the short-term operations for which the site-specific value of 33°C (91°F) slightly exceeds the HI-STORM UMAX criterion maximum of 32°C (90°F). In SER sections 6.3.3 and 6.3.4.5, the staff discusses the site-specific temperature exceedance of the HI-STORM UMAX maximum temperature criterion in SAR tables 4.3.2 and 6.3.1 for short-term operations. For the accident condition of storage, which is defined as the maximum 24-hour average, the applicant elected instead to use a site-specific maximum daily value from SAR table 2.3.1. The staff notes that a maximum daily value will always be greater than a maximum daily average and thus is a conservative representation for evaluating the accident condition. Because of these considerations, the NRC staff concludes that the applicant provided adequate temperature statistics for use in SAR chapter 6, “Thermal Analysis.”

The staff notes that precipitation rates are not directly used as input in SAR section 2.4.3, “Probable Maximum Flood (PMF),” for estimating flooding at the proposed site. To estimate the consequences of flooding, the applicant used the concept of probable maximum flooding, based on probable maximum precipitation (PMP), which does not directly measure precipitation.

The staff reviewed the regional climate information in the SAR and ER and finds the applicant’s description of the regional climate acceptable because it used reliable data sources such as the WRCC, which is a regional center for climate information organized by NOAA’s National Center for Environmental Information, to characterize typical and severe weather representative of climatic conditions for the proposed site. Therefore, the staff determined that the information on typical and severe weather data is an acceptable source of input to develop the design bases of the facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements of 10 CFR 72.90(a), 10 CFR 72.90(b), and 10 CFR 72.122(b).

2.3.3.2 Local Meteorology

In SAR sections 2.3.2 and 2.3.3, the applicant stated that there are no onsite weather stations. The applicant stated that the Lea County Regional Airport weather station adequately represents the onsite conditions due to the proximity of that station to the site “approximately 30 miles [48 km] away.” In the SAR, the applicant stated that additional details were provided in ER appendix A, section A.2, “Meteorological Data,” on the applicability of the Lea County Regional Airport weather station to the proposed site.

In ER appendix A, section A.2, the applicant considered data from the WIPP site 16 mi (26 km) southwest and the Lea County Regional Airport 37 km (23 mi) northeast of the proposed site. The applicant noted the similarity in average annual temperatures for 2014 and the similarity of wind speeds and predominant wind directions. Compared to the WIPP data (20-year record), the applicant indicated that the Lea County Regional Airport station data provided a longer record (60 years) and a more detailed data set with which to assess minimums, maximums, and seasonal variations. In addition, as opposed to the WIPP station, the applicant stated that the Lea County Regional Airport station provided the detailed data needed for the structural and thermal analyses for the proposed facility.

The staff notes that the 2007 GNEP siting study compared a range of weather data from several meteorological stations in the region surrounding the proposed site. This included data from 1971 through 2000 from the WRCC and NOAA for three weather stations: Lea County Regional Airport, Roswell (New Mexico), and Midland-Odessa (western Texas). Located 156 km (97 mi) southeast of the proposed site, Midland-Odessa is a first order NWS station and is the Weather Forecast Office associated with the Lea County Regional Airport cooperative weather station. Roswell is 113 km (70 mi) northwest and is also a first order NWS station. For each weather station, ELEA (2007) provided tables of (1) minimum, maximum, and average monthly temperature and precipitation and (2) monthly wind averages and directions and wind rose data. Based on its review of the data tables, the staff found that the differences among the stations were not significant. Because the minimum, maximum, seasonal, and annual temperatures; wind; and precipitation are similar at the three sites, the staff expects that local conditions at the proposed site would adequately be represented by detailed data from the weather station selected by the applicant, which is the Lea County Regional Airport station. It is the closest cooperative or first order WRCC or NOAA weather station, contains more than seven decades of records, and includes the required detailed data for structural and thermal calculations.

The NRC staff also considered the regional physiography of the proposed site in assessing the adequacy of the applicant’s use of offsite weather data to represent local site weather. The proposed site is in the Querecho Plains of the Pecos Valley Province (ER figure 3.3.10, “Physiographic Features in Vicinity of CISF Site”). In terms of effects on climatic conditions, the Querecho Plains are relatively flat and featureless (Nicholson and Clebsch, 1961). The Lea County Regional Airport is located immediately to the east of Mescalero Ridge, which separates the Querecho Plains to the west from the high plains of the Llano Estacado to the east. Mescalero Ridge is 30 m (100 feet) high (Nicholson and Clebsch, 1961). Midland-Odessa is in the Southern High Plains to the east of the Pecos Valley Province. The WIPP and Roswell are in the Pecos Valley Province near the Pecos River but outside the Querecho Plains. The staff concludes that there are no prominent physiographic features that would significantly affect weather patterns in the area surrounding the site that extends out several tens of miles.

Based on its review of the regional meteorological data and the discussions presented in the SAR, the ER, and the 2007 GNEP siting study, the staff finds that the applicant's use of the Lea County Regional Airport weather station to represent local conditions at the proposed site is acceptable. The staff also finds that the applicant adequately summarized all relevant data to define the expected meteorological conditions of the site. The staff reviewed the representativeness of the four offsite stations presented in the SAR, ER, and the 2007 GNEP siting study and finds them acceptable because the statistical analysis results are similar among the stations and because all four stations lie in a similar physiographic position in the Querecho Plains and bordering areas within the Pecos Valley.

Based on the discussion above, the staff finds that the information presented in the SAR is acceptable for use to develop the design bases of the facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements of 10 CFR 72.92(a), 10 CFR 72.98(a), 10 CFR 72.98(c)(3), and 10 CFR 72.122(b).

2.3.3.3 Onsite Meteorological Measurement Program

2.3.3.3.1 Onsite Meteorological Station

In SAR sections 2.3.2 and 2.3.3, the applicant stated that there are no onsite meteorological stations. The applicant indicated that the Lea County Regional Airport weather station adequately represented the climatic conditions at the proposed site. In addition, in SAR section 2.3.3, the applicant committed to establishing an onsite meteorological system after a license is issued to collect, at a minimum, temperature, precipitation, and wind data. The applicant indicated that the onsite program would follow the guidance in RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, issued March 2007.

Besides onsite measurements of temperature and precipitation, the planned meteorological stations would provide onsite wind data to use during operations to estimate atmospheric dispersion for offsite concentrations of airborne effluents and onsite hazards of potential offsite accidents. The planned onsite meteorological measurement program that follows RG 1.23 would provide adequate support for the development of joint frequency distributions of onsite wind speed and direction (wind rose) by atmospheric stability class for independent atmospheric dispersion analyses, which the NRC staff finds adequate for compliance with the regulatory requirements in 10 CFR 72.90(e) and 10 CFR 72.98(c)(3).

2.3.3.3.2 Atmospheric Dispersion

In SAR section 2.3, the applicant summarized mixing heights and wind rose data previously reviewed in SER section 2.3.1 and described dispersion calculations for the proposed site in ER section 3.6.2.1, "Dispersion," and section 3.6.2.2, "Stability." Diffusion coefficients are required inputs for independent site-specific atmospheric dispersion modeling of potential atmospheric releases.

The applicant described, in ER section 3.6.2.1, the calculations of model inputs needed to estimate the atmospheric diffusion coefficients for (1) normal and off-normal conditions using

D-Stability conditions and a wind speed of 5 m/s (11 mph) and 100 m (328 feet) to the controlled area boundary and (2) accident conditions using F-Stability and a wind speed of 1 m/s (2.2 mph). The applicant stated that atmospheric conditions for the two scenarios are consistent with the guidance of NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” Revision 0, issued March 2000, and NUREG-1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” Revision 1, issued July 2010. In addition, in ER table 3.6.9, “Pasquill-Gifford Stability Categories,” and table 3.6.10, “Percent Frequency of Occurrence of Atmospheric Stability Classes for Hobbs, New Mexico Area,” the applicant provided the Pasquill-Gifford stability categories and frequencies of occurrence based on Lea County Regional Airport station data. Besides the wind data and stability categories, the applicant used a distance of 100 m for the calculations. The applicant stated that this choice was conservative because the distance to the boundary fence is greater than 100 m. ER table 3.6.8, “Atmospheric Dispersion Coefficients,” gives the derived input values and the resulting atmospheric dispersion coefficients. In ER table 3.6.8, the applicant provided the input values for and estimates of atmospheric dispersion coefficients for normal, off-normal, and accident scenarios for the proposed site. The applicant determined the coefficients through the selective use of equations 1, 2, and 3 in RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Revision 1, issued February 1983.

The terrain surrounding the proposed site can impact atmospheric dispersion calculations. The applicant provided topographic maps illustrating the terrain at and surrounding the proposed site. The maps cover the area immediately surrounding the site and regionally out approximately at least 32 km (20 mi), respectively, using two different resolutions of elevation contours. SAR figure 2.1.10, “Topography of the Site and Surrounding Area,” covers the preconstruction topography of the site. ER figure 2.3.1, “Potential Sites Evaluated by ELEA,” and figure 3.4.1, “Topography of Site and Surrounding Area,” show that the area surrounding the site includes several playas and mining excavations within 6 km (4 mi). Bordering the Querecho Plains, the figures show that the Mescalero Ridge 18 km (11 mi) to the northeast of the proposed site is the closest large-scale terrain feature in the otherwise relatively flat Querecho Plains. Directional elevation profiles for atmospheric dispersion calculations could be extracted from the topographic maps.

Based on its review of ER (version 9) sections 3.6.2.1 and 3.6.2.2, the staff finds the applicant’s dispersion calculation acceptable because it is consistent with the guidance in RG 1.145 for determining atmospheric dispersion estimates for both postulated accidents and expected routine releases of gaseous effluents in compliance with the regulatory requirements in 10 CFR 72.90(e) and 10 CFR 72.98(c)(3).

2.3.4 Surface Hydrology

The staff reviewed SAR section 2.4, “Surface Hydrology,” with regard to (1) the hydrologic description, (2) floods, (3) the probable maximum flood (PMF) on streams and rivers, (4) potential dam failures, (5) probable maximum surge and seiche flooding, (6) probable maximum tsunami flooding, (7) ice flooding, (8) flood protection requirements, and (9) environmental acceptance of effluents.

2.3.4.1 Hydrologic Description

The applicant proposed to build the HI-STORE CIS Facility within the Upper Pecos-Black watershed, which lies within the Pecos River Basin, as shown in ER figure 3.5.1, "Pecos River Basin Drainage Area." The Pecos River Basin extends from eastern New Mexico into West Texas. It has a desert climate and overlaps a large portion of the Permian Basin that offers a rich energy supply in its underground rock layers (see the discussion in SER section 2.3.5.1). The Pecos River is the nearest through-flowing surface water feature to the site, with a drainage area of more than 110,000 square km (44,000 square miles). The Pecos River empties into the Rio Grande in Texas. The nearest distance from the proposed facility to the Pecos River is about 26 mi. Two surface water bodies, Brantley Lake and Red Bluff Reservoir, are about 64 to 72 km (40 and 45 mi) to the west and south of the site, respectively. Geographically, the site is 51 km (32 mi) east of Carlsbad and 55 km (34 mi) west of Hobbs, both in Lea County, New Mexico. It is located northeast of the western terminus of New Mexico Highway 176, intersecting U.S. Highway 62 (SAR figure 2.1.1) and is sandwiched between the two playa lakes, Laguna Plata and Laguna Gatuna (SAR figure 2.1.9, "Topography of Site and Surrounding Area"). The applicant also identified two smaller playa lakes, Laguna Tonto and Laguna Toston, to the northeast and southwest of the site, respectively. In GEI Consultants (GEI) Report CIS-RP 003-03, "Probable Maximum Flood Analysis, HI-STORE CISF, Lea County, New Mexico," Revision 3, issued April 2022 (2022 GEI flooding analysis report), the applicant determined that the drainage basin of Laguna Tonto contributes to surface water flow in the neighboring Laguna Gatuna watershed.

The site is situated between the Lower Pecos Valley and Llano Estacado sections of the Great Plains physiographic province (Hawley, 1986; figure 2.3.1-1, ELEA, 2007). Surface drainage at the site is contained within two primary water basins, the Laguna Gatuna and Laguna Plata subbasins, which are classified as closed water basins in the USGS Watershed Boundary Dataset (WBD) (USGS and Natural Resources Conservation Service (NRCS), 2013). The site is positioned on the divide of the two drainage basins sloping northwesterly to Laguna Plata and easterly to Laguna Gatuna (SAR figure 2.4.7, "General Topography around the Proposed CIS Facility Site"). Existing human-made structures lie between the proposed facility and Laguna Gatuna (see SAR figure 2.1.7, "Soils Survey Map"). Two ephemeral streams draining eastwards and westwards are also clearly visible on the satellite image, with the eastward-draining ephemeral stream flowing through the human-made structures. Part of the playa lake is within the applicant's controlled area boundary but outside the protected area (SAR figure 2.1.7). Playas are the most common surface hydrological features of the Llano Estacado (Gitz and Brauer, 2016). They are naturally occurring, circular basins with closed-system watersheds and basin floors containing soils of higher clay content than the surrounding area (Bolen et al., 1989). As a result, retention of water from intense rainfall events and shallow ground water discharge may occur on the basin floor. Playa lakes thus become water retention sinks during large rainfall events and help in creating shallow saline ground water below and around them.

In SAR section 2.4.1, "Hydrologic Description," the applicant described the four playas, Lagunas Plata, Gatuna, Tonto, and Toston, as generally dry but retaining runoff temporarily, and the runoff does not drain to any of the State's major rivers. Surface water is lost through evaporation, resulting in high salinity conditions in soils within the playas. Laguna Gatuna, located on the eastern boundary of the site, covers a surface area of 1.4 square km (0.54 square miles). Its elevation is 1,065 m (3,495 feet) AMSL and drains a watershed of 422 square km (163 square miles), including the Laguna Tonto watershed. It was the site of multiple facilities for collection and discharge of brines coproduced from oil and gas wells, with facility permits authorizing the discharge of almost one million barrels of oilfield brine per month between 1969 and 1992. Laguna Tonto, located approximately 4.0 km (2.5 mi) northeast of the site, covers a surface area of 0.72 square km (0.28 square miles) at an elevation of 1,076 m (3,531 feet) AMSL. It drains a subbasin that covers 71 square km (44 square miles). Laguna Plata, located approximately 2.9 km (1.8 mi) northwest of the site, covers a surface area of 5 square km (2 square miles) at an elevation of 1,046 m (3,432 feet) AMSL. It drains a subbasin that covers 658 square km (254 square miles). The applicant suggested that alluvial ground water in the vicinity of the site appears to flow toward Laguna Plata. Laguna Toston is the smallest of the playas in the vicinity of the site, with a surface area of 0.65 square km (0.25 square mile). The playa is a major input point for potash refinery brine, and water appears to drain radially away from this location. Its elevation is approximately 1,070 m (3,500 feet) AMSL (SAR figure 2.1.9), higher than Laguna Plata. In the 2022 GEI flooding analysis report, the applicant did not include Laguna Toston in its hydrological and hydraulic models for the site. SAR figure 2.4.1, “Regional Map,” and figure 2.4.3, “Lakes/Playas in the Vicinity of the CIS Facility,” indicate that no freshwater lakes, estuaries, or oceans exist in the vicinity of the site.

Hydrologically, the site is situated in the playa-lake basins of the Southern High Plain of New Mexico, where water may collect periodically in surficial depressions (Osterkamp and Wood, 1987). The basins may expand by various hydrological and geomorphic processes. Aeolian removal of clastic materials from the floor of playa lakes may sometimes deepen playa depressions. Playa basins tend to develop where water collects and infiltrates, along ephemeral streams and lineation suggestive of fracture systems. Locally near the HI-STORE CIS facility, the area is generally characterized by ephemeral drainage features, sheet flow, minor gullies and rills, and internally drained playas. A continuous system of drainage channels is not apparent in the area. As shown in SAR figure 2.1.5, ephemeral washes that may temporarily retain surface runoff from locally intense storms exist along the western and northern boundary of the site. Surface water retained in the washes drains eastwards to Laguna Gatuna and westwards to Laguna Plata. Farther from the site, the USGS spot streamflow measurements at Monument Draw (located near Monument, New Mexico, approximately 27 km (17 mi) from the site) indicate a record flow of 36.2 cubic meters per second (m^3/s) (1,280 cubic feet per second (cfs)) in 1972 (http://nwis.waterdata.usgs.gov/nm/nwis/ peak/?site_no=08437620). At Antelope Draw near Jal, New Mexico, approximately 36 mi from the site, a record flow of 19.4 m^3/s (686 cfs) was measured in 2001 (State of New Mexico Interstate Stream Commission, 2016). In SAR section 2.4.1, the applicant suggested that these records are indicative of flows that can occur at other gullies and swales in Lea County. According to the 2007 GNEP siting study, the site has no sensitive or unique aquatic or riparian habitats or wetlands, and surface water in the vicinity is not potable.

In the 2022 GEI flooding analysis report, the applicant described the drainage area encompassing the site as consisting of Laguna Gatuna Basin (307.2 square km (118.6 square miles)), Laguna Tonto Basin (114 square km (43.9 square miles)), HI-STORE Site Basin (9.8 square km (3.8 square miles)), Laguna Plata North Basin (179 square km (69.3 square miles)), and Laguna Plata South Basin (25 square km (9.7 square miles)). The size of the entire drainage area is approximately 635.3 square km (245.3 square miles). Water collected in Laguna Tonto Basin discharges into Laguna Gatuna Basin. Figure 2 of the PMF analysis report shows a discharge point of the entire drainage area located near the southwest shore of Laguna Plata.

The area of a watershed is an important factor for determining PMP (NOAA, 1978). Using the USGS WBD (USGS and NRCS, 2013), the staff verified that the applicant included the subbasin Shilo Well as part of the Laguna Gatuna Basin. The USGS WBD indicated that the Shilo Well subbasin is an upstream basin of the Laguna Gatuna subbasin. The two subbasins have a total area of 308.88 square km (119.26 square miles). The staff notes that a portion of the WBD Laguna Gatuna subbasin was identified as part of the HI-STORE Site Basin in the 2022 GEI flooding analysis report. The USGS WBD also identified two subbasins, Laguna Plata and South Tonto Oil Field, with the latter upstream of the former. These two subbasins contain the Laguna Plata North and Laguna Plata South subbasins and a portion of the HI-STORE Site subbasin in the 2022 GEI flooding analysis report. The two USGS WBD subbasins have a total area of 212.7 square km (82.13 square miles), compared to 214.4 square km (82.80 square miles) of the aforementioned three basins in the 2022 GEI flooding analysis report. The USGS WBD also indicated that the Laguna Tonto subbasin has a total area of 113.8 square km (43.95 square miles). According to the USGS WBD, the total area of the five subbasins enclosing the site is 635.40 square km (245.33 square miles). Based on these data, the staff determined that the watershed delineation by the applicant is acceptable.

In SAR section 2.1.2 and ER appendix D, "Additional Soil Data," the applicant cited soil survey maps of Lea County from the U.S. Department of Agriculture (USDA) NRCS and determined that 90 percent of the soils within the site consist of fine sandy loam and gravelly fine sandy loam underlain by cemented material. The NRC staff reviewed soil survey information from the USDA NRCS (https://websoilsurvey.sc.egov.usda.gov/App/WebSoilSurvey.aspx, last accessed in December 2021) and determined that soils near the site are mostly fine sandy loams and gravelly fine sandy loams that originated from aeolian deposits and derived from sedimentary rocks. The USDA NRCS classified these soils as well drained, with the restricting soil horizon having water transmission rates from very low to moderately high. Minor soil groups consisted of mostly sand and sandy loam to clay loam, particularly near the two ephemeral streams, that are alluvial deposits derived from sedimentary rocks. The NRCS also classified these soils as well drained, with water transmission rates of the restricting layer from high to moderately high. Restrictive features can be encountered within 51 cm (20 inches) below the surface for a few soil groups. The playa floor consists of mostly silty clay loam to clay of mixed alluvium and lacustrine derived from sedimentary rock, with low water transmission rates.

The average annual precipitation of the Southeastern Plains, New Mexico, is about 35.8 cm (14.1 inches) from 1895 to 2020 (NOAA, 2021). Near the site at the City of Roswell, New Mexico, the annual average precipitation was about 30 cm (12 inches) from 1895 to 2020. In Lea County, New Mexico, the annual rainfall averages between 30 and 46 cm (12 and 18 inches), with year-to-year precipitation varying from less than an inch to more than 91 cm (36 inches) (State of New Mexico Interstate Stream Commission, 2016). In SAR section 2.4.1, the applicant stated that 75 percent of the total annual precipitation and 60 percent of the annual Pecos River flow result from intense local thunderstorms in April through September. The runoff from the thunderstorms is collected by intermittent streams, lakes, stock ponds, and small playas. In SAR section 2.5, "Subsurface Hydrology," the applicant stated that evapotranspiration at the site is five times the precipitation rate. The reported average annual pan evaporation rate at Brantley Reservoir northwest of Carlsbad between 1987 and 2005 was 277 cm (109 inches) (Desert Research Institute, 2021), which is about six to nine times the reported average annual rainfall of Lea County, New Mexico. In SAR section 2.5, the applicant reported a shallow ground water table at a depth between 11 and 15 m (35 and 50 feet), based on two wells near the site. The applicant attributed the shallow water table to water infiltrated from and controlled by the playa lakes. The applicant also reported that the regional ground water table is on the order of 90 and 120 m (300 to 400 feet) below the ground surface. Based on monitoring wells at the site, the applicant determined that the primary ground water table depth at the site is approximately 77.1 and 80.1 m (253 to 263 feet) below the ground surface.

In SAR section 2.4.1, the applicant stated that no major surface water supplies are available in Lea County, New Mexico. Only intermittent streams, lakes, stock ponds, and small playas collect runoff during thunderstorms. SAR section 2.4.2, "Floods," states that these surface water supplies are transitory and limited in quantity. In SAR section 2.5, the applicant stated that potable water is generally obtained from potash company pipelines, which convey water from the Ogallala High Plains aquifer of eastern Lea County. The applicant did not explicitly state in the SAR the sources of water supply for the site. However, the 2007 GNEP siting study stated that the planned GNEP facility will be able to tap into the Double Eagle Water Resources System owned and operated by the City of Carlsbad. The 2007 GNEP siting study area is essentially the same as that of the proposed HI-STORE CIS Facility.

According to SAR table 1.0.1, the proposed site area is about 422.9 hectares (1,045 acres). Based on SAR figures 2.5.3, 2.5.4, and 2.5.5, the elevation of the bottom of the UMAX storage pad is designed to be at 1,070 m (3,510 feet) AMSL and 6.1 to 7.6 m (20 to 25 feet) below the ground surface. According to SAR section 2.4.1, the slab foundations for the administration building, security building, CTB, and warehouse will be located at or slightly above the existing grade at the approximate nominal elevations of 1,080, 1,080, 1,079, and 1,079 m (3,545, 3,545, 3,540 and 3,540 feet) AMSL, respectively.

Based on its review, the staff finds that the applicant sufficiently characterized the surface hydrologic features of the region, area, and site to support the hydrologic engineering analyses and appropriately described the location, size, and hydrologic characteristics of all water sources that influence or may influence the site or facilities under severe hydrologic conditions. The applicant provided maps to support a clear understanding of these features. The staff finds that the applicant identified the sources of the hydrologic information and the surface waters that could potentially be affected by normal or accidental effluents from the site. The applicant also identified the onsite structures important to safety that may be adversely affected by extreme hydrologic events. Therefore, the staff finds that the information provided by the applicant is in compliance with the requirements in 10 CFR 72.92(b).

2.3.4.2 Floods

In SAR section 2.4.2, the applicant stated that the site elevation is approximately at 1,076 m (3,530 feet) AMSL according to topographic maps of the USGS. Citing the flood insurance study and online flood map of the Federal Emergency Management Agency (FEMA), the applicant stated that the site is not within the 100-year and 500-year floodplains. The staff verified that the FEMA flood map identified the vicinity of the site, including the area surrounding Lagunas Gatuna and Plata, as flood zone D or an area of undetermined flood hazard. The staff also conducted a general search of USGS flood reports, which did not indicate the site area has experienced flooding in the past.

There is a lack of a definitive, continuous stream channel or drainage feature in the site area. Surface water follows land surface and ephemeral streams and discharges into nearby playa lakes, particularly Laguna Planta and Laguna Gatuna. The applicant concluded that, in this geographical area where there are no significant bodies of water or rivers in the vicinity of the site, the only plausible flooding hazard to the site is from stormwater runoff during rainfall events. The staff determined that this conclusion is acceptable because of the desert climate and the surrounding hydrologic features. The rainfall events in this case may include a local intensive storm or a large, watershed-wide extreme weather event such as a PMP storm, or both.

The applicant conducted the flood hazard study, documented in the 2022 GEI flooding analysis report, using the Hydrologic Engineering Center Hydrologic Modeling System (HEC-HMS) (U.S. Army Corps of Engineers (USACE), 2021a) and River Analysis System (HEC-RAS) (USACE, 2021b) software. The watershed contributing to drainage across the site is about 63,540 hectares (245.33 square miles or 157,011 acres) and is divided into five distinct subbasins in the study (Laguna Plata North, HI-STORE Site, Laguna Plata South, Laguna Tonto, and Laguna Gatuna in SAR figure 2.4.12, “PMP/PMF Watersheds”). To conduct hydrologic and hydraulic modeling studies of the entire watershed, soil groups and land cover information, besides rainfall and precipitation information, are needed as input to the HEC-HMS and HEC-RAS software.

The applicant gathered soil resource reports of the watershed from generic databases such as the USDA Web Soil Survey (USDA, 2020) and the National Land Cover Database (NLCD) (Multi-Resolution Land Characteristics Consortium, 2019). The land cover and hydrologic soil group information (e.g., ER figure 3.3.8, “Soil Survey Map”), in conjunction with USDA TR-55, “Urban Hydrology for Small Watersheds,” issued June 1986, may then be used to derive curve numbers of individual combinations of soil group and land cover for hydrologic and hydraulic modeling of floods. In ER section 3.3.4, “Soils,” the applicant cited the geotechnical site characterization for design phase 1 of the proposed site and indicated that a thin layer of topsoil, generally 8 to 10 cm (3 to 4 inches), is underlain by caliche caprock and clayey sand and sandy clay residual soil. In section 5.2, “Soil and Rock Descriptions,” of GEI Report CIS-RP 001-01, “Geotechnical Data Report: HI-STORE CISF Phase 1 Site Characterization Lea County, New Mexico,” Revision 1, issued February 2018, the applicant indicated that a restricted layer, caliche, was encountered beneath the topsoil in all borings taken at the site. The caliche is continuous in the subsurface in the general area surrounding the site, and field tests indicated that the deposit is a dense to very dense soil. Section 5.2.1.2, “Caliche,” and Section 5.2.1.3, “Residual Soil,” of the 2018 GEI geotechnical data report discuss the clayey sand or sandy clay layer beneath the caliche as a very hard or very dense soil. The caliche layer may restrict water infiltration to subsoils, and the clayey layer is likely to further restrict recharge to bedrocks and ground water during intense storms. In SAR section 2.1.2, the applicant summarized the soil survey map (SAR figure 2.1.7) obtained from the USDA NRCS, which suggested that a great majority of soils at the site consist of sandy loam with hydraulic conductivities varying from nearly impermeable (the cemented materials 20 to 66 cm (8 to 26 inches) from the ground surface) to tenths of meters per hour (inches per hour) (sandy and gravelly loams near the surface). In the hydrologic model developed by the applicant for the design-basis flood, the soils over the entire watershed were conservatively assumed to be impermeable. Evaporation and surface interception of rainfall were conservatively ignored. These two assumptions resulted in no rainfall losses, or the entire PMP rainfall amount was applied to the land surface for runoff generation.

The applicant’s flood study includes rainfall events for PMP estimated by two approaches: the “Colorado—New Mexico Regional Extreme Precipitation Study” (CO-NM), by the Colorado Division of Water Resources, issued 2018, and the methods in NOAA’s Hydrometeorological

Report (HMR) No. 51, “Probable Maximum Precipitation Estimates, United States East of the 105th Meridian,” issued June 1978, and HMR No. 52, “Application of Probable Maximum Precipitation Estimates—United States East of the 105th Meridian,” issued August 1982 (HMR 51/52). In SAR section 2.4.3, the applicant stated that its safety analysis used a conservative, deterministic analytical approach outlined in NUREG/CR-7046, “Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America,” issued November 2011, indicating that the applicant considered the PMF calculated from the PMP as the design-basis flood. The applicant further stated, in SAR section 2.4.3, that computed peak water surface elevations, depths, and velocities were used to understand conditions at Phase I of site activities and to assess the adequacy of the proposed project during a PMP event. Based on its review, the staff finds the applicant’s assessment approach acceptable because it uses commonly accepted models, conservative assumptions, and areal precipitation information to determine the maximum flood levels and flow rates. Based on its confirmatory analyses, described below, the staff also finds that the PMP event chosen by the applicant results in the maximum flood level and flow rate across the upstream boundary of the site. The following sections discuss the PMP event and the impact of the resulting PMF on the site and its SSCs important to safety.

2.3.4.2.1 Probable Maximum Precipitation

In SAR section 2.4.3, the applicant indicated that it used the CO-NM and the HMR 51/52 methods to evaluate the PMPs of four durations: 6 hours, 24 hours, 48 hours, and 72 hours (SAR table 2.4.2, “PMP Values and Governing Method”). In the 2022 GEI flooding analysis report, the applicant described the use of the CO-NM method to calculate the PMP rainfall of the various durations. In section 3.2, “HMR 51/HMR 52,” of the 2022 GEI flooding analysis report, the applicant also used the HMR 51/52 areal storm method with the software HMR52 (USACE, 1987) to evaluate the amount of precipitation within each of the five subbasins from a single probable maximum storm (PMS) sitting above the entire watershed or the site. Distribution of a rainfall amount with a PMP of specific duration, or the hyetograph, is also computed by HMR52 and used as input to the HEC-HMS software. With the PMS sitting above the entire watershed, the applicant calculated a 72-hour PMP of 63.0 cm (24.8 inches) at the site. When the PMS was above the site, the HMR52 software determined a 72-hour PMP of 82.3 cm (32.4 inches) (SAR table 2.4.2). The 72-hour watershed-centered PMS produced a peak flood water depth of 97.5 cm (3.2 feet) at the Phase 1 ISFSI pad, compared with the 6-hour CO-NM local storm, which produced a PMP of 54.1 cm (21.3 inches) at the site and a peak flood water depth of 82.3 cm (2.7 feet) (SAR tables 2.4.2 and 2.4.4, “Peak Flood Depth for Each Storm Duration at ISFSI Center,” respectively). These results suggest long-duration, watershed-wide PMPs are controlling precipitation events at the site (SAR table 2.4.4). Between the two 72-hour PMSs, one above the entire watershed and the other above the site, the applicant further determined that the former is the controlling storm because the latter produced a smaller peak flood water depth at the Phase 1 ISFSI pad than that of the former (SAR table 2.4.4).

Using NOAA HMR Nos. 51 and 52, the staff performed a confirmatory analysis to independently estimate the 72-hour PMP to be 62.7 cm (24.7 inches) at the site when the PMS is above the entire watershed, comparing to 63.0 cm (24.8 inches) estimated by the applicant. Based on its review and the confirmatory analysis, the staff finds that the applicant appropriately determined the PMP because it followed the commonly accepted methodology (for example, see NUREG-1632, “Design of Erosion Protection for Long-Term Stabilization, issued 2002, for assessing rainfall potential with minimal variability) and the NOAA HMRs for specific regions of the United States, and a local 6-hour thunderstorm PMP occurring at the site is bounded by the 72-hour PMP. Because the 72-hour general, watershed-centered PMP generated the highest peak flow rate across the upstream boundary of the site and the highest flood water depth at the site, and it also bounded PMPs of other durations, as shown in SAR table 2.4.4, the staff's evaluation and discussion of models and model results focus on this 72-hour PMP and the resultant flood water depths and water velocity.

2.3.4.2.2 Hydrologic Modeling of Surface Runoff

The applicant described a hydrologic model and model results in the 2022 GEI flooding analysis report. The hydrologic model encompasses the five subbasins noted previously. PMPs with their hyetographs and reservoir and surface flow characteristics are required input to the model. The general engineering practice is to use hydrologic routing methods to transform rainfall excess to surface runoff. Rainfall losses due to evaporation, surface interception, and infiltration into subsoils are usually accounted for with techniques such as the Initial and Constant Loss Method (USACE, 2021a). In the applicant's hydrologic model, the PMP rainfall was applied directly to the five subbasins, without hydrologic transform and rainfall losses. Surface water flow rates and reservoir water levels at Lagunas Plata, Gatuna, and Tonto are the output from the model. Calculated surface water flow rates are then used as boundary conditions for the hydraulic model to determine flood water levels and velocities at and surrounding the site.

There are three large playas in the vicinity of the site (i.e., Lagunas Gatuna, Plata, and Tonto). Immediately to the east of the site, a portion of Laguna Gatuna is within the site boundary. Water discharged from Laguna Tonto eventually joins surface runoff entering Laguna Gatuna. In SAR section 2.4.3, the applicant indicated that, in accordance with NUREG/CR-7046, the hydrologic models assume the lagunas are in an "extreme full" condition before the PMPs. Appendix A, "Laguna Stage Storage Tables," to the 2022 GEI flooding analysis report documents the storage-elevation curves of the lagunas. In appendix C, "HEC-HMS Input/Output Summary," to the 2022 GEI flooding analysis report, the applicant further indicated that the outlets of the Lagunas Tonto and Gatuna were represented by a broad-crested spillway to replicate the natural channel downstream of the lagunas. The channel geometry was estimated based on USGS topographic maps, digital elevation maps, and Google Earth imagery. Elevation-storage-discharge curves were used to characterize the outlet of Laguna Plata. The applicant derived the elevation-storage-discharge curves based on the results obtained from an HEC-RAS hydraulic model of the outlet areas of Laguna Plata.

The staff independently estimated the storage-elevation curves of the lagunas and the outlet geometry using a digital elevation map independently obtained from the USGS. This information was compared with that of the applicant. Based on the outlet area geometry of Laguna Plata and the weir equation used in HEC-HMS (USACE, 2021a), the staff estimated the stage-discharge relationship for Laguna Plata, which, in turn, was converted to a storage-discharge curve and compared with that obtained by the applicant. Based on its review and independent calculations, the staff found the applicant's playa storage-elevation curves, laguna outlet geometry, and Laguna Plata storage-discharge curve acceptable because they were based on an industry-accepted method and USGS digital elevation maps, and the results by the applicant are comparable to those independently calculated by the staff.

As noted previously, the applicant applied PMP rainfall directly to the subbasins without hydrologic transform and ignored rainfall losses caused by evaporation, surface interception, and infiltration. The applicant also evaluated the flow rates across the upstream boundary of the site caused by two PMSs with different storm centers, one above the entire watershed and the other above the site. The combination of no hydrologic transform, no rainfall losses, and a watershed-centered PMS is generally considered as conservative. It is less obvious whether a PMS above the site that generates locally intense storm runoff would have a peak water flow in-phase or off-phase with storm runoff from upstream subbasins (i.e., Lagunas Gatuna and Tonto), if rainfall to surface runoff transforms are applied following generally accepted engineering practices. To confirm that the combination of no hydrologic transform and locally intense storm runoff does not produce unconservative results, the staff conducted an independent analysis to calculate the hydrograph of locally intense storm runoff and a hydrograph of inflow from the upstream areas of the site. The staff's independent calculation indicated that the peak of locally intense storm runoff is off-phase with the peak of upstream inflow from Lagunas Gatuna and Tonto with or without hydrologic transform. However, without hydrologic transform, locally intense storm runoff dominated the total inflow to the site. In contrast, with hydrological transform, flood water from the upstream boundary consistently dominated inflow to the site. In both cases, the watershed-centered PMS produced the higher peak flow rate that directly impacts flood water depth at the site. Furthermore, the flow rate calculated without hydrologic transform (as assumed by the applicant) is more than 12 percent higher than the flow rate with hydrologic transform (as calculated by the staff). Therefore, the combination of hydrological and meteorological conditions used by the applicant's HEC-HMS models can be considered conservative, following the guidance of NUREG/CR-7046.

In summary, the staff considers the hydrologic models, model parameters, and flow rates thus calculated acceptable because the applicant used an industry-accepted modeling approach and followed the guidance of NUREG/CR-7046, and the resultant flow rates that directly impact flood water depth at the site are conservative. With the PMS centered above the entire watershed, in table 2, "Hydrologic Model Results," of the 2022 GEI flooding analysis report, the applicant calculated a peak inflow from Laguna Gatuna of 2,495.0 m^3/s (88,111 cfs) and a respective one from the site surrounding the facility of 414.98 m^3/s (14,665 cfs). In comparison, the staff's calculations suggested inflows of 2,271.4 m^3/s (80,214 cfs) and 194 m^3/s (6,860 cfs) from Laguna Gatuna and the site, respectively, which are lower than those calculated by the applicant because the staff considered hydrologic transform of rainfall to surface runoff. The inflow across the boundary of the site obtained through the applicant's hydrologic models from table 2 of the 2022 GEI flooding analysis report is thus suitable for further hydraulic modeling of flood water depth and velocity at the site.

2.3.4.2.3 Land Surface Cover and Surface Roughness Coefficient

For hydraulic modeling of flood water depth and velocity, the Manning's roughness coefficients (i.e., the n-values) are required for various types of ground surface. The surface roughness coefficient is a key parameter in calculating flood water depth and velocity. The applicant's sensitivity analysis, from section 5.2, "Sensitivity Analysis," and table 5, "Hydraulic Sensitivity Model Results," of the 2022 GEI flooding analysis report, indicated that the peak flood water depth is most sensitive to the roughness coefficient.

In section 4.2.2.3, "Manning's n-values," of the 2022 GEI flooding analysis report, the applicant described the methodology from which the surface roughness coefficients were estimated. The applicant used the land cover information from the 2016 NLCD for the conterminous United States (MRLC, 2016) in conjunction with information from the literature (e.g., Chow, 1959; and

Reid and Hickin, 2008), as listed in appendix G, “Previous Analysis,” to the 2022 GEI flooding analysis report to map land cover types to Manning’s n-values.

The staff reviewed the land cover types obtained by the applicant and compared the derived surface roughness coefficients with those in the literature. Because the applicant followed the general industry practices of hydraulic engineering and the surface roughness coefficients were obtained from the general literature, the staff found the Manning’s n-values used for the HEC-RAS hydraulic models of the site acceptable.

2.3.4.2.4 Hydraulic Modeling of Flood Water Depth and Velocity

In section 4.2.2, “HEC-RAS model set up,” of the 2022 GEI flooding analysis report, the applicant described hydraulic models for the site and a portion of the Laguna Plata watershed (including the north and south shores of Laguna Plata, which are part of the Laguna North and Laguna South subbasins in the HEC-HMS hydrologic models), with an area of approximately 37.0 square km (14.3 square miles). The applicant used the HEC-RAS software developed by the USACE Hydrologic Engineering Center (USACE, 2021b) to simulate flood water depth and velocity within and surrounding the HI-STORE CIS Facility. The applicant assumed the initial water surface elevation of Laguna Plata to be at the “full pool” condition, consistent with the HEC-HMS hydrologic models. As described in appendix D, “HEC-RAS Input Summary,” to the 2022 GEI flooding analysis report, a digital elevation map from the USGS National Elevation Dataset was used for the model terrain, and onsite features such as buildings, railroads, and the vehicle barrier system (VBS) were delineated in refined meshes within the model domain. In the hydraulic model, the VBS is represented with blocks of 4.9 m long by 1.2 m wide (16 feet long by 4 feet wide) with a 20 cm (8-inch) spacing between blocks. Upstream boundary conditions were flow hydrographs calculated by the HEC-HMS hydrologic model, and the downstream boundary near the discharge area of Laguna Plata was set at the normal depth, based on the approximate average channel slope downstream of Laguna Plata.

In figure 11, “Peak PMF Depth Proposed Condition,” of the 2022 GEI flooding analysis report, the applicant’s calculation suggested the VBS, in effect, deflects flood water from upstream of the HI-STORE CIS Facility, resulting in a lower flood water depth at the CTB for the proposed condition in comparison with the existing condition, as shown in SAR table 2.4.3, “Peak PMF Depth and Velocity for Key CISF Structures.” As shown in SAR table 2.4.3, the flood water receiving area near the “Wash to the West” has a relatively large increase of flood water depth of 1.5 m (5.4 feet) with the VBS put in place (from 4.51 m to 6.16 m (14.8 feet to 20.2 feet)). On the downstream side of the facility, the VBS becomes a flood water barrier, resulting in higher flood water depth at the Phase 1 ISFSI pad for the proposed condition, at 0.98 m (3.2 feet) compared to 0.52 m (1.7 feet) in the existing condition. These results suggest that the VBS performs the hydraulic functions of deflecting flood water away or backing up flood water into the SSCs, depending on the location of the SSCs within the facility. The effects of the VBS, however, may depend on the final design of the facility. In section 5.2, “Sensitivity Analysis,” of the 2022 GEI flooding analysis report, the applicant indicated that currently there is no design information indicating whether runoff would enter the site through the VBS. According to the applicant’s calculations, the flood water depth at the CTB can thus vary between a few inches and half a foot. On the other hand, the applicant’s calculation suggests the proposed railroad spur south of the facility does not alter the nearby flood water pattern before or after the construction of the facility.

In section 4.2.2.4, “Boundary Conditions,” of the 2022 GEI flooding analysis report, the applicant described the sources of water that can cause flooding of the facility’s onsite SSCs. The applicant used the flow hydrographs calculated by HEC-HMS, including those for Laguna Gatuna and the site subbasins, as upstream boundary conditions for the HEC-RAS hydraulic model. Precipitation from local intense storms was modeled as an upstream boundary condition rather than direct rainfall. Therefore, the staff conducted an independent analysis of the effect of direct, locally intense rainfall on the PMF water depth near the facility SSCs, using the WASH123D software (Yeh et. al., 1998). The model encompasses the Gatuna and Plata watersheds as delineated by the USGS (USGS and NRCS, 2013). PMP rainfall was used directly as the source of flood water at the site with the same full-pool condition at Lagunas Plata and Gatuna as that used by the applicant. The staff’s results suggested that, under existing conditions, flood water depth is approximately 0.24 m (0.8 foot) south of the CTB, compared to 0.15 m (0.5 foot) calculated by the applicant, as shown in SAR table 2.4.3. At the Phase 1 ISFSI pad area, the staff calculated a flood water depth of approximately 0.43 m (1.4 feet), compared to 0.58 m (1.9 feet) calculated by the applicant. While the watershed model, source of flood water, and boundary conditions used by the staff differ from those used by the applicant, the staff’s results are comparable to those of the applicant. The staff thus determined that using upstream boundary flux in place of local intense precipitation for the site subbasin was acceptable.

In SAR section 2.4.3, the applicant stated that there is uncertainty in the hydraulic model representation of the VBS and in the passage of water through the gaps due to the disparate mesh sizes around the modeled VBS blocks. In the HEC-RAS hydraulic model in the 2022 GEI flooding analysis report, the height of VBS blocks was not specified, and the blocks were modeled as areas with high roughness coefficients. To address the model uncertainty, the applicant committed, in SAR section 1.1, “General Description of Installation,” to constructing the VBS surrounding the facility as continuous runs with a minimum height of 3 feet (0.91 m) and stated that the continuous runs act as an obstruction to rainwater runoff flow entering the facility during design basis flood events.

With the proposed Phase 1 ISFSI pad and other onsite SSCs, including the VBS as proposed by the applicant, the CTB, and the railroad spur, the staff conducted a confirmatory calculation using the HEC-RAS software. Results obtained by the staff suggested that the flood water maximum depth and peak velocity at the Phase 1 ISFSI pad area are 0.91 m (3.0 feet) and 1.3 m/second (m/s) (4.3 feet/second (feet/s)), compared to 0.98 m (3.2 feet) and 0.73 m/s (2.4 feet/s) calculated by the applicant and indicated in SAR table 2.4.3. The staff also calculated the flood water maximum depth and peak velocity south of the CTB as less than 2.5 cm (1 inch) and 0.3 m/s (1 foot/second (foot/s)), compared to no flood water accumulation obtained by the applicant. Thus, the flood water depths from the staff’s independent calculation are comparable to those from the applicant. The staff’s calculation also suggested that the storm runoff peak velocity at the ISFSI pad, northeastern and southeastern corners of the VBS line, a portion of Haul Path Section 3, and a portion of the rail track inside the HI-STORE CIS Facility boundary may exceed the erosion allowable velocity of 1.0 m/s (3.3 feet/s). To address the potential for erosion, the applicant committed to a 1.5 m (5-foot) interior and exterior border of a material (e.g., gravel or riprap), with sufficient erosion resistance to withstand the velocity well above the erosion allowance of native soils, which will serve to eliminate any local erosion from undermining the VBS and fences. Additionally, the applicant will replace any eroded materials within the HI-STORE CIS Facility and regrade the ground surface appropriately after storm waters recede.

2.3.4.2.5 Flood Impacts to Onsite Structures

In SAR section 2.4.1, the applicant described the slab foundation elevation of site structures important to safety. The elevation of the site proper ranges from approximately 1,076 m to 1,082 m (3,530 to 3,550 feet) AMSL. The slab foundations for the administration building, security building, CTB, and warehouse are to be located at or slightly above existing grade at approximate nominal elevations of 1,080, 1,080, 1,079, and 1,079 m (3,545, 3,545, 3,540, and 3,540 feet) AMSL, respectively. The top surface of the ISFSI pad is to be located at or slightly above existing grade at an approximate nominal elevation of 1,077 m (3,535 feet) AMSL. The applicant's HEC-RAS model results, as shown in SAR figure 2.4.16, "Peak PMF Depth Hydrographs," indicates that the PMF water depth at the administration building and security building is approximately 0.55 m (1.8 feet). At the CTB, identified in SAR figure 4.2.1, "ITS Classification of SSCs that Comprise the HI-STORE CIS Facility," and SAR section 5.3, "Reinforced Concrete Structures," and section 5.4, "Other SSCs Important to Safety," as a structure containing components important to safety, a few inches of flood water may accumulate. At the warehouse (the storage building appearing on SAR figure 2.1.6(c)), the water depth is a few inches. SAR figure 2.4.16 shows that within the Phase 1 ISFSI pad area, several feet of flood water may accumulate within the boundary of the ISFSI area.

The applicant cited the HI-STORM UMAX FSAR, table 2.3.1, "Loads, Criteria, Applicable Regulations, Reference Codes, and Standards for the VVM," and indicated that the UMAX system is able to withstand a maximum flood height of 38.1 m (125 feet). At the CTB, the applicant cited the reinforced concrete walls of the building, modern overhead doors, and other building penetration seals as being sufficient to prevent infiltration of flood water of a few inches into the building. Further, the applicant cited the flood water hydrograph at the CTB, appearing in SAR figure 2.4.16, and suggested that more than 36 hours of delay in flood water arrival time after the onset of the PMP provide ample time to prepare the building for an impending flood.

At the northeast corner of the VBS, the applicant's HEC-RAS model indicated a maximum PMF water depth of 2.6 m (8.4 feet) and a peak water velocity of 1.4 m/s (4.6 feet/s). Relatively high flood water velocity was also calculated for the areas surrounding the Phase 1 ISFSI pad. As shown in SAR figure 2.4.15, "Peak PMF Water Velocity," in areas around these locations, the potential for surface erosion is high. The staff's confirmatory calculations, with a continuous VBS line, also suggested that peak storm runoff velocity may exceed the erosion allowance at the southeastern corner of the VBS, along a portion of Haul Path Section 3, along a portion of the rail track inside the HI-STORE CIS Facility boundary, and within some area on the ISFS pad footprint. In SAR section 2.4.3, the applicant indicated that engineered backfill materials in the HI-STORE CIS Facility areas would likely improve erosion resistance. Additionally, the ISFSI is located in-ground with the base at a depth around 6.1 to 7.6 m (20 to 25 feet). The effect of surface erosion would not adversely impact the safety of the ISFSI. After storm waters recede, the applicant will replace any eroded materials within the facility and regrade the ground surface appropriately. Therefore, the staff found the water levels at SSCs important to safety and the potential of soil erosion within the facility boundary, as calculated by the applicant, acceptable and determined they can be used for further evaluations of PMF impact to onsite SSCs.

In conclusion, the staff reviewed the applicant's PMF analysis for the site and finds it acceptable because the staff's independent analysis resulted in flood water depths and velocities and surface erosion potentials that are higher than but comparable to those of the applicant. The staff determined that this information is acceptable to use to develop the design bases of the facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements of 10 CFR 72.90(c), 10 CFR 72.90(d), 10 CFR 72.90(f), 10 CFR 72.92(a) and (c), and 10 CFR 72.122(b).

2.3.4.3 Probable Maximum Flood on Streams and Rivers

The applicant provided a hydrologic description of the site in SAR section 2.4.1. There are no perennial streams in the vicinity of the proposed facility. The Pecos River is the nearest through-flowing surface water feature to the site. The nearest distance from the proposed facility to the Pecos River is about 42 km (26 mi). The washes to the north and west of the facility are ephemeral and not hydrologically connected to any other nearby water bodies under non-PMP conditions. The two washes will be flooded during a PMP condition, and they were accounted for in the applicant's PMF model calculations.

In SAR section 2.4.1, the applicant indicated that two nearby streams, Monument Draw (near Monument) and Antelope Draw (near Jal), are both ephemeral streams without through-flowing water in typical weather conditions. In June 1972, spot measurements by the USGS indicated a record flow of 36.2 m^3/s (1,280 cfs) (http://nwis.waterdata.usgs.gov/nm/nwis/peak/?site_no=08437620) occurred at Monument Draw. Spot measurements by the USGS also indicated a record flow of 15.0 m^3/s (530 cfs) in July 1994 for Antelope Draw (State of New Mexico Interstate Stream Commission, 2016). The locations where the USGS obtained the spot measurements of Monument Draw and Antelope Draw are approximately 27 and 58 km (17 and 36 mi) from the proposed facility, respectively.

Based on the review of the applicant's analysis and the staff's confirmatory analysis, the staff finds that PMF in the washes near the facility and stream flow in Monument Draw, Antelope Draw, and Pecos River are unlikely to impact the structures that are important to safety at the proposed site.

2.3.4.4 Potential Dam Failures (Seismically Induced)

As described in SAR section 2.4.4, "Potential Dam Failures (Seismically-Induced)," no dams are on or in the vicinity of the proposed site. The Red Bluff Dam is about 79 km (49 mi) southwest of the proposed site. The Brantley Dam and the Avalon Dam are approximately 61 km and 50 km (38 mi and 31 mi) west of the proposed site, respectively. The amount of water stored in these water bodies is unlikely to impact the site in the event of a dam failure, considering the distances from and elevations of the water bodies with respect to the proposed facility. Therefore, the applicant concluded that dam or control structure failure would have no potential impacts on the site.

Because there are no water storage, flow control, or embankment structures in the drainage area upstream of the proposed facility, the staff finds the applicant's analysis acceptable because it considered all the relevant hydrologic and hydraulic structures that could affect the site in the event of structural failure. The staff finds this information acceptable to use to develop the design bases of the facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements of 10 CFR 72.90(c), 10 CFR 72.90(d), 10 CFR 72.90(f), and 10 CFR 72.122(b).

2.3.4.5 Probable Maximum Surge and Seiche Flooding

Surge and seiche affect sites adjacent to large water bodies or the oceans. The site is adjacent to the large playa Laguna Gatuna. In SAR figure 2.4.14, "Peak PMF Water Depth," the applicant's hydrologic and hydraulic calculations indicated that the western boundary of Laguna Gatuna reached the HI-STORE CIS Facility boundary under the PMP condition. In SAR section 2.4.5, "Probable Maximum Surge and Seiche Flooding," the applicant concluded that the site is not expected to experience flooding caused by amplified seiche because the oscillation period for Laguna Gatuna from seiche and the oscillation period for wind are significantly different.

The applicant calculated the natural oscillation period of Laguna Gatuna, as shown in table E.1, "Fundamental Oscillation Period Calculations," in the applicant's PMF analysis report, to be 861 seconds, compared to 855 seconds calculated by the staff. In appendix E, "Seiche," to the 2022 GEI flood analysis report, the applicant reported a maximum wind velocity of 101.8 km/h (63.25 mph) based on two datasets obtained from Midland International Airport, Texas, and the Hobbs Lea County airport. With this wind velocity, the applicant calculated a wind wave significant height of 1.2 m (4.1 feet), as shown in table E.2, "Wind Wave Calculations," of the 2022 GEI flood analysis report, compared to the staff's estimate of 1.2 m (3.8 feet). The applicant's calculated wind wave period of 4 seconds falls in the general range of wind wave periods, from 4 to 20 seconds (Wells, 2012), suggesting the wind wave is too short to drive a resonant seiche in the longitudinal direction of a full-pool Laguna Gatuna. The independent calculation by the staff suggested a wind wave period of 3 seconds, limited by the longitudinal length of Laguna Gatuna.

The staff reviewed the applicant's calculation of probable maximum seiche flooding and found it acceptable because the significant wind wave height, the wind wave period, and the natural oscillation period of Laguna Gatuna calculated by the applicant are comparable to those independently calculated by the staff. Furthermore, the wind wave period is too short to drive a resonant seiche in Laguna Gatuna.

2.3.4.6 Probable Maximum Tsunami Flooding

The site is inland and not located adjacent to a coastal area. The Gulf Coast is about 800 km (500 mi) to the southeast. Therefore, the applicant stated that flooding attributed to seismically induced ocean waves is not applicable to the site.

The staff reviewed the applicant's discussion of probable maximum tsunami flooding in SAR section 2.4.6, "Probable Maximum Tsunami Flooding," and finds it acceptable because the proposed site is not located on or near the coast and will not be affected by this phenomenon.

2.3.4.7 Ice Flooding

Ice flooding, primarily flooding caused by ice jams, could occur under winter weather conditions. In SAR section 2.4.7, "Ice Flooding," the applicant evaluated the likelihood of ice flooding according to the three approaches identified in NUREG/CR-7046. The applicant evaluated the likelihood of temperatures near the site falling below -12°C (11°F) at around 4 percent and the proximity of the site to perennial streams or rivers with turbulent water, two factors conducive to the forming of ice dams. Finally, the applicant concluded that, because PMF is the dominant cause of potential flooding, ice flooding is unlikely to be a major flooding mechanism for the site.

The applicant also evaluated records of ice flooding events from the database maintained by the USACE and determined that ice jamming or ice dam-breaking had never occurred in the geographical area of the site.

The staff reviewed the applicant's discussion of probable maximum ice flooding and finds it acceptable because the local climate and hydrologic conditions are unlikely to support this phenomenon, and there are no prior instances of ice flooding in the geographical area of the site.

2.3.4.8 Flood Protection Requirements

In SAR section 2.4, "Floods," the applicant stated that the proposed facility is not located in an area where flooding protection is required and that the site is not designated by FEMA as a special flood hazard area. Therefore, the applicant did not propose a flood protection structure.

The staff reviewed the applicant's discussion of flood protection requirements and found it acceptable because the UMAX system was designed to sustain 38.1 m (125 feet) of water pressure, which is much higher than the PMF flood water depth at the Phase 1 ISFSI pad area. Additionally, at the CTB, which is an SSC important to safety that may be subject to a few inches of flood water, the reinforced concrete walls of the building, modern overhead doors, and other building penetration seals are sufficient to prevent infiltration of flood water into the building. Further, the PMF flood water hydrograph at the CTB suggested that there may be more than 36 hours of lead time before the arrival of a PMP flood wave to install flood mitigation measures. Along the VBS line, the applicant committed to a 1.5 m (5-foot interior) and exterior border of a material (e.g., gravel or riprap), with sufficient erosion resistance to withstand the velocity well above the erosion allowance of native soils, which will serve to eliminate any local erosion from undermining the VBS and fences. Additionally, the applicant will replace any eroded materials within the HI-STORE CIS Facility and regrade the ground surface appropriately after storm waters recede.

2.3.4.9 Environmental Acceptance of Effluents

In SAR section 4.3.1.4, "Confinement," the applicant determined that the multipurpose canisters (MPCs) of the UMAX system provide confinement for all design-basis normal, off-normal, and postulated accident conditions, and the confinement criteria for the MPCs are incorporated by reference from section 2.0.6, "Confinement," of the HI-STORM UMAX FSAR. In SAR section 9.2.1, "Storage Systems," the applicant also stated that all these conditions relevant to confinement integrity for which the canister is certified in the HI-STORM UMAX are equal to or less severe than at the HI-STORE CIS Facility. Based on the staff's review, this is consistent with the applicant's statement in SAR section 4.7, "Summary of Design Criteria," that the design criteria will ensure the confinement boundary is not breached. Section 12.1.3, "Leakage of One Seal," of the HI-STORM FW FSAR concludes that the MPC design, welding, testing, and inspection, including a redundant welded closure, meet the requirements of 10 CFR 72.236(e) such that leakage from the confinement boundary is considered not credible. In SER section 9.3.2, "Radionuclide Confinement Analysis," the staff also finds that the applicant's analysis that collapse of the CTB is not credible under normal and accident condition loads and that the HI-STAR 190 transportation packages remain leak tight while on site are acceptable. The staff finds that the HI-STORM UMAX canister storage system with MPC37 and MPC89 canisters is leak tight at the HI-STORE CIS Facility and, therefore, that the quantity of radioactive nuclides released to the environment from this storage system satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

In SAR section 2.4.9, “Environmental Acceptance of Effluents,” the applicant stated that the proposed facility will not have any radioactive waste treatment system on site. Therefore, the staff concludes that stormwater runoff on the site is unlikely to contain radiological effluents beyond the regulatory limit specified in 10 CFR Part 20, “Standards for Protection Against Radiation,” Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.” Storm water runoff from the site will be gravity drained off site, following local topography towards the two neighboring playas, Laguna Gatuna and Laguna Plata, and then into the underlying soil horizons, or it will be evaporated into the air. Since it is not credible for facility operation to result in radioactive liquid effluent, the staff finds that the application meets the requirements of 10 CFR 72.122(b)(4) precluding transport of radioactivity to a surface water body or an aquifer. The staff reviewed the applicant’s discussion of the environmental acceptance of effluents and finds it acceptable because the applicant’s design basis precludes radioactive effluents as discussed in SER section 14.3.1.

2.3.5 Subsurface Hydrology

The staff reviewed the information presented in SAR section 2.5. The applicant provided information on the regional ground water system of eastern New Mexico and southern Lea County and the nature of groundwater at the proposed site, based on initial site characterization of the bedrock and overlying unconsolidated sediments. The applicant provided supporting information in SAR section 2.6, “Geology and Seismology”; SAR section 2.4; and ER section 3.5.2, “Groundwater.” The primary information sources that the applicant used for SAR section 2.5 were “Ground-water Report 6, Geology and Ground-water Condition in Southern Lea County, New Mexico,” issued in 1961 by the New Mexico State Bureau of Mines and Mineral Resources (Nicholson and Clebsch, 1961); the 2007 GNEP siting study, and an onsite borehole program described in the 2018 GEI geotechnical data report.

This section discusses (1) regional groundwater characteristics, (2) site groundwater characteristics, and (3) contaminant transport analysis.

2.3.5.1 Regional Groundwater Characteristics

In SAR section 2.5, the applicant listed the aquifers of southern Lea County. The proposed site falls within the Querecho Plains physiographic feature, as discussed in SAR section 2.4.1, and overlies the New Mexico State Engineer’s declared Capitan Underground Water Basin. The Querecho Plains are an area of thin eolian deposits overlying variable thicknesses of alluvial unconsolidated sediments on top of the bedrock (Nicholson and Clebsch, 1961). The Capitan Underground Water Basin is named in reference to the buried Permian Capitan Reef. The State of New Mexico assumes jurisdiction over groundwater appropriation and use in the administrative groundwater basin. The applicant listed in SAR section 2.5 other aquifers overlying the Capitan Aquifer in the region as contained in the Permian Rustler Formation, Triassic Santa Rosa Sandstone, Tertiary Ogallala Formation, and Cenozoic Alluvium, in order from deep to shallow. The Cenozoic, which includes Tertiary and Quaternary sediments, are unconsolidated to semiconsolidated (Nicholson and Clebsch, 1961) except for locations where postdepositional caliche formation formed during the early Quaternary led to poorly to strongly indurated caliche horizons. In its review, the staff considered the aquifers used as a groundwater source in the region and the zones of recharge to those aquifers. The discussion below proceeds from the deepest to the shallowest aquifers.

The applicant noted in SAR section 2.5 that the Capitan Reef horizon consists of dolomites and limestones. The groundwater quality of the Capitan Aquifer was stated to be very poor due to total dissolved solids ranging from 10,065 to 165,000 milligrams per liter (mg/L). For reference, the EPA secondary drinking water standard is 500 mg/L for total dissolved solids. The applicant noted in SAR section 2.6.1, “Basic Geologic and Seismic Information,” that the Capitan Reef Complex is connected to prolific oil and gas production in eastern New Mexico.

In the 2007 GNEP siting study, section 2.4.2.2, “Groundwater at the Site,” discusses how the Rustler Aquifer falls within a formation consisting of siltstone and very fine-grained sandstone interbedded with gypsum and anhydrite. The applicant indicated, in SAR section 2.5, that the top of the Permian Rustler Formation is 270 to 340 m (900 to 1,100 feet) below the ground surface. The 2007 GNEP siting study states that the producing well closest to the site in the Rustler Aquifer is 10 km (6 mi) southwest, where the depth to the Rustler Formation is significantly shallower than at the site.

The applicant stated in SAR section 2.5 that potable groundwater is available from the Triassic Santa Rosa Sandstone, the Tertiary Ogallala, and the Cenozoic alluvium aquifers within the southern Lea County portion of the Capitan Underground Water Basin. Nicholson and Clebsch (1961) stated that virtually all the wells in southern Lea County bottom in the Triassic or younger rocks.

For the Triassic aquifer, table 3 of Nicholson and Clebsch (1961) described the Dockum Group in southern Lea County as claystones with minor sandstones and siltstones of the Chinle Formation overlying the sandstones of the Santa Rosa Formation. Nicholson and Clebsch (1961) further stated that the Santa Rosa Sandstone is the principal aquifer in the western third of southern Lea County, which is where the proposed site is located. According to Nicholson and Clebsch (1961, table 3), the Santa Rosa Aquifer yields small quantities, up to 380 L/minute (100 gpm), of groundwater with high sulfate content over much of southern Lea County. In the Querecho Plains, recharge to the Triassic aquifers may occur where permeable sandstone layers of the Chinle Formation and Santa Rosa Sandstone subcrop below the highly permeable layers of Cenozoic unconsolidated sediments (Nicholson and Clebsch, 1961). Plate 2 of Nicholson and Clebsch (1961) illustrates areas of the recharge zone to the Triassic units in the Querecho Plains as mounded contours of potentiometric or water table elevations. The staff notes that the area in the vicinity of the playas surrounding the proposed site is a depression in the potentiometric or water table surfaces on plate 2, which indicates that the Triassic sandstone units are not likely recharged in this area.

The Cenozoic alluvial and eolian sediments overlying the bedrock in southern Lea County are highly variable in thickness and generally contain a postdepositional early Quaternary caliche horizon, according to Nicholson and Clebsch (1961). Within Lea County, the applicant stated in SAR section 2.4.1, that groundwater is provided primarily by the Ogallala Aquifer of the southern High Plains (Llano Estacado). However, the Ogallala Aquifer is limited to eastern Lea County and does not extend southwestward into the Querecho Plains, according to Nicholson and Clebsch (1961). In SAR section 2.6.1, the applicant stated that the Mescalero Ridge delineates the southwestern extent of the southern High Plains, and thus the closest occurrence of the Tertiary Ogallala Aquifer is approximately 19 km (12 mi) northeast of the proposed site. According to the 2007 GNEP siting study, the alluvial sediments of the Tertiary Ogallala Formation overlap in time with those of the Gatuna Formation in the Querecho Plains and further west, although the relationship and presence of Gatuna and Ogallala alluvial sands are complex, according to J.W. Hawley's "The Ogallala and Gatuna Formations in the southeastern New Mexico region," issued in 1993. In SAR section 2.6.1, the applicant concluded that the late Tertiary alluvial sediments in the vicinity of the proposed site are from the Gatuna Formation. Shallow groundwater in the Querecho Plains occurs in the generally unconsolidated Tertiary and Quaternary alluvium and eolian deposits, according to Nicholson and Clebsch (1961) and the 2007 GNEP siting study, although they indicated that the saturated portions may be discontinuous and areas may be unsaturated. In the vicinity of the lagunas area of the Querecho Plains, the discontinuous nature of saturation in the alluvial sediments overlying the bedrock is illustrated in figure 2.4.2.2-3, "Shallow Groundwater Map," of the 2007 GNEP siting study. The saturated thicknesses in the sediments were shown to not exceed 20 feet (6.1 m) thick and are noted as a source of groundwater for livestock.

In SAR section 2.5, the applicant indicated that potable groundwater is available from Triassic, Tertiary, and Quaternary units in southern Lea County, but no potable groundwater exists in the immediate vicinity of the proposed site. The applicant described the deeper groundwater, in the Permian aquifers in southern Lea County, as not potable. Nicholson and Clebsch (1961) noted that shallow groundwater occurs in the Cenozoic alluvium of the Querecho Plains, including the area surrounding the site, but the applicant stated that it is of poor quality due to brine discharges from potash refining or oil and gas production. Potable water for the site is obtained from pipelines that convey water from the Ogallala High Plains Aquifer in eastern Lea County.

The NRC staff reviewed the applicant's discussion of regional subsurface hydrological characteristics and finds it acceptable because the applicant used the general literature and reports from the USGS in collaboration with the State of New Mexico. The staff concludes that the information in the SAR is consistent with the primary source documents cited in the SAR and with other external published literature. The staff determined that the application's description of the regional groundwater characteristics provides context for the site groundwater characteristics described in the next subsection and is acceptable to use to develop the design bases of the facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements in 10 CFR 72.98(c) and 10 CFR 72.122(b).

2.3.5.2 Site Groundwater Characteristics

In SAR section 2.5, the applicant described the groundwater conditions found at the site primarily based on borehole data from the 2007 GNEP siting study drilling program and the applicant's more recent drilling program performed in support of characterization and reported in the 2018 GEI geotechnical data report. The applicant described (1) unsaturated conditions within the shallow Tertiary and Quaternary unconsolidated sediment layers, (2) local saturated horizons in the uppermost bedrock of the Chinle layer (Tecovas Formation), and (3) saturated conditions in the underlying Santa Rosa Sandstone except for the unsaturated uppermost portion. Additionally, the applicant stated that evapotranspiration was five times the precipitation rate, indicating to the applicant that little infiltration to the subsurface occurs at the site. The applicant indicated that the HI-STORE CIS Facility would not impact the groundwater system because excavation during construction of the facility would not reach the saturated groundwater and "there are no radioactive effluents from the proposed spent fuel storage system."

In SAR section 2.6.1, the applicant described the lithological units below the site, from uppermost to lower lithologies, as supplemented by information in the 2018 GEI geotechnical data report. The staff separated the five lithologic horizons into two groups:

(1) shallow unconsolidated sediments, generally 0 to 7.6 m (0 to 25 feet) below grade

- thin eolian surface layer, well-drained, nonindurated clayey sand with gravel and lean clay with sand
- Quaternary Mescalero caliche, poorly indurated silty sand, sand, and gravel
- Tertiary Gatuna Formation, also referred to as a residual soil layer, poorly indurated clayey sand, sandy lean clay, clayey sand with gravel

(2) uppermost bedrock of Triassic Dockum Group, greater than 7.6 m (25 feet) below grade

- Chinle layer (Tecovas Formation) poorly indurated clay shale, siltstone, clayey sandstone, poorly indurated at bedrock interface but indurated at depth
- Santa Rosa Formation mudstone and sandstone

The applicant based lithologic descriptions, depths, elevations, and unit thicknesses on nine borehole locations from the 2018 GEI geotechnical data report and two borehole locations from the 2007 GNEP siting study. The staff subdivided the five subterranean lithologies listed by the applicant in SAR section 2.6.1 into two hydrogeological units to support the discussion in the remainder of this SER section. In addition, the staff will use the term shallow unconsolidated sediments to encompass the applicant's eolian, caliche, and residual sediment layers of Tertiary and Quaternary age.

In SAR section 2.5, the applicant described locally present near-surface water levels at depths of 11 to 15 m (35 to 50 feet) based on two boreholes (ELEA-2 and B107), and a primary aquifer at a depth of 77.1 to 80.1 m (253 to 263 feet) based on one borehole (B101). For the near-surface groundwater, the applicant provided water level measurements from October and November of 2017 for monitoring well B017 and from November 2017 for monitoring well ELEA-2. For the primary aquifer, the applicant provided water level measurements from the same period for monitoring well B101. No water was found in borehole B106 after well installation was completed down into the Chinle Formation.

According to the borehole data in the 2018 GEI geotechnical data report, the locally present near-surface saturated horizon is in the mudstone (indurated) and clay portions (unconsolidated) of the Chinle layer. The applicant indicated that shallow groundwater was encountered on the eastern side of the site and was described as brackish due to the proximity to playas and likely association with solution discharges related to potash mining and oil and gas production before 2001 (ER section 3.5.2.1, "Regional Groundwater"). The applicant stated that shallow groundwater was not encountered on the west side of the site but provided SAR figure 2.5.2, "Water Wells and Piezometer Locations," that shows the presence of shallow saturated groundwater in alluvial sediments in wells near the site. The interface between the clay of the Chinle layer and the unconsolidated Tertiary sand and gravels corresponds approximately with the 10.7 m (35-foot) near-surface water level depths, which would be at an elevation of approximately 1,070 m (3,500 feet) AMSL, based on borehole data in the 2018 GEI geotechnical data report. For the deeper saturated horizon, the applicant indicated that the primary aquifer is associated with the regional Santa Rosa Sandstone aquifer. Based on two boreholes (B101 and B105) at the site, the top of the Santa Rosa Sandstone is at 1,021 m (3,321 feet) AMSL, or approximately 15 m (50 feet) above the water level found in monitoring well B101. The second borehole (B105) only reached the upper 2.4 m (8 feet) of the Santa Rosa and did not exhibit saturated conditions.

The applicant described the groundwater quality in the vicinity of the site as poor because of high total dissolved solids. In SAR section 2.1, the applicant stated that there were no water wells located on the site. In SAR section 2.5, the applicant stated that no potable groundwater was known to exist in the vicinity of the site. The applicant provided a map with all the wells near the site (SAR figure 2.5.2), with the wells delineated by use and aquifer tapped. Most of the wells are in the shallow alluvial aquifer and are categorized as windmill, livestock, or abandoned or not in use, or by water quality (e.g., brine). The applicant stated that potable water for the site will be obtained from the local potash mining companies that have water conveyed to the area though pipes.

From the applicant's description of subsurface hydrological conditions, the conceptual site model for flow at the site consists of (1) little to no recharge into unsaturated shallow unconsolidated sediment layers, (2) local presence of saturation in the clay layer of the Chinle tied to industrial brine discharge in and near playas at the interface with alluvial sediments, and (3) little to no downward flow at the site through the low-permeability clay and shale bedrock units to underlying aquifers. The staff notes that the latter infers that recharge to aquifers below the site occurs in other areas of Lea County and beyond, and not at the site.

<u>Review of Hydrogeology of the Unconsolidated Sediments</u>

The staff evaluated the applicant's conclusion that the shallow unconsolidated sediments at the site are unsaturated, including at the interface with the Chinle clays and mudstones. In the evaluation, the staff considered hydrogeologic features and hydrologic conditions in the vicinity of the site and general hydrologic processes. The staff also considered the limited area and number of boreholes used to characterize the site.

The staff considered the conceptual site model for flow in the unconsolidated sediments as a prelude for evaluating the potential effects on the engineered systems on the groundwater system. In the conceptual site model, generally vertical flow from any infiltration and recharge in the unconsolidated sediments is redirected sub-horizontally along the bedrock interface towards the playas. Based on the lithologies and the near-horizontal layering, lateral flow would be small in a saturated Chinle clay layer due to its low permeability. Lateral flow rates towards the playas would be much faster in the more permeable Tertiary and Quaternary alluvial sediments. In the conceptual site model, the upper bedrock clays and mudstone of the Chinle layer act both to redirect vertical recharge to sub-horizontal along the bedrock interface and as an aquitard to separate the near-surface groundwater from the aquifers at depth in the Dockum Group and below.

To provide context to the shallow groundwater, the staff notes that the elevation of the site is 1,073 to 1,079 m (3,520 to 3,540 feet) AMSL. The site will be excavated a maximum of 7.6 m (25 feet), generally leaving a residual soil layer on which to construct the foundation pad, subgrade, and undergrade of the UMAX system. The thickness of the shallow unconsolidated sediments (residual, caliche, and eolian layers) varies from 8.41 to 12.2 m (27.6 to 40.6 feet), with an average of 10.2 m (33.5 feet) for the seven boreholes in the 2018 GEI geotechnical data report. The bedrock below the residual soil layer is relatively horizontal with the interface varying from 1,066 to 1,067 m (3,499 to 3,502 feet) AMSL, based on the boreholes in the 2018 GEI geotechnical data report, mostly in the area of the proposed Phase 1 ISFSI pad, according to SAR figure 2.1.7. The staff notes that the Phase 1 pad is a small portion of the site. In SAR section 2.4.1, the applicant stated that the bottom of the Laguna Gatuna is at an elevation of 1,065 m (3,495 feet) AMSL.

Several aspects of site information bear on the possibility and likely amount, distribution, and movement of groundwater in the shallow sediments at the site:

- presence surrounding the site of saturated groundwater in shallow unconsolidated sediments, including in shallow wells and springs
- bedrock interface below the site compared to Laguna Gatuna playa elevation
- potential for infiltration and recharge at the site

For the first aspect, the staff considered the potential for saturated groundwater conditions in the shallow unconsolidated sediments at the site, based on the presence of wells and springs surrounding it. The 2007 GNEP siting study provided two figures with water levels in wells tapping groundwater from the shallow unconsolidated sediments. The first, figure 2.4.2.2-1, "Water Wells and Piezometer Locations," provided the water levels, use, and status of each well. This figure is reproduced as SAR figure 2.5.2. The second, figure 2.4.2.2-2, "Piezometric Surface of Water in Triassic Units in the Area of the Site," provided water level contours of the shallow aquifer in the unconsolidated sediments. The limit of the shallow groundwater saturation in the alluvium abuts the ELEA site boundary used in the 2007 GNEP siting study. The labels on the wells include windmill, stock, not used, abandoned, and brine. In addition to the shallow wells, several springs have been identified near or on the site that may be related to a shallow aquifer. Figure 2.3.2.1-4, "Oilfield Disposal Sites and Impact Areas," in the 2007 GNEP siting study identified two springs south of south and southwest of the Laguna Gatuna between the ELEA site boundary and the highway. The springs are at an elevation of approximately 1,070 m (3,500 feet) AMSL, based on a comparison of the location on the figure and topographic maps. The GNEP siting study also identified three brine seeps on the north side of the main drainage on the site to the Laguna Gatuna at an elevation of approximately 1,070 m (3,500 feet) AMSL. However, these springs are said to be related to industrial discharge and may not indicate the presence of a natural shallow saturation horizon. ER figure 3.5.8, "Seep at Laguna Plata, UTM 3605874N 618430.1E (NAD 83)," identified the coordinates for a seep to the northwest of the site, towards Laguna Plata. Based on topographic maps, the ground surface at the specified location is approximately 1,053 m (3,455 feet) AMSL. The bedrock interface elevation at this location is not known, but the elevation is approximately 14 m (45 feet) lower than at the Phase 1 pad area on the site. As a caveat, the staff notes that the GNEP siting study observations in wells and springs were made before 2007 and, as such, may not reflect 2017 conditions for correspondence with the 2017 borehole development. The staff notes that the 2007 GNEP siting study sampling of groundwater from springs, the ELEA-2 well, and Laguna Gatuna occurred at a time when precipitation was heavy enough to fill the laguna. Standing water in Laguna Gatuna is not the typical condition. In the "Approved Jurisdictional Determination Form (Interim) Navigable Waters Protection Rule," issued in 2021, the USACE (2021c) noted that a megadrought not seen for several centuries was occurring in several western states, including New Mexico; the last 20 years ranked as the second-driest period in the past 1,200 years. Regardless, based on the distribution of wells in SAR figure 2.5.2 and springs or seeps near the site, the staff cannot discount the possibility of saturated horizons in the shallow unconsolidated sediments above the bedrock interface.

The staff considered the topology of the bedrock interface with the shallow unconsolidated sediments to assess the possibility of a local saturated zone in the shallow unconsolidated sediments. Regionally, Nicholson and Clebsch (1961) indicated that infiltration and recharge to the shallow aquifer in the Querecho Plains was sufficient to sustain the shallow aquifer but that there are locations where the shallow alluvial sediments are not saturated. The staff notes that two possible features of the shallow hydrogeology are plausible. For one, Nicholson and Clebsch (1961) indicate that at locations where the Santa Rosa Sandstone subcrops, it is sufficiently permeable to accept most of the water that infiltrates through the shallow unconsolidated sediments, thus preventing the buildup of saturation in the shallow unconsolidated sediments. However, based on borehole data in the 2018 GEI geotechnical data report and the cross-sections presented in SAR figures 2.5.3, 2.5.4, and 2.5.5, the Santa Rosa Sandstone does not subcrop at the site. Two, the bedrock interface may not be flat but rather may be represented by a paleotopographic surface with channels and depressions. The wells immediately south of the site shown in SAR figure 2.5.2 and 2007 GNEP siting study figure 2.4.2.2-2 exhibit the total depths of the wells completed in Quaternary alluvium (assumed to be bedrock interface) and saturated thicknesses that suggest an uneven bedrock interface. The line of wells south of the site shown in SAR figure 2.5.2 may reflect a channel or lens of saturation, especially considering that wells are often biased to locations with the most readily available groundwater. The chain of wells immediately south of the site may reflect a paleosurface channel at the bedrock interface, although the limit of saturation lines to the east and to the west of these six wells is not constrained.

Based on the staff’s assessment of the wells surrounding the site and observations on the paleotopography of the bedrock interface, the staff cannot discount the possible presence of channelized saturated horizons intersecting the site. With the focus of boreholes for characterization mostly limited to the Phase 1 pad area, the remainder of the site remains mostly uncharacterized. Considering this uncertainty, the staff discusses the consequences of possible discontinuous or channelized saturated zones in the shallow unconsolidated sediments in “Review of Hydrogeological Interaction with the Engineered System,” below.

For the second aspect listed above, the staff evaluated the relationship between the bedrock interface below the site and the playa. In SAR section 2.4.3 and figure 2.4.13, “Laguna Extreme Full Conditions for PMP/PMF,” the applicant assigned a normal full level for Laguna Gatuna of 1,070 m (3,500 feet) AMSL, or 1.5 m (5 feet) above the bottom of the playa. The normal full level was set to the salt line, which is a line of evaporation minerals reflecting the historical high-water mark. The 2022 GEI flooding analysis determined the salt line from observations of aerial imagery and topography. The staff notes that the top of the Chinle clay crops out near the Laguna Gatuna shoreline at approximately 1,070 m (3,500 feet) AMSL, which is approximately the elevation of the salt line delineated in the 2022 GEI flooding analysis report. The salt line could also be the location of seepage along the shale bedrock interface. Because the elevation of the bedrock interface with the shallow unconsolidated sediments at the site is close to the elevation of the Laguna Gatuna, saturation at the interface will be influenced during periods when the laguna contains water from precipitation events.

For the third aspect listed above, the staff evaluated the potential for infiltration and recharge at the site. The staff considered the applicant's comparison of evapotranspiration to precipitation in SAR sections 2.4 and 2.5 to indicate that little to no infiltration and recharge occur at the site. However, the staff notes that potential evaporation estimates cannot be compared directly to precipitation to determine whether or how much recharge occurs. This comparison ignores the fact that evapotranspiration and precipitation do not occur at the same time. In semiarid areas of the western Unites States, runoff and evapotranspiration significantly reduce the amount of precipitation that results in groundwater recharge, which is infiltrating water that percolates below the rooting zone. For the semiarid climates of the desert southwest, the seasonal and annual variations in precipitation, in combination with the near-surface hydrogeology, determine the amount of infiltration that escapes the rooting zone and continues downward, becoming recharge. Bedinger (1987) summarized the results of watershed-scale analyses across the arid western United States to span a range of 0.01 to 25 percent of precipitation that becomes recharge, depending on local conditions and hydrogeology. Areas with clayey soils and extensive vegetation fall at the lower end, and areas with sandy and gravelly soils and sparse vegetation fall at the upper end of the recharge range. Annual average recharge over the long term for environments in the western United States typically ranges from 1 to 4 percent of precipitation. The typical hydrological regime in semiarid areas of the western United States is for infiltration to be highly variable year to year with a prominent seasonal difference (Meixner et al., 2016; Hogan et al., 2013; Stonestrom et al., 2007; Scanlon et al., 2005). In semiarid areas, significant annual recharge generally occurs during winter and spring of years with El Niño weather patterns. In these areas, significant recharge passing below the root zone may not occur every year. Wet winters, especially during El Niño years, with long duration but low precipitation intensities from frontal weather systems generally lead to more recharge than that associated with high-intensity convective summer weather systems when the runoff percentage is large. Considering typical processes for desert southwest recharge discussed above, the presence of well-drained surficial soils with low water-holding capacity (stated in the USACE (2021c) jurisdictional determination), the staff expects that recharge at the site is some small percentage of precipitation and not little to no recharge, as stated by the applicant based on the borehole observations documented in the 2018 GEI geotechnical data report.

The staff considered how the small depressions with prominent changes in vegetation reflected on near-surface hydrologic processes. Because these depressions have restricted drainage outlets, they intermittently hold small quantities of water from both distributed precipitation and runon from upgradient areas. The well-vegetated (relative to other areas) depressions occur on and in the vicinity of the site based on satellite imagery, including that shown in SAR figure 2.6.17, "Circular Depression Features Near the Site (Buffalo Wallows)." The density of the small depressions is low over the Phase 1 pad area and increases significantly in areas surrounding the site. In SAR section 2.6.4, "Stability of Subsurface Materials," the applicant suggested that the depressions are associated with leaching of the caliche and subsequent removal of loosened material through eolian processes. The small, circular depressions are distinct from the large, saline playas such as Laguna Gatuna where vegetation is absent. The staff notes that the small depressions are sometimes isolated and sometimes aligned along incipient drainages. Regardless of how the vegetated small depressions formed, their features reflect on hydrologic processes across the site. Runoff and infiltration on the depression provide a sufficient amount of stored groundwater to sustain the increased density of vegetation. The staff suggests that the sediments in the depressions are finer grained compared to the eolian sands blanketing the remainder of the site. Finer grained sediments reflect a larger water-holding capacity compared to the eolian sands elsewhere. While the vegetated depressions show that infiltration occurs on the depressions, the eolian sand elsewhere on the site (i.e., between depressions) would more readily accept a higher infiltration rate. For cover areas between the small depressions, the well-drained eolian soils have a lower water-holding capacity (USACE, 2021a), which, during precipitation events, would allow deeper propagation of wetting fronts compared to the finer grained soils of the depressions.

Based on the discussion above, the NRC staff considers the site conceptual model for shallow groundwater flow to potentially include laterally discontinuous or transient saturation. This conclusion is based on (1) shallow aquifer wells surrounding the site, (2) springs or seeps near the site, (3) permeability and low water capacity of surficial soils that may be conducive to seasonal or annual net infiltration periods leading to recharge, (4) a subsurface stable supply of water for vegetation in depressions enhanced by runoff, (5) distributed or focused infiltration through eolian sands in non-depression areas, and (6) the potential interaction of water in Laguna Gatuna during strong precipitation events due to the correspondence of the bedrock interface below the site with the playa. The staff could not discount the possibility of saturated portions of the shallow unconsolidated sediments spatially discontinuous on the paleosurface of the bedrock interface and possibly transient in nature. Therefore, the consequence of intermittent and spatially variable saturation above the bedrock interface is discussed below under “Review of Hydrogeological Interaction with the Engineered System.” The applicant stated that a groundwater monitoring network would be implemented before facility construction to obtain baseline information for the groundwater system.

Review of Hydrogeology of Bedrock

The staff considered the hydraulic connection between the near-surface groundwater and the uppermost primary aquifer at the site, which is in the Santa Rosa Sandstone. Without a hydraulic connection, natural recharge or releases from the facility activities would not reach the primary aquifer below the site. SAR section 2.5 describes the primary groundwater table depth as approximately 77.1 to 80.2 m (253 to 263 feet) below the ground surface at the site, based on measurements in well B101. The presence of isolated pockets of saturated horizons in sandy or fractured bedrock horizons in discontinuous aquifers above the lower permeability zones in the Chinle layer were identified in wells B107 and ELEA-2.

Based on the presence of an unsaturated zone in the upper Santa Rosa Sandstone identified in two boreholes, the staff concludes that the low-permeability clays and mudstones of the overlying Chinle Formation act as an effective hydraulic barrier between the near-surface saturated horizons and the primary aquifer in the Santa Rosa Sandstone.

Review of Hydrogeological Interaction with the Engineered System

The staff evaluated both the potential for groundwater to impact the cavity enclosure containers (CECs) in the ISFSI pads, and the potential for groundwater to impact the under-grade of the storage system and cause changes to differential settlement.

The elevation of the site is 1,073 to 1,079 m (3,520 to 3,540 feet) AMSL. The site will be excavated a maximum of 25 feet (7.6 m), leaving a thin residual soil layer on which to construct the foundation pad and subgrade of the UMAX system. The bedrock below the residual soil layer is flat-lying clay and mudstone of the Chinle Formation, the top of which is at approximately 1,070 m (3,500 feet) elevation. The applicant planned to excavate the (1) ISFSI and support foundation pads (SFPs), subgrade, and under-grade as illustrated in SAR figure 4.3.1, “Sub-Grade and Under-Grade Space Nomenclature,” and (2) canister transfer facility (CTF) in the CTB. For the ISFSI, SAR figure 4.3.1 shows a cement SFP, and the ISFSI pad will be separated by a subgrade layer described as controlled low-strength material (CLSM). SAR section 1.2.2, “Constituents of the HI-STORM UMAX Vertical Ventilated Module and ISFSI Structures,” described the CLSM as a cementitious material that will be used as backfill. Alternatively, the applicant indicated that a lean concrete may be used, which is further described as flowable fill, unshrinkable fill, controlled density fill, flowable mortar, flowable fly ash, fly ash slurry, plastic soil-cement and soil-cement slurry. In SAR section 2.6.6, “Construction Excavation,” the applicant indicated that the excavated caliche and residual (“soil”) unconsolidated layers will be used as backfill for the lateral subgrade Space B between the ISFSI pads and for the under-grade Spaces C and D. The applicant stated in SAR table 4.3.3, “Applicable Earthquake and Long Term Settlement data for the Certified HI-STORM UMAX System and the HI-STORE CIS Facility,” that subgrade Spaces B, C, and D will be packed to a specified density.

In SAR section 2.8, the applicant stated that the water table was sufficiently below the subterranean HI-STORM UMAX system to preclude the possibility of intrusion into the storage cavity. The staff notes that the CEC will be placed on a concrete SFP on top of the thick undergrade and backfilled with CLSM, which as a lean concrete would have low permeability. The backfill of Space B will surround the emplacement and will overlie Space D. The applicant plans to use the same residual and caliche soil for backfill in Spaces B and D; hence, no perching should occur. The staff notes that groundwater flow from the shallow unconsolidated sediments laterally into the subgrade is unlikely because no indication of perching or lateral flow of groundwater has been observed or is expected in the upper portion of the shallow unconsolidated sediments at the elevation of the subgrade. No laterally extensive low-permeability layers in the shallow unconsolidated sediments were identified in boreholes at the site that could cause perching at the elevations of the ISFSI pads. Furthermore, the applicant stated that the CEC has no penetrations or openings; thus groundwater has no path into the interior of the CEC.

The applicant also stated in SAR section 2.6.1 that the excavation to a depth of 7.6 m (25 feet) below grade will not come into contact with the groundwater table. Because the staff could not discount the possibility of discontinuous and transient saturated groundwater in the lower portion of the shallow unconsolidated sediments above the bedrock interface, it considered the effect of potentially spatially variable or transient saturated conditions on differential settlement of the ISFSI system. The applicant expects that a thin layer of residual soil also will remain below the under-grade of the ISFSI pad and below the CTB. In SAR section 4.3, “Design Criteria for SSCs Important to Safety,” and section 4.6, “Design Criteria for the Cask Transfer Building (CTB),” the applicant described long-term differential settlement analyses. In SER section 2.3.6.4, the staff reviewed the applicant’s assessment and found the analyses adequately encompassed the effect of variable saturation on differential settlement.

<u>Summary</u>

The staff reviewed the application and supporting documents for site characteristics and finds them acceptable because the groundwater characteristics are adequately described and are based on site-specific observations and data. In “Review of Hydrogeology of the Unconsolidated Sediments,” the staff could not discount the possible presence of saturated groundwater in the shallow unconsolidated sediments, based on its conceptual site model and observations from areas surrounding the site. However, in “Review of Hydrogeological Interaction with the Engineered System”, the staff determined that there would be no consequence to the performance of the facility from groundwater intrusion into the engineered system or groundwater affecting differential settlement of the subgrade and under-grade of the ISFSI structure. In addition, in “Review of Hydrogeology of the Unconsolidated Sediments,” the staff noted that limited data were available on the saturation state of the shallow unconsolidated sediments for a large portion of the site. However, the applicant stated, in SAR section 2.5, that baseline groundwater monitoring, sampling, and testing will be performed before construction of the facility to establish baseline measurements.

Based on the unsaturated conditions above the primary aquifer at the site in the Santa Rosa Sandstone, the staff concluded in “Review of Hydrogeology of Bedrock” that near-surface groundwater in contact with the facility was not in hydraulic connection with the Santa Rosa Aquifer. The staff concluded in SER section “Regional Groundwater Characteristics” that groundwater at the site was not hydraulically connected to the Ogallala Aquifer in eastern Lea County because the Ogallala Formation is not present in the vicinity of the site.

The staff determines that this information is acceptable to use to develop the design bases of the HI-STORE CIS Facility, to perform additional safety analyses, and to demonstrate compliance with the regulatory requirements in 10 CFR 72.98(c) and 10 CFR 72.122(b).

2.3.5.3 Contaminant Transport Analysis

In SAR section 2.5, the applicant stated that no radioactive effluents will be released from the proposed spent fuel storage system and that there was no usable groundwater at the site. Consequently, the applicant did not provide a delineation and evaluation of contaminant pathways were there to be releases to the subsurface from facility activities, as described in NUREG-1567 guidance in support of 10 CFR 72.122.

Together with the information provided on the subsurface hydrology, the applicant stated that no buildup of radionuclides will occur in the subsurface hydrological system. The information the applicant provided described the lack of groundwater of sufficient quality and quantity for any use in the vicinity of the proposed site or substantially connected hydraulically to groundwater at the site. The applicant stated that there were no saturated horizons found in the Tertiary and Quaternary unconsolidated sediments based on several boreholes developed for site characterization. The 2007 GNEP siting study indicated that a saturated zone is unlikely in the shallow alluvial sediments beneath the site, the water in surrounding areas is a brine, and there are no users. For deeper groundwater, several aquifers occur at depth in southern Lea County. At the proposed site, the applicant identified the primary uppermost groundwater at a depth of 77 to 80 m (253 to 263 feet) below the ground surface, with overlying relatively impermeable shales and mudstones. At the proposed site, several intervening low-permeability layers would likely preclude significant recharge or a hydrologic transport connection between the ground surface and deeper aquifers.

Whereas the NRC staff could not preclude the possible presence of spatially and temporally variable saturated horizons in Tertiary and Quaternary unconsolidated sediments, such as at the bedrock interface, the applicant stated in SAR section 2.5 that the HI-STORE CIS Facility will not have any liquid effluents, including for normal, off-normal, and accident conditions, as indicated in SAR chapter 9, “Confinement Evaluation,” and chapter 15, “Accident Analysis.” Therefore, in combination with the unlikely hydrologic transport connection with potable groundwater for potential users, a description and analyses of radionuclide transport to the nearest downstream groundwater user are not needed.

The applicant stated that the SAR did not include a hydrologic transport analysis because there are no radioactive effluents from the proposed HI-STORE CIS Facility. Based on the evaluations in SER chapter 9 and chapter 15, the staff finds the omission of the contaminant transport analysis acceptable for compliance with 10 CFR 72.122, because the facility and cask designs are expected to preclude the release of effluents for normal, off-normal, and credible accident conditions.

2.3.6 Geology and Seismology

SAR section 2.6 describes the geological and seismological setting of the proposed site, geographically located in Lea County, New Mexico. This review corresponds to SAR section 2.6.1; section 2.6.2, "Vibratory Ground Motion;" section 2.6.3, "Surface Faulting;" section 2.6.4; and section 2.6.5, "Slope Stability." The review considers the applicant's SAR, responses to RAIs, and the staff's in-person observations at the proposed site and in its vicinity.

The geology and seismology sections of the HI-STORE CIS Facility application fall under 10 CFR 72.103(a)(1) because the application is dated after October 16, 2003, and the site lies east of the Rocky Mountain Front (approximately 104 west longitude); the site is located at 32.583 north latitude and 103.708 west longitude.

2.3.6.1 Basic Geologic and Seismic Information

The staff reviewed the basic geologic and seismic characteristics of the site and vicinity as presented in SAR section 2.6.1. These include a brief discussion of the near-surface stratigraphy and the geologic history of the area, to include the origin of some structural features in the site region. The staff also reviewed relevant literature cited in the SAR and supporting documentation provided in the 2018 GEI geotechnical data report and ER section 3.3, "Geology and Soils."

The 2018 GEI geotechnical data report documents the boring program performed to confirm site-specific subsurface conditions. The site subsurface conditions were explored with nine borings and six offset borings; rock core was recovered in six of the borings to depths ranging from 30 to 120 m (100 to 400 feet). The applicant installed monitoring wells in three of the offset borings to measure groundwater elevation and inclinometers in the three other offset borings for crosshole seismic testing in soil and bedrock.

SAR section 2.6.1 describes the geology of the site and its surrounding areas. The applicant presented the geology in the vicinity of the Holtec site in SAR figure 2.6.3, "Surficial Geology in the Vicinity of the Site." SAR figure 2.6.1, "Major Regional Geological Structures near the Site," shows the location of the Holtec site in relation to the regional geologic structures, including the Delaware Basin, Capitan Reef, and Central Basin Platform. The applicant described the Delaware Basin as a southward-lunging, asymmetrical trough with a thickness of more than 6,100 m (20,000 feet) from the Precambrian basement rocks to the surface. The Delaware Basin, along with the Midland and Val Verde basins, and the Central Basin Platform collectively form the Permian Basin, a regional structure in west Texas and southeastern New Mexico known for its oil and gas production.

The applicant identified the stratigraphic units encountered in the site subsurface from the Permian-age sediments in the basin to the surficial alluvium. The subsurface stratigraphy of the Holtec site above the basement rocks displays a history of formation in an inland basin, erosion and submergence by a shallow sea during the Cretaceous period, and then uplift and erosion in the early Tertiary period.[1] SAR figure 2.6.5, “Stratigraphy of the Delaware Basin,” provides the generalized stratigraphic column for the Delaware Basin. The discussion of the near-surface geology in SAR section 2.6.1 focuses on the units encountered in the borings at the site from the Triassic age to present. SAR section 2.6.1 states that most boreholes penetrate a topsoil layer, caliche layer, residual layer, Chinle layer, and Santa Rosa layer. The applicant also stated that the near-surface layers consisted of surface soil, Mescalero caliche, Quaternary sands, and Dockum Group material in the uppermost 7.6 m (25 feet) at the site. SAR figure 2.6.5 shows the uppermost layers at the site as Quaternary Pediments, Valley Fills, Upper Gatuna Formation, Tertiary Lower Gatuna Formation, and Ogallala, all of which overlie the Triassic Dockum Group. Although SAR figures 2.6.2 and 2.6.5 list formation names, as noted below, the applicant also provided detailed descriptions of the composition of the individual units.

The staff consulted regional geologic maps produced by the USGS as part of its review of the Holtec application and noted that although the applicant divided the Dockum Group into the Chinle Formation and Santa Rosa Sandstone, some USGS geologic maps subdivide the Dockum Group into the Chinle, Trujillo, and Tecovas Formations overlying the Santa Rosa Sandstone, each with a distinct composition. Additionally, a geology report for a nearby site, “Section VI, Geology Report,” by Waste Control Specialists (WCS), issued February 2004, states that some strata that were previously identified as Chinle Formation have been more recently identified as Cooper Canyon Formation. The report divides the Chinle Formation into the Cooper Canyon and Trujillo and Tecovas Formations, all of which overlie the Santa Rosa Sandstone.

In its response to RAI 2-28, dated September 16, 2020, the applicant provided additional information to clarify the stratigraphic units present in the subsurface of the site, including their correlative units at a regional scale. The applicant clarified that the site subsurface consists of Eolian sand deposits, Mescalero caliche, Gatuna Formation, and Dockum Group members. The applicant provided an updated stratigraphic column for the observed lithology at the site in SAR figure 2.6.5. The applicant also clarified the recognized subdivisions of the Dockum Group present in the site subsurface, including the depths to each of the subunits. The NRC staff reviewed the applicant’s response to RAI 2-28 and determined that the information provided was sufficient to characterize the subsurface units at the site within the context of the regional geology.

The boring logs provided in the 2018 GEI geotechnical data report describe the presence of numerous areas of fractured rock and several occurrences of slickensides. SAR section 2.6.1 describes the subsurface materials but does not mention fractures or slickensides observed in the rock units at the site. SAR section 2.6.3 does not address any faults or other mechanisms of deformation that may have fractured the subsurface materials at the Holtec site. The 2018 geotechnical data report also does not include the observed fractures or slickensides in the description of the materials encountered in the borings. As described below, the applicant also

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[1] The NRC staff recognizes that the term “Tertiary period” is no longer used to describe the period of geologic time between the end of the Cretaceous (66 million years ago) and start of the Quaternary (2.6 million years ago). However, because the applicant included the Tertiary in its application materials, the NRC staff is using similar nomenclature in this SER.

provided information describing the origin of these fractures and slickensides and assessed the potential for surface deformation or settlement of the storage pads and the CTF associated with the mechanism that formed the fractures and slickensides.

In its response to RAI 2-29, dated September 16, 2020, the applicant provided full-color photos of the recovered core from the site investigations and clarified that some missing sections were removed for laboratory testing before the core box was photographed. The applicant clarified that the fractures observed in the core likely originate from the relieving of pressures and stresses following the Triassic period. The fractures are due to regional geologic processes that are unlikely to cause localized deformation during the licensed life of the facility and are not due to recent changes in the stress regime for the site. Finally, the applicant clarified that the observed slickensides were pedogenic in origin and therefore developed during deposition and induration of the materials and did not result from tectonic processes. The applicant updated SAR section 2.6.1 to provide additional clarifying information on regional geologic nomenclature and the observed fractures and slickensides.

The NRC staff reviewed the applicant’s response and, during the June 2019 site visit, observed the fractures and slickensides in the recovered core. Based on this information, the staff determined that the fractures are not due to recent changes in the regional stress regime and the slickensides are pedogenic in origin. Accordingly, the staff concludes that the fractures and slickensides are not tectonic in origin and will not affect the facility or result in surface deformation at the site.

SAR figure 2.6.2 shows the site is underlain by over 300 m (1,000 feet) of the Capitan Reef, a carbonate formation in the site region that is well known for large-scale karst features that have developed at the surface and at depth. Most notable of these features are the Carlsbad Caverns, approximately 105 km (65 mi) southwest of the site. The staff also notes the presence of additional carbonate and evaporite rocks both above and below the Capitan Reef and the association of subsurface halite dissolution above the Capitan Reef with several sinkhole features in southeastern New Mexico. Additionally, as shown in the red circles on SER figures 2-1 and 2-2, SAR figures 2.1.2 and 2.1.5 show numerous circular features that appear similar to swales or sinkholes throughout the site and surrounding areas.

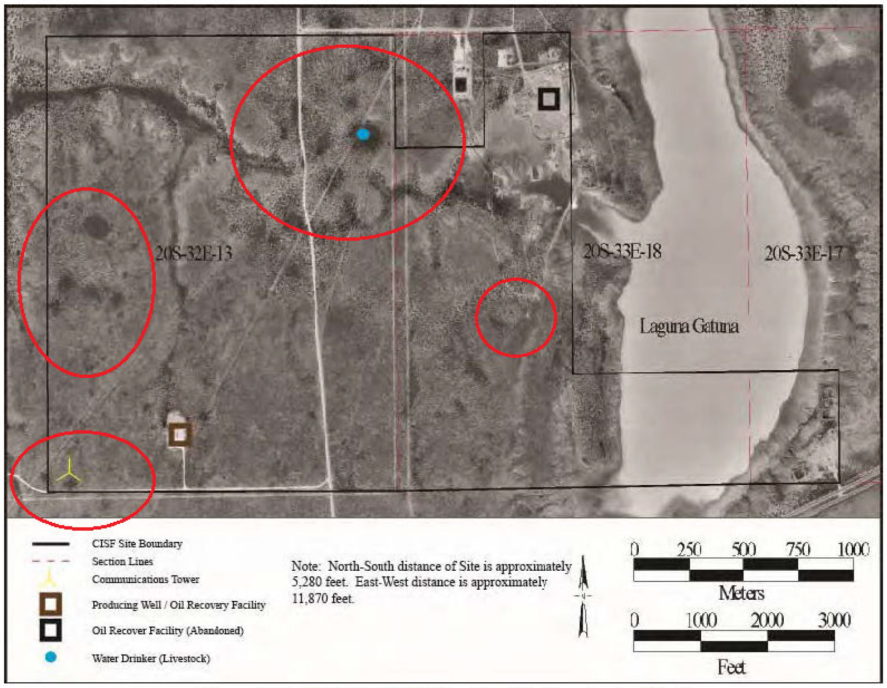

Figure 2.1.2: HI-STORE CIS Facility Site Boundaries [2.1.3]

Figure 2-1 SAR figure 2.1.2 with localized surface depressions circled in red

Figure 2.1.5: Aerial View of the Site (Full Build-Out) [2.1.8]

Figure 2-2 SAR figure 2.1.5 with localized surface depressions circled in red (from SAR Revision 0B, dated January 23, 2018)

In its September 16, 2020, response to RAI 2-26, the applicant indicated that the circled features are localized surface depressions without a surficial drainage outlet. The features are small scale, generally less than 30 m (100 feet) in diameter and about 0.6 m (2 feet) deep. The applicant attributed the presence of these features in the site vicinity to vegetation changes, which may also account for the features' ability to intermittently hold small quantities of water.

The applicant explained that undrained depressions such as these are common in southeastern New Mexico and likely formed by leaching of the caliche cap and wind removal of the loosened materials, and not by dissolution of the Capitan Reef or other subsurface carbonate and evaporite deposits. The applicant clarified that because of the localized origins of these features, they are unlikely to impact constructed features at the site. Furthermore, because the caliche caprock will be excavated beneath and adjacent to the cask storage pads, these features are not likely to form at the site during the licensed life of the facility. SAR figure 2.6.17 identifies the locations of these localized depression features.

The staff reviewed the information in the applicant's response, including the inferred origin of these features. Additionally, during a site visit in June 2019, the staff viewed several of these features in the field and noted the change in vegetation within the features as well as the small scale of the features. Based on the staff's observations of the vegetation and scale, the inferred surficial origin of these features, and the planned removal of the caprock caliche at the site, the staff concludes that features such as these will not affect the facility or result in surface deformation at the site.

The staff reviewed SAR sections 2.6.1 and 2.6.4 to determine the potential for tectonic and nontectonic surface deformation at the proposed site, including surface deformation due to the dissolution of subsurface layers, or karst. SAR section 2.6.1 describes a caliche layer encountered near the surface as a well-developed, naturally cemented calcium carbonate; a silty sand with gravel; and a clayey sand with sandy lean clay layers. SAR section 2.6.4 states that the "Mescalero caliche is soluble and situated at or near land surface" and that "dissolution of this unit may have resulted in the development of a number of small shallow depressions in the area; however, this is not regarded as an active or significant karst process at the Site." The staff reviewed the reference cited for this conclusion (the 2007 GNEP siting study, SAR reference 2.1.3) and a reference cited therein (Bachman, 1976). Bachman (1976) noted that although the dissolution of soluble rocks has caused subsidence and surface collapse in southeastern New Mexico, these processes have been active in this area for long intervals in geologic time. Bachman (1976) further concludes that the Mescalero caliche is a "widespread datum for determining the relative time of solution and collapse of many modern karst features." Although deformation of the Mescalero caliche records three episodes of solution and collapse, Bachman (1976) does not attribute the deformation observed in the Mescalero caliche to dissolution of the Mescalero caliche.

In its response to RAI 2-27 dated September 16, 2020, the applicant provided additional information clarifying that borings show no evidence of deformation of the Mescalero caliche or other surficial deposits from underlying dissolution. Furthermore, there are about 300 m (1,000 feet) of insoluble rock units before encountering soluble rocks in the subsurface, and regional drillhole data show no evidence of dissolution in the shallowest soluble layers. The applicant also updated the SAR to clarify that leaching of the caliche cap and subsequent removal of the loosened material is the cause of the small shallow depressions observed in the site area. SAR figure 2.6.17 shows the locations of these depressions, which appear to be the same features discussed in the applicant's September 16, 2020, response to RAI 2-26. The response to RAI 2-25 clarified that these small shallow depressions were the result of leaching of the caliche cap and wind removal of the loosened materials, not of the dissolution of the subsurface materials.

Based on the clarification that the underlying soluble layers are not the origin of the small shallow depressions, and the description of the formation of these shallow depressions in response to RAI 2-26, the staff concludes that these shallow depressions are the result of surficial processes and not surface deformation resulting from the dissolution of subsurface layers.

The staff also considered the potential for surface deformation at the site due to nontectonic anthropogenic activities, such as oil and gas exploration and extraction or mine collapse. SAR section 2.6.4 states that "any future oil drilling or fracking beneath the site would occur at greater than 930 m (3,050 feet) depth... which ensures there would be no subsidence concerns." SAR section 2.1.4 states that "because of the extent of the evaporites (salt and anhydrite), drilling and completion operations have to be conducted in a manner that prevents the dissolution of the salt and protects the well during drilling and through the productive lives of the wells, often 20 to 30 years or more." SAR section 2.6.4 further states that "there are no surface, drillhole, or mining indications that subsidence and collapse chimneys occur at the site or surrounding area." However, SAR section 2.1.4 states that "there are several examples in the Permian Basin of catastrophic subsidence as a result of suspected oil field casing corrosion and dissolution of salt," including the Wink Sinks and Jal Sink, both of which have similar subsurface stratigraphy to the Holtec site.

Accordingly, in its response to RAI 2-25, dated September 16, 2020, the applicant provided additional information to address the potential for subsidence at the site due to oil and gas extraction at the site, clarifying that it revised the minimum horizontal drilling depth to 930 m (3,050 feet) throughout the SAR, which reflects the shallowest oil and gas deposits in the area. The applicant stated that fracking would pose a negligible potential for surface subsidence since most extracted materials are removed by replacement and would not leave large void spaces in the subsurface. The applicant also stated that the existing oil and gas wells are located a sufficient distance from the proposed site so as not to be considered a surface deformation hazard at the site. The applicant also discussed the potential for subsidence or other surface deformation in response to RAIs 2-8, 2-10, 2-11, and 2-12, as described in SER section 2.3.1.

The staff reviewed the RAI response and determined that surface deformation due to fracking is unlikely because the removed materials are replaced, which eliminates the open space, supports the overburden, and limits the potential for surface deformation. Furthermore, the staff notes that the response to RAI 2-8 states that because the existing wells and drill islands are located more than 300 m (1,000 feet) from the storage areas of the proposed facility and have not been used for water extraction, subsidence due to casing corrosion or failure leading to catastrophic collapse is not likely to affect the proposed site. SER section 2.3.1 contains the staff's discussion and conclusions related to the potential for subsidence due to potash mining.

The staff reviewed the information in SAR section 2.6.1 and found it acceptable because the basic geologic and seismic characteristics of the site and vicinity have been adequately described in detail to allow the investigation of the facility's seismic characteristics. The staff determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the facility, perform additional safety analyses, and demonstrate compliance with regulatory requirements in 10 CFR 72.92(a), 10 CFR 72.92(b), 10 CFR 72.103(f), and 10 CFR 72.122(b) with respect to basic geologic and seismic information.

2.3.6.2 Vibratory Ground Motion

SAR section 2.6.2 discusses the development of design-basis vibratory ground motion associated with credible levels of vibratory ground motions that may be experienced at the site. In reviewing the applicant’s development of vibratory ground motion, the staff considered factors related to the principal design elements of seismic hazard analyses and procedures for determining the DE. The staff reviewed the applicant’s investigations of basic geologic and seismic information, as discussed in SER section 2.3.6.1, and the following aspects of ground motion at the site: (1) fault seismic sources, (2) distributed seismic sources, (3) applicable ground motion attenuation relations, (4) site response analyses, and (5) development of site-specific design ground motion.

The applicant cited 10 CFR Part 72 as the basis for determining facility DE ground motions. Until its revision in October 2003, this regulation, in particular 10 CFR 72.103, required the development of a DE in accordance with 10 CFR Part 100, Appendix A. The seismic hazard methodology in 10 CFR Part 100, Appendix A, is based on a deterministic approach in which the largest credible earthquake that could occur on the closest approach of the seismic source to the site is considered as the DE. In 2003, the NRC amended 10 CFR Part 72. The rule change requires that uncertainties inherent in estimates of the DE be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA) or suitable sensitivity analysis, as set forth by 10 CFR 72.103. RG 3.73, “Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations,” issued October 2003, provides general guidance on procedures acceptable to the NRC staff for conducting the PSHA and developing the DE to satisfy the requirements of 10 CFR Part 72. RG 3.73 further specifies that the controlling earthquakes are to be developed for the vibratory ground motion corresponding to the mean annual frequency of exceedance of 5×10^{-4} per year.

In characterizing the vibratory ground motion developing the licensing-basis response spectra, the applicant used a probabilistic approach, consistent with 10 CFR 72.103 and RG 3.73.

2.3.6.2.1 Fault Seismic Sources

SAR section 2.6.2 provides details about the approach used by the applicant to determine the site-specific design ground motion. The applicant applied the USGS National Seismic Hazard Mapping Project (NSHMP) results for the Holtec location to determine the DE for the site. The USGS NSHMP is a nationwide study of seismic hazard published on a periodic basis and used by government planning and regulatory bodies. The applicant applied the 2008 version of the NSHMP (Peterson et al., 2008) at the Holtec site. The applicant stated that based on NSHMP inputs, there are a total of 27 Quaternary faults or fault zones within a 320 km (200-mi) radius of the Holtec site. Of these 27 faults, the applicant considered only four faults to be capable based on the criterion that there is evidence for movement at or near the ground surface at least once within the past 35,000 years or recurring movement within the past 500,000 years. The applicant stated that the Guadalupe fault is located at a minimum distance of 140 km (85 mi) from the site and is the nearest fault to the site. The applicant stated that the fault is short at approximately 5.8 km (3.6 mi) long and has a low slip rate of 0.025 cm (0.01 inch) per year.

The staff reviewed the information available in SAR section 2.6.2 and performed an independent survey of faults within 320 km (200 mi) of the site. Specifically, the staff searched the USGS Quaternary fault database for faults located within 200 km (120 mi) of the site. The staff's confirmatory search identified the same faults used in the USGS NSHMP calculations. Based on the applicant's use of hazard-significant faults through its application of the USGS NSHMP results, the staff determined that the applicant's model of fault seismic sources is acceptable.

2.3.6.2.2 Distributed Seismic Sources

SAR section 2.6.2 provides details on the inputs used for determining the site-specific design ground motion. The applicant stated that three distributed seismic source zones are within 200 mi of the site: the Southern Basin and Range, the Rio Grande Rift, and the Central Basin Platform. The applicant stated that the most active seismic source zone within 200 mi of the site is the Central Basin Platform. The applicant also stated that the seismicity of the Central Basin Platform is likely associated with oil production activities (i.e., induced seismicity) in the vicinity of the earthquake locations. The USGS NSHMP, which forms the basis for the applicant's DE, explicitly excludes the existence of, or potential for, induced seismicity in its calculations of earthquake frequency.

In its RAI response dated September 16, 2020, the applicant explained how its analysis accounts for the existence of induced seismicity in the region by stating that the bounding earthquake for its design is the design extended condition earthquake (DECE). SAR section 4.3.6, "Applicable Earthquake Loadings for the HI-STORE CIS Facility," describes the DECE, defined as 1.67 times the DE. The applicant stated that it applied the DECE for design purposes to account for uncertainties in seismic hazard and the potential for hazard estimates to change in the future.

The NRC staff reviewed the applicant's response to RAI 2-30 and finds it acceptable because the applicant used the DECE for the design of seismically qualified SSCs and because the DECE envelopes the seismic hazard estimates for the site developed by the staff and those from the USGS for the site and appropriately accounts for uncertainties in ground motion estimates. The staff's independent confirmatory analysis is described below.

2.3.6.2.3 Ground Motion Attenuation Relation

SAR section 2.6.2 details the applicant's approach to developing its site DE. The applicant used information available from the USGS NSHMP to determine the DE for the site. The USGS NSHMP uses a suite of up-to-date ground motion attenuation relations to perform its analysis of seismic hazard. Because the USGS NSHMP uses a broad range of inputs in its PSHA that accounts for uncertainties in ground motions across the United States, the staff considers this approach acceptable.

2.3.6.2.4 Site Response Analysis

The applicant used information available from the USGS NSHMP to develop its site DE. The USGS NSHMP is a nationwide study and does not incorporate site-specific information in developing its PSHA results. Rather, the USGS calculates hazard based on site classes. These site classes are generally based on the seismic velocities in the upper 30 m (100 feet) of the subsurface and are determined on a regional basis. However, the determination of the DE does not incorporate site-specific information that would impact site-specific earthquake response.

In its RAI response dated September 16, 2020, the applicant justified not using site-specific subsurface geologic and geophysical properties in the development of its site-specific DE. The applicant stated that the bounding earthquake for its design is the DECE. SAR section 4.3.6 describes the DECE, defined as 1.67 times the DE. The applicant stated that it applies the DECE for design purposes to account for uncertainties in seismic hazard and the potential for hazard estimates to change in the future.

The NRC staff reviewed the applicant's response to RAI 2-30 and finds it acceptable because the DECE is used for the design of seismically qualified SSCs and because the DECE envelopes the seismic hazard estimates for the site developed by the staff and those from the USGS for the site.

2.3.6.2.5 Design-Basis Ground Motion

In SAR section 4.3.2, "VVM Components and ISFSI Structures," the applicant provided its design response spectrum. The applicant stated that the design-basis earthquake/safe-shutdown earthquake is set to bound the 10,000-year return period earthquake and is represented by a spectrum based on RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 2, issued July 2014, anchored at 0.15g. The applicant also stated that for additional conservatism, it identified a DECE with an RG 1.60 spectrum anchored at 0.25g.

The staff reviewed the applicant's approach to developing the DE and DECE. The applicant's approach is acceptable because it is based on determining the 1/10,000 mean annual frequency of exceedance ground motion using the USGS NSHMP results and enveloping that with an RG 1.60 spectrum.

2.3.6.2.6 NRC Staff Confirmatory Analysis

In order to better assess the adequacy of the applicant's DE, the NRC staff performed a confirmatory PSHA for the Holtec site. The staff independently developed PSHA inputs using information available in the applicant's SAR and the Interim Storage Partners' WCS CISF application, dated September 2, 2020, and information from the USGS and previous NRC reviews.

The Holtec site rests within the region modeled in NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities" (CEUS-SSC), published jointly in January 2012 by the NRC, the U.S. Department of Energy, and the Electric Power Research Institute (EPRI). The CEUS-SSC is an NRC-endorsed seismic source model for the central and eastern United States and is an acceptable starting model for performing PSHAs. Because the 320 km (200 mi) region surrounding the Holtec site is not entirely within the CEUS-SSC model domain, the NRC staff developed a model for the western portion of the study area. The staff treated this region as an area seismic source with seismicity parameters similar to those used by the USGS in its development of the NSHMP.

Because the Holtec site region crosses the boundary from the stable central and eastern United States to the geologically active western United States, the NRC staff used two separate ground-motion models (GMMs) to develop its PSHA. For the portion of the study region covered by the central and eastern United States, the NRC staff used the NRC-accepted GMM published by EPRI (EPRI, 2013). For the western portion of the study area, the NRC staff applied the GMM developed during the southwestern U.S. GMM project (GeoPentech, 2015). Specifically, the NRC staff applied the elements of this model developed for use in the Southern Basin and Range province. The NRC staff applied a Vs-kappa correction to adjust the western U.S. GMM for use at the Holtec site. A Vs-kappa correction is a correction applied to a GMM to adjust for seismic velocities and damping in the upper crust and represents the change in shape of a seismic response spectrum as a result of changing crustal conditions. This correction produces seismic hazard curves that are consistent with the crustal characteristics of the Holtec site and the source characteristics consistent with those found in the western United States.

The staff's confirmatory PSHA included a site response analysis. The staff used site-specific depth and velocity information collected by the applicant, as well as information provided in the Interim Storage Partners' WCS CISF application dated April 12, 2021, over the upper 1,200 feet (365 m). At depth, the staff used information from regional geologic profiles and seismic velocities from the literature for similar rock types (Kenter, et al., 1997). To represent input rock motions, the staff used an **M**6.5 earthquake with 11 input peak ground accelerations (PGAs) ranging from 0.1g to 1.5g. The staff convolved the results of its amplification function with its confirmatory base rock PSHA results to develop site-specific seismic hazard curves at the surface.

The staff compares its site-specific 1×10^{-4}/year uniform hazard response spectrum to the applicant's DE and DECE in SER figure 2.6.2-1. As shown in SER figure 2.6.2-1, the NRC staff's 1×10^{-4}/year uniform hazard response spectrum exceeds the applicant's DE at frequencies greater than approximately 15 hertz. However, the DECE envelopes the staff's results at all frequencies. The DE and DECE also envelope the 1×10^{-4}/year hazard estimates from the "Documentation for the 2014 Update of the United States National Seismic Hazard Maps," issued by the USGS in 2014. Based on these results, the staff concludes that the DECE adequately accounts for uncertainty in the site-specific seismic hazard and that the applicant adequately addressed site specific ground motion parameters.

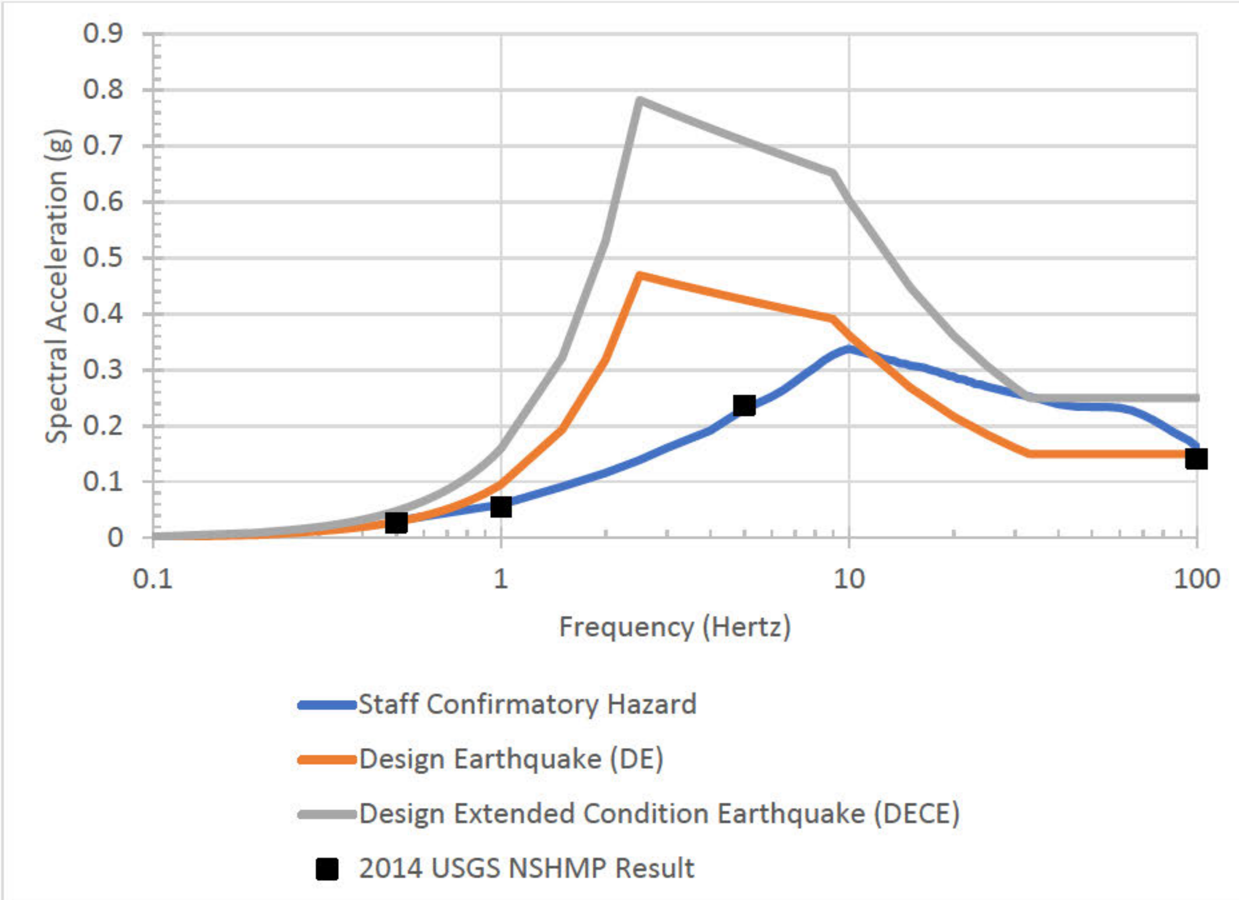

Figure 2-3 Comparison of the NRC staff's confirmatory PSHA with the DE and DECE for the Holtec site, including the 2014 USGS NSHMP results for selected frequencies

2.3.6.2.7 NRC Staff Conclusions Regarding Vibratory Ground Motion

The NRC staff reviewed the information submitted by the applicant regarding vibratory ground motion at the Holtec site. Based on the NRC staff's review of the submitted information and confirmatory PSHA results discussed in the preceding sections, the NRC staff concludes that the DECE developed by the applicant is acceptable for use in the HI-STORE CIS Facility design analyses. The staff further concludes that the DECE is acceptable for use in other sections of the SAR to develop the design bases of the HI-STORE CIS Facility, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.92(a)–(b), 10 CFR 72.103(f), and 10 CFR 72.122(b) with respect to this topic.

2.3.6.3 Surface Faulting

In SAR sections 2.6.2 and 2.6.3, the applicant presented information on surface faulting in the area of the proposed site. The applicant stated that 27 faults within a 320 km (200 mi) radius of the site demonstrate Quaternary-age activity. The closest Quaternary-age fault to the Holtec site is the Guadalupe fault at the western base of the Guadalupe Mountains approximately 140 km (85 mi) southwest of the site. The applicant noted that the northern Delaware Basin in the area of the site is characterized by slow uplift relative to the surrounding areas and concluded that no recent faulting or changes in geology processes have occurred in the vicinity of the site.

The NRC staff assessed the information presented in the SAR and accompanying sources with respect to the presence or absence of surface faults and faulting in the vicinity of the proposed site. The preponderance of information presented indicates that there are no surface, near-surface, or Quaternary faults capable of movement in the vicinity (within 8 km) of the proposed site. The staff confirmed that the closest fault with determined recent movement is 85 mi to the southwest.

The staff finds the information provided related to surface faulting acceptable because the lack of surface faults in the site area is well described and documented. The information is acceptable for use in other sections of the SAR to develop the design bases of the Holtec facility, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.90(b)–(d), 10 CFR 72.92(a)–(c), 10 CFR 72.98(c)(3), and 10 CFR 72.122(b) with respect to surface faulting.

2.3.6.4 Stability of Subsurface Materials

The staff reviewed the information presented in SAR section 2.6.4. In addition, the staff also reviewed the 2018 GEI geotechnical data report and two calculation packages associated with estimating the bearing capacities and settlements of the foundations of both the storage pads and the CTF (Holtec Report No. HI-2188143, "HI-STORE Bearing Capacity and Settlement Calculations" (proprietary), Revision 6, dated January 20, 2023) and a third report that addressed the potential for liquefaction during a seismic event (GEI Report CIS-CS 001-00, "Liquefaction Resistance of Soils for HI-STORE CISF" (proprietary), Revision 0, August 25, 2020 (2020 GEI soil liquefaction calculation)). Additionally, the staff reviewed the responses provided by the applicant to RAIs 2-29, 2-32, 2-32-S, 2-33, 2-33S, 2-33-S-1, 2-34, 2-35, 2-36, 2-37, 2-38, 2-39, 2-40, 2-40-S, 2-41, 2-42, 2-42-S, and 2-42-S-1. The staff review focused on the geotechnical site characterization, the stability of the storage pads and the CTF foundation, and the potential for soil liquefaction. The sections below describe the staff's evaluation and conclusions.

2.3.6.4.1 Geotechnical Site Characterization

The applicant provided subsurface information at the proposed site in SAR section 2.6.1 and the 2018 GEI geotechnical data report. Nine borings (SAR figure 2.1.8, "Phase I Boring Location Map") were drilled at the Phase 1 section of the proposed site, and the geotechnical data report provides detailed information about the borings. Seven of these borings (B101 through B107) are within the proposed ISFSI pad areas, and the footprint of the CTB (B109) and the road for transporting canisters from the CTB to the storage pad area (B108) have one boring each. The applicant used information obtained from these borings to identify the subsurface strata (both soil and rock layers) and determine the geotechnical characteristics for the design of the ISFSI storage pads, CTB, and other facility structures. Another six offset borings were drilled (boring number with a suffix A or B) to accommodate inclinometer casing installation, monitoring hole installation, and hammer energy measurement; however, these borings were not used for collecting soil samples or conducting the standard penetration test (SPT).

All borings were drilled with the hollow-stem auger without drilling fluid, followed by rock coring using HQ wireline tooling and water as the drilling fluid. The offset borings were drilled with open-hole roller bit using air cooling. A GEI engineer or a geologist was present during the drilling, sampling, and testing operations.

Information from borings B101 through B107 was used for site characterization of the storage pad foundation. Information from boring B109 was used for designing the CTB. The depth of borings B102, B105, B106, and B107 varies from 18.3 to 45.6 m (61 to 152 feet). Santa Rosa sandstone was not reached in these borings. Santa Rosa sandstone was encountered in borings B101 and B105, which are 120.2 m (400.5 feet) and 66.6 m (222 feet) deep, respectively. The depth of boring B108 is 18.3 m (61 feet), and the depth of boring B109 is 30.6 m (102 feet).

All soil samples were collected and tested using the appropriate ASTM International (ASTM) standards, listed in a table in section 4.4.1, “Geotechnical Laboratory Testing,” of GEI Work Plan 1, Revision 3, issued 2017, which appears as appendix A to the 2018 GEI geotechnical data report. The work was conducted in accordance with a nuclear quality assurance program in compliance with Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and followed appropriate GEI procedures, listed in table 1 of GEI Work Plan 1, Revision 3.

Thick-walled split spoon samples were collected at selected depths by using a Modified California split spoon sampler fitted with six liner rings. As discussed in the 2018 GEI geotechnical data report, thin-walled samplers were not used because the soil at the site was too dense to push the thin-walled tubes.

Bulk soil samples of auger cuttings from depths varying from 0 to 3.3 m (0 to 10 feet) were collected from all borings except B106. GEI used two CME-75 and CME-85 drill rigs with an automatic hammer for all SPT measurements. SPTs were performed to measure the penetration resistance of the soil at the proposed site using split spoon samplers having an outer diameter of 5 cm (2 inches) and length of 61 cm (24 inches) without liners and with internal diameters of 3.8 cm (1.5 inches). The drop of the drive weights was 76 ± 2.5 cm (30 ± 1 inch) in accordance with ASTM D1586-11, “Standard Test Method for Standard Penetration Test (SPT) and Split-Barrel Sampling of Soils.” The method used to measure the SPT blow count N follows ASTM D1586-11.

The soil samples were tested in the laboratory to evaluate the physical, index, and engineering properties. Laboratory tests performed included moisture content and unit weight, Atterberg limits (liquid limit and plastic limit), consolidation, grain-size distribution, consolidated-undrained and unconsolidated-undrained triaxial compression, and unconfined compression. The 2018 GEI geotechnical data report gives the results of the soil laboratory tests and logs of the sample borings.

During the drilling program, groundwater was observed in boring B107(MW) at a depth range of 28.4 to 30.5 m (93.1 to 100.0 feet) from the surface. In the monitoring well B101(MW), which was screened in the Santa Rosa Formation, groundwater was observed at a depth of 77.2 to 80.4 m (253 to 264 feet) from the surface. Groundwater was also observed in boring ELEA-2 at 11.5 to 11.5 m (37.6 to 37.7 feet), likely within the upper Chinle Formation. Citing the literature (e.g., Nicholson and Clebsch, 1961), the 2018 GEI geotechnical data report states that the Santa Rosa Formation is the principal aquifer in the western part of southern Lea County, although water may be present in isolated layers or lenses of sandy material in the Chinle

Formation. The 2018 GEI geotechnical data report interprets that the deep groundwater level in boring B101 indicates that the primary groundwater at the site occurs at a depth of 77.1 to 80.2 m (253 to 263 feet) below the surface. Groundwater observed above this horizon has been interpreted to be discontinuous aquifers above the low-permeability zones in the Chinle Formation.

The 2018 GEI geotechnical data report discusses the measurement of crosshole seismic shear wave velocity in an array of three borings (B102A, B103, and B104). The shear wave velocity measurements were conducted in accordance with a procedure that incorporates by reference the international standards ASTM D4428-14, “Standard Test Methods for Crosshole Seismic Testing,” and ASTM D6230-21, “Standard Practices for Monitoring Earth or Structural Movement Using Inclinometers.” Wave velocity was measured at 1.5 m (5-foot) interval starting at 0.6 m (2 feet) below the ground surface and ending at 32.0 m (105 feet) below the ground surface. SAR figure 2.6.10, “Phase 1 Shear Wave Velocity Results,” shows the measured shear wave velocity as function of depth.

The 2018 GEI geotechnical data report classifies the subsurface at the proposed site based on visual observation of the change in stratigraphy; field classification, modified by the laboratory results, if required; and difficulty in advancing the augers. The subsurface is described as follows, along with some characteristics for each material layer.

<u>Top Soil</u>: Thickness of the top soil layer is 8 to 10 cm (3 to 4 inches) at the proposed ISFSI pad site. Thickness of the top soil increases to 2.5 m (8.1 feet) at the CTB site, as observed in boring B109. SPT N values range from 5 to 14 blows per foot, which indicates loose to medium-dense soil. Based on SAR figure 2.6.8, “Phase 1 Detailed Subsurface Profile C,” the top soil will be excavated before construction of both the storage pads and the CTB.

<u>Caliche</u>: This Mescalero caliche layer is approximately 1.3 to 4.1 m (4.4 to 13.5 feet) thick. The caliche has varying amount of sand and gravel with silt. SPT N values vary from 20 blows per foot to more than 100 blows per foot (split spoon refusal). The median SPT blow count is 43 blows per foot, indicating dense to very dense soil. Based on SAR figure 2.6.8, the caliche layer will be excavated before construction of both the storage pads and the CTB.

<u>Residual Soil</u>: The thickness of the residual soil layer varies from 5.2 to 8.5 m (17 to 28 feet). It is formed due to weathering of the underlying Chinle Formation. The residual soil typically consists of sandy clay or clayey sand with a trace of gravel. The residual soil was differentiated in the field from the Chinle Formation based on visual and gradational changes in stratigraphy, SPT refusal, and difficulty in advancing the auger. The top portion of the residual soil layer will be excavated for constructing the storage pads and the CTB, as shown in SAR figure 2.6.8.

SPT N values vary from 37 blows to more than 100 blows per foot (split spoon refusal) in residual soil at the proposed site. Typical N values are more than 100 blows per foot, indicating a very hard or very dense soil. The specific gravity is 2.74, and the unit weight varies between 1,579 to 2,025 kilograms per cubic meter (kg/m^3) (98.6 and 126.4 pounds per cubic foot (lb/ft^3)). The fines in the residual soil are generally moderately plastic clay with a liquid limit ranging from 40 to 42 percent, and the plasticity index varies from 22 to 28 percent. Natural moisture content varies between 8.3 percent and 10.7 percent, significantly below the plastic limit.

<u>Chinle Formation</u>: The Chinle Formation is the upper member of the Dockum Group and underlies the residual soil at the proposed site. The Chinle Formation is a poorly indurated mudstone. It has interbedded lenses of moderate to well-indurated siltstones and conglomerate.

The Chinle Formation was encountered at a depth of 8.4 to 12.3 m (27.5 to 40.5 feet) below the surface. The thickness of the Chinle Formation was measured as 64.9 to 65.2 m (213 to 214 feet) in borings B101 and B105. Other borings terminated within the Chinle Formation.

SPT values measured in the Chinle Formation indicate a transition from very dense or very hard soil to soft rock with depth. Rock quality designation (RQD) in the Chinle Formation varies from 0 to 100 percent, with a median of 74 percent. Lower core recoveries and lower RQD values are typical in the very poorly indurated mudstones of the upper portion of the Chinle Formation. Specific gravity ranges between 2.78 and 2.81. Unit weight of the Chinle Formation ranges between 2,034 and 2,211 kg/m^3 (127 and 138 lb/ft^3), with a mean of 2,146 kg/m^3 (134 lb/ft^3). Unconfined compressive strength measured on laboratory samples varies between 0.25 and 1.23 MPa (5.3 and 25.7 kips per square foot (ksf)), with a mean of 0.77 MPa (16.1 ksf).

<u>Santa Rosa Formation</u>: The Santa Rosa Formation is the lower member of the Dockum Group and underlies the Chinle Formation. It is a water-bearing formation in the region. The Santa Rosa Formation is fine to coarse-grained sandstone with minor reddish-brown siltstones and conglomerates. At the proposed site, this sandstone was encountered in borings B101 and B105 at a depth of approximately 66 m (215 feet) below the surface. Both of these borings terminate within this rock unit. The applicant considered the Santa Rosa sandstone as the bedrock in analyzing the foundation stability.

RQD values are typically greater than 80 percent and vary between 33 to 100 percent, with a median of 98 percent. The unit weight of the Santa Rosa Formation ranges from 2,499 to 2,627 kg/m^3 (156 to 164 lb/ft^3), with a mean of 2,563 kg/m^3 (160 lb/ft^3). The unconfined compression strength ranges between 14.0 and 48.0 MPa (293 to 1,003 ksf), with a mean of 34.4 MPa (718 ksf).

The staff reviewed the geotechnical site characterization information provided in the SAR and in the 2018 GEI geotechnical data report and concludes that (1) the depth and thicknesses of soil layers and the water table depth at the site were determined using standard methods and procedures consistent with the staff guidance in RG 1.132, “Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants,” Revision 3, issued December 2021, and (2) the index properties and strength and compressibility of the soil layers were determined using an appropriate combination of field and laboratory testing consistent with the guidance in RG 1.138, “Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants,” Revision 3, issued December 2014. Based on the review of the information presented in the SAR and the 2018 GEI geotechnical data report, the staff concludes that the geotechnical site characterization has been adequately described and assessed. In addition, the staff concludes that the information provided by the applicant is sufficient to assess the stability of the foundations of the storage pads and the CTB to allow them to perform their safety functions as required under 10 CFR 72.122(b). Additionally, the staff concludes that the geotechnical site characterization information presented in the SAR is adequate for use in other sections of the SAR to develop the design bases for the proposed facility, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.103(c)–(d) and 10 CFR 72.122(b).

2.3.6.4.2 Stability of Subsurface Materials and Foundations

2.3.6.4.2.1 Stability of Foundations

In Holtec Report No. HI-2188143, the applicant presented the analysis conducted to show that the subsurface of the UMAX storage pads and the CTF remains stable from the loads imposed by the facilities and design-basis seismic events. Attachment A.1 (untitled) of appendix A, “Bearing Capacity and Settlement Calculations of the UMAX,” of Holtec Report No. HI-2188143 illustrates the proposed storage pads and the subsurface soil and rock layers. SAR figure 4.3.1 schematically shows the pad configuration and its subgrade materials. Details of the storage pads appear in appendix A to Holtec Report No. HI-2188143, while the report’s appendix B, “Bearing Capacity and Settlement Calculations of the CTF,” contains details on the CTF foundation. Attachment B.1 (untitled) of Holtec Report No. HI-2188143 and SAR section 1.5, “Licensing Drawings,” drawing 10912, “Cask Transfer Building (CTB) Licensing Drawing,” sheet 3 of 7, Revision 0.2 (proprietary), illustrate the CTF foundation subgrade.

2.3.6.4.2.2 Description of UMAX Storage Pads

The proposed facility will have two identical storage pads in Phase I. The applicant described the design of the ISFSI pad and the SFP in SAR section 5.3.1, “HI-STORM UMAX ISFSI Pad and Support Foundation Pad. The UMAX ISFSI pad and the SFP are integral components of the HI-STORE UMAX storage system. Table 5.1, “Size and Loading Information for UMAX,” of Holtec Report No. HI-2188143 presents the dimensions and other relevant design features of the two UMAX ISFSI pads. Each pad is and will hold 250 vertical ventilated modules (VVMs) in a 25 × 10 array. Table 5.1 of Holtec Report No. HI-2188143 also provides the unit weight of the concrete and the CLSM to be used in constructing the storage pads and the subgrades. These values are consistent with those in SAR table 4.3.3.

As stated in the SAR, the ISFSI pad at the top provides the riding surface for the loaded transporter. The SFP at the bottom provides robust support to the loaded VVMs and limits long-term settlement (consolidation settlement). There is a small mismatch between the weight of the excavated subgrade above the SFP and the combined weight of the refilled subgrade and the loaded VVMs. Consequently, a small amount of long-term settlement is expected. The long-term or consolidation settlement is taken as part of the “dead load” in the load combination assumed in designing the storage pads following the American Concrete Institute (ACI) national design code ACI 318-14, “Building Code Requirements for Structural Concrete.” The subgrade, the Space A between the ISFSI pad and the SFP as shown in SAR figure 4.3.1, is to be filled with the CLSM or lean concrete, which will not undergo long-term settlement.

The 2018 GEI geotechnical data report indicates that the SFP will be embedded in the residual soil layer, which overlies the Chinle Formation. The Chinle Formation is a single stratum at the proposed site. There is no geologic difference between the “Chinle Formation Soil” and “Chinle Formation Mudrock.” In the boring logs, the stratum is called “clay” (i.e., soil) if the SPT could be performed and split spoon samples could be obtained. The boring logs describe material that needed more than 100 SPT blow count N (i.e., hammer refusal, as defined in ASTM D1586-11) as “Mudrock.” Typical auger refusal was not observed in transitioning from “Soil” to “Mudrock.” Rather, a transitional increase of resistance with depth was observed. An increase in SPT N values indicates the transition of the Chinle Formation from a very dense or very hard soil to soft rock. At the proposed site, the transition to rock takes place within the elevation of 1,047.1 m and 1,066.7 m (3,434.5 feet and 3,499.6 feet) AMSL, as stated in appendix B to Holtec Report No. HI-2188143.

2.3.6.4.2.3 Description of the Canister Transfer Facility Foundation

The applicant presented the dimensions and relevant design parameters of the CTF foundation pad in SAR section 5.3.3. A reinforced concrete slab at the base of the CTF, called the CTF foundation, supports the transport cask during transfer operations. The foundation slab will be embedded in the residual soil layer, which overlies the Chinle Formation, as shown in appendix B to Holtec Report No. HI-2188143.

SAR section 1.5, drawing 10912, sheet 3 of 7, Revision 0.2 (proprietary), shows a layer of engineered fill placed beneath the CTF foundation. This fill layer would have a minimum thickness of 30 cm (12 inches). SAR table 4.6.2, “Reference Design Data for the CTB Slab,” also indicates the 12-inch minimum thickness. For conservatism, the applicant did not take any credit for this layer of engineered fill in estimating the bearing capacity and settlements. The CTF foundation was assumed to be directly over the residual soil. This engineered fill will be stiffer than the residual soil, which it will replace. The staff agrees that ignoring the fill layer is conservative, as the estimated bearing capacity of the foundation would be smaller than the actual capacity. In addition, the expected immediate and long-term or consolidated settlement of the foundation would be larger than the actual settlement.

2.3.6.4.2.4 Subsurface Beneath the UMAX Storage Pads

As illustrated in SAR figure 4.3.1, the SFP sits on the residual soil layer. The thickness of the residual soil beneath the storage pad is 3.7 m (12 feet). The applicant divided the Chinle Formation into two portions. The upper portion of the Chinle Formation is 9.4 m (31 feet) thick and called “soil” for analysis purposes only. For analysis purposes, the lower portion of the Chinle Formation is called “mudrock,” which is 45 m (148 feet) thick. The Chinle Formation overlies the Santa Rosa Formation, which is taken as the rigid base. The groundwater table is identified at elevation 999.6 m (3,279.5 feet) below the existing ground surface at elevation 1,076.9 m (3,533 feet).

2.3.6.4.2.5 Subsurface Beneath the Canister Transfer Facility

The CTF foundation will be embedded in the residual soil layer, which overlies the Chinle Formation. The thickness of the residual soil layer below the CTF is 3 m (10 feet). Again, for analysis purposes only, the Chinle Formation is subdivided into “soil” and “mudrock” layers with thicknesses of 7.3 m (24 feet) and 12.8 m (42 feet), respectively. The Chinle Formation overlies the Santa Rosa Formation, which is taken as the rigid base. The groundwater table is in the Santa Rosa Formation. As discussed above in SER section 2.3.6.4.2.4, this assessment ignored the presence of the engineered fill layer beneath the CTF foundation for conservatism.

2.3.6.4.2.6 Subsurface Material Properties

The applicant presented the material properties necessary to estimate the bearing capacity and settlements of the foundation pads in table 5.3, “Soil Parameters,” of Holtec Report No. HI-2188143. These values were determined based on the 2018 GEI geotechnical data report. The applicant calculated the unit weight of the residual soil to be 1,802 kg/m^3 (112.5 lb/ft^3) based on measurements made in the 2018 GEI geotechnical data report. The applicant assumed a nominal value for cohesion of 9.6 kPa (200 pounds per square foot (lb/ft^2)) to represent the clay present in the residual soil. The staff finds this acceptable as the residual soil at the proposed site typically consists of stiff clayey sand or sandy clay with a trace of gravel. Based on the SPT blow counts corrected for clean sand, $(N_1)_{60}$, as given in Holtec Report No. HI-2188143 and the 2020 GEI soil liquefaction calculation, the residual soil is very hard or very dense soil. The applicant assumed the Poisson's ratio of the residual soil selected from table 2-7, "Values or value ranges for Poisson's ratio μ," of "Foundation Analysis and Design," by J.E. Bowles, published in 1996, to be appropriate for sandy clay. The staff finds this acceptable.

The applicant estimated the friction angle ϕ of the residual soil to be 44 degrees, based on figure 4-12, "N versus $\bar{f}_{tc}$," in "Manual on Estimating Soil Properties for Foundation Design," by F.H Kuhlhawy and P.W. Mayne, published in 1990. The applicant also estimated the elastic modulus of the residual soil to be 8.2 mPa (1,424 ksf) using the correlation given between the SPT N and elastic modulus in table 5-6, "Equations for stress-strain modulus E_s by several test methods," of Bowles (1996). The staff finds the values of friction angle and elastic modulus acceptable as these correlation equations are widely used.

The applicant also presented the material properties of the Chinle Formation in table 5.3 of Holtec Report No. HI-2188143. The Chinle Formation stratum is sandy lean clay or clayey sand. A larger unit weight has been measured in the rock portion of the stratum. The applicant determined the unit weight of Chinle Formation soil to be 1,978.1 kg/m^3 (124.05 lb/ft^3) and Chinle Formation mudrock to be 2,200.1 kg/m^3 (137.35 lb/ft^3). As discussed above in SER section 2.3.6.4.1, lower RQD values and core recoveries were typical in the upper soil portion of the Chinle Formation, which is very poorly indurated.

For this analysis, the applicant assumed that the Chinle Formation is saturated. The staff finds this assumption acceptable based on the discussion given in section 5.3, "Groundwater," of the 2018 GEI geotechnical data report. The 2018 GEI geotechnical data report interprets the observed groundwater in the monitoring wells B107(MW) and ELEA-2 as the presence of limited water in a discontinuous aquifer. The presence of water in the Chinle Formation will induce the primary consolidation or long-term settlement of the CTF foundation and the storage pads. The applicant assessed the potential primary consolidation settlement and presented the results in Holtec Report No. HI-2188143. The applicant did not account for any secondary settlement, as stated in its response to RAI 2-39, dated September 16, 2020. The staff agrees, as the secondary settlement is significantly smaller and occurs due to creep of the soil skeleton. The secondary settlement is important for highly organic or highly plastic soil deposits. As the soils at the proposed site are neither organic nor highly plastic, any secondary settlement is expected to be negligible.

The 2018 GEI geotechnical data report also measured the compression ratio, C_R (in strain per log cycle stress), of the Chinle Formation mudrock for two samples, C4 and C9, collected in boring B107 using a one-dimensional consolidation and swell test. The initial and final void ratios of the samples tested were also measured. The consolidation curve using the measured results with sample C9 does not look like the typical consolidation curve given in ASTM D2435-11, "Standard Test Methods for One-Dimensional Consolidation Properties of Soils Using Incremental Loading." Consequently, the applicant used the consolidation curve obtained using sample C4 in this analysis. Following the procedure (Casagrande method) given in the ASTM D2435-11, the applicant determined the maximum preconsolidation stress, P_c, of approximately 0.3 mPa (6 ksf). As the current overburden stress at the depth of the sample is 0.42 mPa (8.704 ksf), the applicant concluded that the Chinle Formation is normally consolidated. The staff agrees with the determination of the preconsolidation stress as it follows the well-known Casagrande method recommended in ASTM D2435-11. The results from the consolidation test give the compression ratio, C_R, of 0.007, with an initial void ratio of 0.415.

As the Chinle Formation is assumed saturated, the friction angle ϕ is taken as 0 degrees and the cohesion would be half of the compressive strength, based on section 2.18, "Unconfined Compression Test," of Das (2016). Table 4, "Rock Core Test Results," of the 2018 GEI geotechnical data report gives the measured unconfined compressive strength from three samples (C4, C6, and C6) collected from the Chinle Formation in boring B107. The average compressive strength from these three samples is 771 kPa (16.1 ksf). Therefore, the estimated cohesion is 386 kPa (8.06 ksf). The elastic modulus of the Chinle Formation was also measured on these three samples from boring B107. The average elastic modulus is 130.4 mPa (2,724 ksf). The staff accepts the values of the material properties used in the analysis as reasonable.

2.3.6.4.2.7 Methodology Used to Calculate Bearing Capacity

The ultimate bearing capacity of the foundation of the UMAX storage pads and the CTF is calculated using the methodology presented in equation (10.6.3.1.2a-1) in section 10.6.3.1.2a, "Basic Formulation," of American Association of State Highway and Transportation Officials (AASHTO) standard, "AASHTO LRFD Bridge Design Specifications," issued 2012. This method is commonly used in the industry and is consistent with other similar methods (e.g., those given in Bowles, 1996). As the applied load is borne by the underlying strata close to the surface, use of the shallow foundation method, as presented in the AASHTO Standard, is appropriate. As the subsurface is a layered strata with each layer having different material properties, the applicant used the methodology presented in Bowles (1996) to determine the equivalent material properties to be used in estimating the bearing capacity of the foundations. This averaging method to determine the equivalent material properties is commonly used in the industry and is acceptable.

To estimate the allowable bearing capacity, the applicant selected the safety factor of 3 for the static loading combination and a safety factor of 2 for the seismic loading combination. Uncertainties associated with the available soil bearing strength and the loads are combined into these factors of safety. The selected values for the safety factors are within the range typically used and are acceptable to the staff.

2.3.6.4.2.8 Methodology Used to Calculate Immediate Settlement

The immediate settlement is the deformation of the subsurface as the foundation is loaded by construction of the proposed facility. In practice, immediate settlement takes place within 7 to 10 days after application of the construction load (Bowles, 1996). The immediate settlement is generally accommodated in the facility design and construction.

Although the subsurface is not generally a truly elastic half-space, elastic solution assuming uniform, homogeneous, isotropic, semi-infinite, elastic half-space has produced acceptable results and is widely used in the industry. In Holtec Report No. HI-2188143, the applicant used the elastic solution, given in section 5-6, "Immediate Settlement Computations," of Bowles (1996), to calculate the immediate settlement as a result of constructing the storage pads and the CTF. This solution has two influence factors to account for the foundation (pad) shape, the embedment depth, and the particular strata characteristics: the Steinbrenner influence factor I_s and the depth influence factor I_f. Table 5-2 of Bowles (1996) provides the Steinbrenner influence factor, and figure 5-7 of Bowles (1996) gives the depth influence factor. The staff finds that the method presented by Bowles (1996) is widely used in the industry to estimate the immediate settlement of a footing and is acceptable for determining the immediate settlement at the proposed site.

2.3.6.4.2.9 Methodology Used to Calculate Long-Term or Primary Consolidation Settlement

The primary consolidation settlement occurs in a cohesive soil, such as clay, due to dissipation of excess pore fluid pressure caused by the applied load. Gradual expulsion of the pore fluid leads to compression of the soil skeleton, resulting in a decrease of soil volume. This dissipation is time dependent as clay generally has low permeability and requires a long time to complete the pore fluid expulsion. Consolidation is taken as complete when all excess pore pressure has dissipated. As discussed above in SER section 2.3.6.4.2.5, the secondary compression settlement at the proposed site would be negligible.

The applicant designed the SFPs and the CTF foundation as a mat foundation. Although a mat foundation reduces the settlement of the foundation (Bowles, 1996), a small amount of settlement is expected for both the SFPs and the CTF. The applicant used the definition of consolidation of a soil layer to calculate the long-term or consolidation settlement of the CTF and the UMAX storage pads. Equation (10.6.2.4.3-5) of the AASHTO standard or equation (2-44) of Bowles (1996) was used since the Chinle Formation at the proposed site is found to be normally consolidated, as discussed in SER section 2.3.6.4.2.5. Although the overlying residual soil does not undergo consolidation (assumed dry soil), it provides the load on the Chile Formation stratum. This load and the weight of the constructed facility induce the consolidation settlement of the Chinle Formation, which increases with time until completed.

The consolidation settlement was calculated for each stratum of the subsurface and summed to determine the total consolidation settlement below the UMAX storage pads and the CTF. In Holtec Report No. HI-2188143, the subsurface beneath the UMAX storage pads and CTF was subdivided into sublayers for better estimation of the consolidation settlement. The Chinle Formation soil layer was subdivided into four sublayers: the upper three were each 2.4 m (8 feet) thick, and the bottommost one was 2.1 m (7 feet) thick. The Chinle Formation mudrock was divided into two sublayers, each 22.6 m (74 feet) thick.

2.3.6.4.2.10 Calculated Bearing Capacity of the UMAX Storage Pads

In Holtec Report No. HI-2188143, appendix A, the applicant estimated the ultimate and allowable bearing capacities of the UMAX storage pads. The bearing capacities were calculated using the procedure for the layered subsurface given in Bowles (1996). The estimated depth of influence of the storage pad foundation for calculating the bearing capacity is 66 m (216 feet) from the bottom of the storage pads, following the method given in Bowles (1996). Consequently, the subsurface of the storage pads was modeled as a two-layer system: a relatively thin layer of residual soil overlying the Chinle Formation. The equivalent cohesion C and friction angle ϕ were calculated using the formulas given by Bowles (1996). The values of various terms in equation (10.6.3.1.2a-1) of section 10.6.3.1.2a of the AASHTO standard were determined using the equivalent subsurface properties C and ϕ.

The applicant determined that the ultimate bearing capacity of the foundation of the UMAX storage pads is 772.8 kPa (16.14 ksf), following the method in Bowles (1996). Using the assumed factor of safety of 3 to account for the uncertainties, the allowable bearing capacity is 257.6 kPa (5.38 ksf) for the static load combination. The actual bearing pressure from static loads (storage pad weight plus weight of all VVMs) is 159.7 kPa (3.33 ksf). Therefore, the applicant concluded that the foundations have adequate bearing capacity to withstand the fully loaded pads without encountering a foundation failure. The staff finds that the bearing capacities of the UMAX storage pads at the proposed site have been estimated using an acceptable method. The estimated bearing capacity of the subsurface has enough margin against excessive penetration of the foundations under static loads. Therefore, the staff accepts the assertion that the storage pads at the proposed site will not sustain a foundation failure under the static loads.

Using a factor of safety of 2 for the seismic loads, the applicant estimated the allowable seismic bearing capacity to be 386.4 kPa (8.07 ksf). The applicant also calculated the bearing pressure of a storage pad under the seismic load. The seismic force is calculated by applying the PGA of 0.25g to the UMAX storage pad. Both horizontal and vertical seismic forces were calculated. These additional forces cause eccentricity of the loads applied to the storage pads. The applicant calculated the load eccentricity and its effects on the bearing pressure using the procedure given in section 4-6, “Footings with Eccentric or Inclined Loadings,” of Bowles (1996). The applicant calculated the effective pad dimensions and the seismic bearing pressure on the foundation following Bowles (1996). As the calculated seismic bearing pressure 207.7 kPa (4.34 ksf) is smaller than the allowable seismic bearing capacity 386.4 kPa (8.1 ksf), the applicant concluded that the foundation of the storage pads has adequate seismic bearing capacity to withstand the seismic loads. Based on the preceding discussion, the staff finds that the methodologies selected to estimate the bearing capacities are widely used, and it also accepts the conclusion that the UMAX storage pads at the proposed facility will have adequate bearing capacity to withstand the seismic loads without foundation failure.

2.3.6.4.2.11 Calculated Bearing Capacity of the Canister Transfer Facility

In appendix B to Holtec Report No. HI-2188143, the applicant presented the ultimate and allowable bearing capacities of the CTF foundation, calculated using the procedure for the layered subsurface given in Bowles (1996). The depth of influence for calculating the bearing capacity was 5.8 m (18.9 feet) from the bottom of the CTF foundation. Consequently, the subsurface of the CTF was modeled as a two-layer system: a layer of residual soil overlying a layer of Chinle Formation. The equivalent cohesion C and friction angle ϕ were calculated following Bowles (1996). The values of various terms in equation (10.6.3.1.2a-1) of section 10.6.3.1.2a of the AASHTO standard were determined using the equivalent subsurface properties C and ϕ.

The applicant estimated the ultimate bearing capacity of the subsurface beneath the foundation of the CTF to be 2,504 kPa (52.3 ksf) under static loads. Using the assumed factor of safety of 3 to account for the uncertainties, the allowable bearing capacity is 833 kPa (17.4 ksf) for the static load combination. The actual bearing pressure from the static loads of the CTF is 235 kPa (4.9 ksf). Therefore, the applicant concluded that the subsurface below the proposed CTF would have adequate bearing capacity to withstand the static load imposed by the CTF. The staff agrees with the assessment as it used an acceptable method with site material properties.

Using a factor of safety of 2 for the seismic loads, the applicant calculated the allowable seismic bearing capacity of the subsurface beneath the CTF to be 1,253 kPa (26.2 ksf). The applicant also calculated the bearing pressure on the subsurface from the site-specific seismic load of PGA 0.25g to the CTF foundation. Both horizontal and vertical seismic forces were calculated. The eccentricity of the loads caused by the seismic loads was applied on the CTF foundation to reduce the effective pad dimensions, and the resultant seismic bearing pressure on the foundation was calculated following Bowles (1996). As the calculated seismic bearing pressure 340 kPa (7.1 ksf) is smaller than the allowable seismic bearing capacity 1,254 kPa (26.2 ksf), the applicant concluded that the foundation of the storage pads has adequate seismic bearing capacity to withstand the seismic loads. The staff agrees with the assessment that the subsurface below the CTF will have adequate bearing capacities to withstand the static and seismic loads, as the assessment used an acceptable method.

2.3.6.4.2.12 Calculated Immediate Settlement of the UMAX Storage Pads

Using the methodology presented in Bowles (1996), the applicant calculated in Holtec Report No. HI-2188143 the immediate settlement at the midpoint of the subgrade beneath the UMAX storage pads to be 3.8 cm (1.5 inches). The staff accepts the results as the applicant used an acceptable methodology to estimate the immediate settlement. An immediate settlement of 3.8 cm (1.5 inches) is small compared to the size of the storage pads and can easily be accommodated during construction of the storage pads. It should be noted that this calculation of the immediate settlement inherently assumes that the total weight of the loaded pads is applied on the subsurface instantaneously. In reality, the construction of the storage pads would progress in stages. At each stage, the immediate settlement will increase with the addition of new loads until completion of the construction.

2.3.6.4.2.13 Calculated Immediate Settlement of the Canister Transfer Facility Foundation

The applicant calculated the immediate settlement at the midpoint of the subgrade beneath the CTF foundation using the methodology of Bowles (1996) to be 2.5 mm (0.1 inch). The staff accepts the estimated immediate settlement of the CTF foundation as an acceptable methodology was used. This estimated immediate settlement can be accommodated during construction of the CTF.

2.3.6.4.2.14 Calculated Consolidation Settlement of the UMAX Storage Pads

The applicant determined the stress removed due to initial excavation and also calculated the stress added by construction of the storage pad (i.e., the weight of the CLSM layers, the ISFSI and SFPs, and the loaded VVMs) at each sublayer of the subsurface. The stress increases at a point below the pad are a function of the location of the point and the load applied by the pad, assuming elastic media (Bowles, 1996). The applicant used the approximate “2:1 method,” as given in Das (2016), to estimate this increase in stress in the subsurface. The staff accepts this method for estimating the stress increase in the subsurface from construction of the storage pads because the method is widely used. The increase in stress at the midpoint of the sublayer beneath the storage pad, resulting in the consolidation settlement, is the original in situ stress at that point minus the stress reduction due to removal of the materials excavated (the caliche layer along with top portion of the residual soil layer) to construct the storage pads, plus the increased stress from construction of the storage pad.

The applicant calculated the total consolidation settlement under the storage pads as a summation of the consolidation settlement estimated at each of the six sublayers. The total consolidation settlement of the UMAX storage pads with 25 × 10 loaded VVMs is estimated to be 6.4 mm (0.25 inch), which is slightly larger (only in the second place after the decimal) than that specified in SAR table 5.0.1, "Comparison of DBLs for HI-STORM UMAX System and Site-Specific Loads for HI-STORE CIS Facility." SAR table 5.0.1 refers to table 2.3.2, "Design Data for HI-STORM UMAX ISFSI," of the UMAX FSAR, which gives the maximum long-term settlement of the SFP to be 5.1 mm (0.2 inch). The UMAX FSAR referred to Holtec Report No. HI-2125239, "Structural Analysis of HI-STORM UMAX ISFSI Structures" (proprietary), latest revision (i.e., Revision 1), dated July 14, 2014, for estimating the permissible long-term settlement to be used in designing the SFP and the ISFSI pad, and to show compliance with the design code ACI 318-14.

The long-term settlement limit specified in table 2.3.2 of the UMAX FSAR was calculated based on a 5 × 5 VVM array on a 28 m × 28 m (92 foot × 92 foot) pad. In Holtec Report No. HI-2125239, the corresponding subgrade modulus was used in a finite element analysis to determine the stress field and the moments in the SFP and ISFSI pads and compared with the ACI 318-14 code to show compliance. The staff notes that the long-term settlement from a 5 × 5 VVM array at the proposed facility is 3.8 mm (0.15 inch).

Holtec Report No. HI-2188143 notes that Holtec Report No. HI-2125239 uses a softer soil half-space that would result in 5.1 mm (0.2-inch) settlement from the 5 × 5 VVM array. The estimated soil modulus of 23.2 MPa (3,361 psi) corresponding to 5.1 mm (0.2 inch) of settlement has been reduced to 22.1 MPa (3,200 psi). Consequently, the SFP can withstand a larger settlement by at least 5 percent.

In addition, table 3.4.5, "Moment Results and Corresponding Minimum Safety Factors for the ISFSI Structures," of the UMAX FSAR shows that the SFP has a safety factor of more than 2.0 against flexural bending. Therefore, the SFP design includes enough margin to withstand a small increase (in the second place after the decimal) in the estimated long-term settlement without causing any flexural failure. The staff agrees with Holtec's assessment of the estimated long-term settlement of the UMAX storage pads and the maximum settlement prescribed in table 3.4.5 of the UMAX FSAR. The staff notes that the calculated factor of safety is more than 2.0 in all cases analyzed, as listed in table 3.4.5 of the UMAX FSAR. Based on these results, the staff concludes that the SFP has a significantly larger reserve strength than that required by ACI 318-14. As noted in the UMAX FSAR, the safety factors were determined using a conservative analysis by ignoring the dynamic increase factor for the SFP made of reinforced concrete. Because of the large factor of safety available, the staff concludes that a settlement increase in the second place after the decimal will cause a negligible decrease in the reserve strength of the SFP and will not result in a failure.

2.3.6.4.2.15 Calculated Consolidation Settlement of the Canister Transfer Facility

The applicant estimated the total consolidation settlement at the center of the CTF foundation pad using the same methodology and material properties of the subsurface underneath as those used in calculating the potential consolidation settlement of the storage pads. The estimated consolidation settlement is 2.5 mm (0.1 inch). The staff finds this estimated settlement acceptable as the applicant has used an acceptable methodology with site-specific material properties to calculate the consolidation settlement. It should be noted that unlike the storage pads at the proposed facility, there is no limit on the consolidation settlement of the CTF foundation pad.

2.3.6.4.2.16 Summary and Conclusions

In Holtec Report No. HI-2188143, the applicant calculated the bearing capacity and both the immediate and long-term consolidation settlements of the storage pads and the CTF foundation at the proposed facility. The applicant followed the AASHTO standard to calculate the ultimate bearing capacity of both the SFP and the CTF foundations. The applicant calculated the allowable bearing capacities of each foundation using appropriate safety factors. The calculated allowable bearing capacities show significant margins against both the static and the site-specific seismic loads. Consequently, the staff finds the design of both the storage pads and the CTF acceptable as they have adequate bearing capacities.

The applicant also calculated the immediate and consolidation settlements at the centers of the SFP and the CTF foundations. The staff finds both methodologies to estimate the immediate and long-term settlements acceptable.

The applicant calculated the consolidation settlement of the CTF foundation pad to be 2.5 mm (0.1 inch). The CTF design is expected to accommodate this small settlement. Being a mat foundation, any differential settlement is expected to be small. The estimated consolidation settlement of the storage pads is 6.4 mm (0.25 inch). As discussed above in SER section 2.3.6.4.2.14, the estimated settlement exceeds the specified value in SAR table 5.0.1 and table 2.3.2 of the UMAX FSAR only in the second place after the decimal. The maximum settlement of 5.1 mm (0.2 inch) has some margin as the corresponding soil medium used to derive it was somewhat softer than assumed in the analysis. Based on table 3.4.5 of the UMAX FSAR, the staff finds there is enough margin in the SFP design to withstand such a small increase in settlement without causing a flexural failure. Therefore, the staff finds the expected consolidation settlement of the storage pads at the proposed site to be acceptable. In addition, the applicant committed to inspect the ISFSI pads and the SFPs annually to ensure that the pads are free of visible cracks, as stated in SAR table 10.3.1, “Maintenance and Inspection Activities for the HI-STORM UMAX VVM Systems,” and table 18.6.1, “Periodic Inspection Frequency of HI-STORE CIS ISFSI Components.” The applicant would inspect ISFSI settlement every 5 years to ensure that the VVM settlement is within the design basis, as stated in SAR table 10.3.1. Any cracking of the ISFSI concrete pad due to excessive long-term consolidation settlement of the SFP subgrade would be detected on the ISFSI pad, and remedial measures would be taken.

The staff reviewed section 5-13, “Reliability of Settlement Computations,” of Bowles (1996) to assess the reliability of the predicted consolidation settlement of the foundation pads of the proposed facility. The consolidation settlements are typically adequately predicted, and the predicted consolidation settlement is generally conservative within acceptable limits. Predictions are better for inorganic clays, such as the Chinle Formation. Table 8.1, “Settlement of Mat Foundations on Sand and Gravel,” of Das (2016) lists case studies of observed settlement against maximum predicted settlement. The observed settlement is generally smaller than the predicted settlement. Therefore, based on the preceding discussion, the staff finds that the proposed design of the storage pads and the CTF foundation would have adequate allowable bearing capacities under static and seismic loads and would undergo limited but tolerable immediate and consolidation settlements.

The staff also concludes that the design of the storage pads and the CTF foundation has been adequately described for use in other SAR sections to develop the design bases of the proposed facility, perform additional analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.90(a), 10 CFR 72.103(d), and 10 CFR 72.122(b)(1).

2.3.6.4.3 Liquefaction Potential

The applicant analyzed the potential liquefaction of the subsurface at the proposed site and submitted the 2020 GEI soil liquefaction calculation documenting the analysis. SAR section 2.6.4 summarizes the results of the analysis. The analysis used the methodology described by Youd et al. (2001), which is widely used in the industry. The staff finds that the methodology described by Youd et al. (2001) has been developed from the 1996 National Center for Earthquake Engineering Research (NCEER) and 1998 NCEER/National Science Foundation (NSF) workshop on the evaluation of liquefaction resistance of soils as a consensus recommended procedure to assess the potential for liquefaction of soil. RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” issued November 2003, references Youd et al. (2001) extensively and has adopted the methodology. The National Academies of Sciences, Engineering, and Medicine (NASEM, 2016) has also recommended this method as an acceptable approach to assess triggering of liquefaction in soil using the stress-based approach. Therefore, the staff concludes that it is appropriate to use the method by Youd et al. (2001) to assess whether the soils at the proposed site would liquefy during a design-basis earthquake.

The 2020 GEI soil liquefaction calculation used the information from borings B-101, B-101A, B-102, B-105, B-105A, B-106, B-107, B-108, and B-109 at the proposed site to assess the potential for liquefaction, as the SPT was conducted in these borings. Other borings at the site have been logged using drill cuttings; therefore, information from these borings is not suitable for liquefaction assessment.

The safety factor against liquefaction has been evaluated at 95 locations where SPT measurements have been conducted in these nine borings. In addition, a PGA of 0.25g from a magnitude 5.5 earthquake is assumed following SAR table 4.3.3, based on the response spectrum for the DECE with a return period of more than 10,000 years. The applicant corrected the field-measured SPT blow count N values for the overburden pressure, hammer energy, sampling equipment, borehole diameter, rod length, and fines content to determine the equivalent $(N_1)_{60}$ values in clean sand to assess the potential for liquefaction, following Youd et al. (2001). The correction factors used are given by Youd et al. (2001). The staff agrees with the approach to determine the corrected $(N_1)_{60}$ values from the field-measured SPT blow count N values using the correction factors suggested by Youd et al. (2001), as RG 1.198 and NASEM (2016) have recommended it.

Only a limited number of samples were tested for particle gradation in the caliche (sand and gravel with silt) and residual soils (clayey sand or sandy clay with trace gravel). Consequently, the 2020 GEI soil liquefaction calculation assumed that the percent of fines to be approximately the lower bound value of each material: 0 percent fines in the caliche and 20 percent fines in residual soil. Split spoon samples were obtained from poorly indurated mudstone from the upper portion of the Chinle Formation that has been classified as very dense clayey sand or hard sandy lean clay. The fines content of this material is greater than 50 percent, as given in attachment H, "Geotechnical Laboratory Soil Test Data by GEI," of the 2018 GEI geotechnical data report. Because the measured clay content is more than 50 percent by weight, the applicant concluded that the Chinle Formation is not susceptible to liquefaction, following Youd et al. (2001). Therefore, the assessment for liquefaction is limited to the topsoil, caliche, and residual soil layers only. The staff agrees with this conclusion because the assessment follows the widely used methodology by Youd et al. (2001) and is also supported by RG 1.198 and NASEM (2016). Consequently, the staff finds the conclusion that the Chinle Formation would not liquefy due to its high fines content to be acceptable.

Additionally, the 2020 GEI soil liquefaction calculation assumed the highest measured groundwater elevation at the proposed site to be in well ELEA-2 at elevation 1,065.6 m (3,495.9 feet) AMSL for all borings at the site. This elevation of the water table is below the elevations at which all samples are evaluated for assessing the liquefaction potential. Consequently, the applicant concluded that the subsurface at the proposed site is not susceptible to liquefaction. The staff finds this acceptable and consistent with Youd et al. (2001), RG 1.198, and NASEM (2016).

The factor of safety against liquefaction was determined for each of the 95 measured locations of N values in the topsoil, caliche, and residual soil in the borings following the method given in Youd et al. (2001) and RG 1.198. Youd et al. (2001) states that soils with corrected SPT blow count $(N_1)_{60}$ values larger than 30 blows per foot are too dense to liquefy in an earthquake of any magnitude. Results given in the 2020 GEI soil liquefaction calculation show that, except for three samples from the topsoil in boring B-109, located at the proposed site of the CTB, all the other 92 samples have corrected $(N_1)_{60}$ values greater than 30 and, therefore, are not liquefiable. In addition, these three samples from the topsoil in boring B-109 are above the water table; therefore, they are classified as non-liquefiable. Additionally, SAR figure 2.6.8 shows that the foundation of the CTB would be significantly below the topsoil. Therefore, the topsoil layer will be excavated to construct the CTB and will not be susceptible to liquefaction after the proposed CTB is constructed.

In summary, the staff finds that the applicant has used several techniques to screen out the proposed site from having the potential to liquefy during a design-basis earthquake. These techniques are extensively used in the industry (NASEM, 2016; Youd et al., 2001) and recommended by RG 1.198. Using these techniques, the applicant has shown that the subsurface material layers at the proposed site are not susceptible to liquefaction during an earthquake. The Chinle Formation is not liquefiable because of its high clay content. Soil layers above the Chinle Formation are not liquefiable as they are too dense (high SPT $(N_1)_{60}$ values). In addition, all the subsurface layers are above the groundwater table. Therefore, based on the preceding discussion, the staff concludes that the soils both at the cask storage pads and the CTB areas at the proposed site are not susceptible to liquefaction. The information presented is acceptable for use in other sections of the SAR to develop the design bases of the spent fuel storage facility at the proposed site, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.24(a), 10 CFR 72.90(a)–(b), 10 CFR 72.92(a)–(c), 10 CFR 72.103(c)–(d), and 10 CFR 72.122(b) with respect to subsurface liquefaction.

2.3.6.5 Slope Stability

SAR section 2.6.5 discusses the potential slope stability hazard at the proposed site. As discussed in SAR section 2.1.2, a high point at the proposed site is located near its southern border, and gentle slopes lead to two drainages (Laguna Plata and Laguna Gatuna). The topography of the proposed site is mostly flat (SAR figure 2.6.5) and gently slopes between 0 and 3 percent to the drainage basins. Elevation of the terrain ranges from approximately 1,076 m (3,530 feet) to 1,082 m (3,550 feet) AMSL. Therefore, the applicant concluded that there is no risk from slope instability or landslides in the vicinity of the proposed site.

The staff reviewed SAR figure 2.6.15, “Elevation Contours at the Site.” In addition, the staff reviewed SAR figures 2.1.9, 2.1.10, 2.1.11, and 2.4.7, which illustrate the site topography using the elevation contours. The staff agrees with the applicant’s assessment and concludes that the terrain at the proposed site is mostly flat. The information presented is acceptable for use in other sections of the SAR to develop the design bases of the spent fuel storage facility at the proposed site, perform additional safety analyses, and demonstrate compliance with the regulatory requirements in 10 CFR 72.24(a), 10 CFR 72.90(a)–(b), 10 CFR 72.92(a)–(c), 10 CFR 72.103(c)–(d), and 10 CFR 72.122(b) with respect to slope stability.

2.3.6.6 Construction Excavation

In SAR section 2.6.6, the applicant described the excavations to be made at the site to construct the structures of the proposed facility. During construction of Phase 1 of this proposed facility, multiple excavations will be made to construct the below-ground facilities. The below-ground structures include the CTF and the UMAX storage pads. The expected excavation depth is approximately 7.6 m (25 feet) for both structures. The caliche layer and the top 4.0 m (13 feet) of the residual soil layer will be excavated for constructing the site structures. The applicant does not expect to encounter the Chinle Formation in any excavation.

The construction will have a minimum 1:1 slope around the excavation for accommodating construction vehicle access. The applicant expects to generate approximately 94,805 m^3 (124,000 cubic yards (yd^3)) of caliche and 92,893 m^3 (121,500 yd^3) of residual soil spoils. The applicant proposed to use the excavated residual soil as the backfill material of the excavated areas and expects to use approximately 18,350 m^3 (24,000 yd^3) for backfilling. Residual soil was selected as the backfill material because it meets the minimum density and shear wave velocity requirements for Space B in SAR figure 4.3.1.

The bottom of the excavated pit at the appropriate elevation (top of Space C in SAR figure 4.3.1) will be prepared for constructing the SFP. Before construction of the pad, the residual soil surface will be proof rolled by a heavy vibration compactor. A professional engineer licensed in New Mexico or an approved representative will be present during proof rolling to identify any areas of soft, yielding soil, which may require over-excavation and soil replacement. The compaction would be conducted at or close to the optimum moisture content, as indicated by the modified Proctor test procedure specified in ASTM D1557-12R21, “Standard Test Methods for Laboratory Compaction Characteristics of Soil Using Modified Effort (56,000 ft-lbf/ft^3 (2,700 kN-m/m^3),” dated 2021. The soil in Space C shall be confirmed to have reached a compaction of at least 95 percent of the modified Proctor maximum dry density, in accordance with ASTM D1557-12R21.

After the subgrade is prepared, the SFP will be constructed and the UMAX CECs will be placed on top of the SFP. Space A and Space B in SAR figure 4.3.1 will be backfilled. Space A will be backfilled with CLSM or lean concrete with the minimum strength specified in SAR table 4.3.3. The ISFSI pad will be placed on top of the CLSM or lean concrete layer to provide a riding surface for the loaded transporter. Fill materials will bring Space B back up to the finished grade level.

The staff reviewed the description of construction excavations presented in SAR section 2.6.6. The staff finds that the applicant will use ASTM D1557-12R21 to prepare the subgrade to construct the storage pad and accepts this approach as appropriate. The properties of each subgrade layer of the storage pads will be controlled by the UMAX FSAR design, as presented in SAR table 4.3.3. The staff notes that the UMAX storage system is an NRC-approved system to store spent nuclear fuel in an ISFSI. The staff also finds that a significant amount of backfill material for Space B is available, as only a small portion of the excavated residual soil (18,350 m^3 (24,000 yd^3)) is needed to use as backfill material. Based on the preceding discussion, the staff finds that the excavations needed to construct the proposed facility have been adequately described for use in other SAR sections to develop the design bases for the facility, perform additional analyses, and demonstrate compliance with the regulatory requirements as specified in 10 CFR 72.103(d).

2.4 Evaluation Findings

Based on its review of the information in the application, the staff finds the following:

- The application provides an acceptable description and safety assessment of the site on which the ISFSI is to be located, in accordance with 10 CFR 72.24(a).

- The proposed site complies with the criteria of 10 CFR Part 72, Subpart E, “Siting Evaluation Factors,” as required by 10 CFR 72.40(a)(2).

- The analyses of off-normal and accident events and conditions show that the design of the HI-STORE CIS Facility will acceptably meet the requirements without endangering public health and safety, in compliance with the overall requirements of 10 CFR 72.122.

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3 OPERATION SYSTEMS

In chapter 3, “Operation Systems,” of Revision 0T of the safety analysis report (SAR), dated January 20, 2023, Holtec International (the applicant) described its operations at the proposed HI-STORE Consolidated Interim Storage (CIS) Facility, including receiving and inspecting transportation casks with canisters containing spent nuclear fuel (SNF), transferring canisters from the transportation packages to transfer casks, transferring canisters from transfer casks to storage modules at the storage pads, and other operations of the facility. These other operations relate to surveillance of the storage system, security, radiation protection, maintenance, removal of canisters from the facility, operating systems for the safe storage of SNF, and operation support systems.

3.1 Scope of Review

The staff evaluated the applicant’s operation systems for the HI-STORE CIS Facility by reviewing the information in the SAR; the applicant’s responses to the staff’s requests for supplemental and additional information; and the SARs, final SARs (FSARs), and previous staff evaluations for the multipurpose storage canisters (MPCs) to be stored at the facility. This includes portions of these SARs and FSARs that the applicant incorporated by reference in the HI-STORE CIS Facility SAR. The staff reviewed the information in SAR chapter 3 to determine whether the applicant provided a clear and complete description of all operations, including systems, equipment, and instrumentation related to the handling and storage of SNF, confinement of nuclear material, and management of expected and potential radiological dose. In addition, the staff reviewed applicable information that the applicant incorporated by reference.

3.2 Regulatory Requirements

The regulatory requirements relevant to the overall function of the facility and the operation of the separate functional subsystems of the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste”:

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.40, “Issuance of license”
- 10 CFR 72.44, “License conditions”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI [independent spent fuel storage installation] or MRS [monitored retrievable storage installation]”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.124, “Criteria for nuclear criticality safety”

- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”
- 10 CFR 72.150, “Instructions, procedures, and drawings”

3.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed and evaluated the operation systems of the proposed site discussed in chapter 3 of SAR Revision 0T, documents cited in or attached to the SAR, the applicant’s responses to the staff’s requests for supplemental and additional information, and other relevant documents and literature. The staff reviewed and evaluated the functions and functional subsystems associated with receipt, transfer, maintenance, and retrieval of SNF at the HI-STORE CIS Facility. Specifically, the staff reviewed and evaluated the operations and the canister handling systems at the HI-STORE CIS Facility. The staff used NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000, to support the review.

3.3.1 Description of Operations

In SAR section 3.1, “Description of Operations,” the applicant provided an overview of the following specific operations that will be performed at the HI-STORE CIS Facility:

- receipt and inspection of incoming transportation casks with canisters containing SNF
- transfer of canisters from HI-STAR 190 transportation packages to HI-TRAC CS [concrete shielded] transfer casks at the canister transfer facility (CTF)
- transfer of canisters from HI-TRAC CS transfer casks to subterranean HI-STORE vertical ventilated modules (VVMs) at the ISFSI pad
- surveillance of the HI-STORM UMAX [underground maximum safety] canister storage system
- security
- radiation surveillance (radiation protection)
- maintenance
- removal of canisters from the facility
- inventory documentation management

In SAR section 3.1, the applicant established that compliance with 10 CFR Part 72 begins when the transportation cask enters the cask transfer building (CTB).

SAR figure 3.1.1, “Cask Handling Summary Illustrations,” illustrates the sequence of operations for canister receipt, transfer, and placement into storage. This section of the SAR identifies the important-to-safety structures, systems, and components (SSCs) used at the facility, including key design features and functions for each SSC. The staff compared the description of the operations and SSCs in SAR section 3.1 to descriptions of the same transportation and storage systems and SSCs appearing in other SARs and FSARs previously reviewed and found acceptable by the staff. The other SARs and FSARs describing these SSCs and operations include the HI-STAR 190 Package SAR, Revision 3, dated November 2, 2018; the HI-STORM UMAX Canister Storage System FSAR, Revision 3, dated June 29, 2016; and the HI-STORM FW [flood and wind] MPC Storage System FSAR, Revision 4, dated June 24, 2015. In comparing the descriptions of operations, functions, and equipment provided in SAR section 3.1 with the corresponding descriptions in the other SARs and FSARs, the staff considered how they met the applicable regulatory requirements for structural integrity, thermal, shielding, criticality, confinement, radiation protection, and quality assurance and found the descriptions sufficiently detailed and compatible. Chapters 5, 6, 7, 8, 9, 11, and 12, respectively, of this safety evaluation report (SER) summarize the staff's specific evaluations for these disciplines.

The staff reviewed the applicant's discussion of the operations at the HI-STORE CIS Facility in SAR chapter 3 and evaluated the applicant's commitment to ensure that the design of the SSCs involved in the handling and storage of SNF complies with the overall requirements in 10 CFR 72.122. SAR chapter 3 describes the short-term operations at the HI-STORE CIS Facility from the receipt of SNF in a HI-STAR 190 transportation cask to final storage in a HI-STORE VVM at the ISFSI pad. As defined in the SAR glossary, short-term operations also include the activities involved in retrieving the canisters and packaging them for offsite shipment. The staff's evaluation of selected natural phenomena events and use of administrative controls in short term operations is summarized in chapter 15.3.2.3 of this SER.

The staff reviewed the described operations and how different SSCs are used to conduct them, as described in SAR chapter 3. In SAR chapter 4, "Design Criteria for the HI-STORE CIS Systems, Structures and Components," the applicant assessed the demands on each SSC from conducting these operations, including considerations for natural phenomena events that may occur during these operations and the environmental conditions under which these operations will be conducted. The staff compared these demands with the descriptions in SAR chapter 3 in evaluating the design acceptance criteria established by the applicant in SAR chapter 4. The staff evaluated the structural design of each SSC in chapter 5 of this SER to confirm that the design acceptance criteria are satisfied and thus ensure the expected performance of the SSC during the conduct of all operations at the HI-STORE CIS Facility.

Based on the staff's review of the commitments and descriptions in SAR chapter 3, the staff concludes that the facility meets the requirements of 10 CFR 72.122(b)(1) and 10 CFR 72.122(b)(2)(i) and (ii) by including all forms of demands made on the SSCs during their operational life, including demands related to short-term operations.

Regarding the description of thermally relevant operations in SAR section 3.1.4.2, "Transfer of Canister from Transportation Cask to HI-TRAC CS," and SAR section 3.1.4.3, "Placement of the Canisters into the Vertical Ventilated Modules (VVMs)," the staff determined that those descriptions are consistent with the applicant's thermal analyses, which the staff reviewed and found acceptable in SER chapter 6, because the conditions of the described operations are consistent with the conditions of normal operations considered in the applicant's thermal analyses.

The staff reviewed information regarding canister receipt inspection leak testing in SAR section 3.1.4 and found it consistent with descriptions of receipt inspection leak testing in SAR chapters 9 and 10, entitled “Confinement Evaluation,” and “Conduct of Operations,” respectively. The staff evaluates receipt inspection leakage rate testing of each MPC in chapter 9 of this SER.

The staff reviewed the information regarding radiation surveillance in SAR section 3.1.4.6, “Health Physics Operations,” and found that information consistent with the evaluation of radiation protection in SAR chapter 11, “Radiation Protection Evaluation.” SER chapter 11 contains the staff's review of the applicant’s radiation protection program.

Based on the evaluation provided above, the staff finds the descriptions of operations and SSCs acceptable and that they meet the requirements of 10 CFR 72.24(b).

3.3.2 Spent Fuel Handling Systems

SAR section 3.2, “Spent Fuel and High-Level Waste Handling Systems,” and proposed Technical Specification 4.2.1, “Storage System,” discuss the fuel handling operations and the general technical requirements and criteria for the necessary equipment. SNF canisters arrive at the HI-STORE CIS Facility inside HI-STAR 190 transportation packaging with impact limiters installed. These protect the spent fuel canisters and provide them with shielding and cooling. Radiological surveys are performed for all incoming canisters upon receipt at the facility. The applicant will return a canister to the shipper if contamination above the acceptance levels is discovered. The fuel handling equipment, which is used inside the CTB, consists of the cask handling crane, the spent fuel transfer cask (the HI-TRAC CS), and the HI-PORT cask transport vehicle. The HI-STORM UMAX canister storage system FSAR, Revision 3; the HI-STORM FW MPC storage system FSAR, Revision 4; and the HI-STAR 190 transportation package SAR, Revision 3, contain descriptions of certain cask and canister handling operations that the staff has previously found acceptable.

The unloading of each transportation package is described in detail in chapter 7 of the HI-STAR 190 transportation package SAR, which the staff previously reviewed and found acceptable.

Transferring the MPCs unloaded from the transportation package to the HI-TRAC CS occurs in the CTB. The staff reviewed the cask transfer operations and finds that the applicant has provided a detailed description of the operations and corresponding safety analyses that include normal, off-normal, and accident conditions and potential radiological effects to the general public beyond the controlled area boundary and to occupational workers who are performing these operations. Based on its review, the staff finds that safety analyses for the canister transfer operations are consistent with the canister transfer design, with considerations of the “as low as reasonably achievable” (ALARA) principle, and hence acceptable.

The staff reviewed the canister handling operations using the HI-STAR 190 transportation system, the HI-STORM UMAX canister storage system, and the HI-TRAC CS and finds that handling operations at the HI-STORE CIS Facility are consistent with the handling operations described in the previously approved storage system SARs.

The staff reviewed the description of the operating systems in SAR sections 3.1 and 3.2, proposed Technical Specification 4.2.1, and relevant information in the appropriate sections of SAR chapters 1, 4, and 5. The staff reviewed the ALARA considerations in chapter 11 of this SER. Based on the review described above and in chapter 11 of this SER, the staff found the design and operations in accordance with the requirements of 10 CFR 72.104(b) and 10 CFR 72.106, "Controlled area of an ISFSI or MRS."

3.3.3 Other Operating Systems

In SAR section 3.3, "Other Operating Systems," the applicant stated that the safe storage of spent fuel does not require other operating systems (i.e., systems other than those already addressed in SAR sections 3.2, 3.4, and 3.6 for handling spent fuel, supporting operations, and analytical sampling, respectively). The staff reviewed the SAR and verified that the HI-STORE CIS Facility does not rely on any other operating systems that are important to safety that are not addressed in other sections of this SER. Therefore, the staff finds the applicant's determination that no additional system descriptions are needed to be acceptable.

3.3.4 Operation Support Systems

SAR section 3.4.1, "Instrumentation and Control Systems," states that operation of the HI-STORE CIS Facility is passive and self-contained, thereby not requiring control systems to ensure the safe operation of the system. In addition, SAR sections 3.1.5.4, "Instrumentation," and 3.4.1 state that temperature monitors (including data recorders and alarms) used to monitor temperatures and thermal performance associated with the cask during storage would be classified as important to safety if used as the sole means of surveillance of the storage systems. SAR section 16.1, "Functional/Operating Limits, Monitoring Instruments, and Limiting Control Settings"; section B 3.1.2, "SFSC Heat Removal System," of SAR appendix 16.A, "Technical Specification (LCOs) Bases for the Holtec HI-STORE CIS Facility"; and the applicant's proposed technical specification (TS) Surveillance Requirement 3.1.2 indicate that the HI-STORE CIS Facility has two surveillance techniques for confirming that the spent fuel storage cask (SFSC) heat removal system is operable to support TS Limiting Condition of Operation (LCO) 3.1.1: (1) a visual verification and (2) the use of a temperature monitoring system. Therefore, staff finds that TS LCO 3.1.1 and the associated Surveillance Requirement 3.1.2 indicate that at least one of the available surveillance techniques does not rely on utility systems to ascertain compliance with TS section 3.1, "SFSC Integrity."

As described above, the proposed design of the HI-STORE CIS Facility does not require important-to-safety utility systems or redundant systems to maintain the ability to perform safety functions assuming a single failure. Therefore, the requirements of 10 CFR 72.122(k)(1) regarding utility service redundancy and ability to meet emergency conditions are not applicable. The proposed design of the HI-STORE CIS Facility does not require utility systems during spent fuel storage. Therefore, the emergency utility services required by 10 CFR 72.122(k)(2) are not applicable. The proposed design does not include systems and subsystems that require continuous electrical power to permit continued functioning. Since the design of the HI-STORE CIS Facility does not require emergency power, 10 CFR 72.122(k)(3) also is not applicable.

3.3.5 Control Room and Control Area

In the SAR section 3.5, “Control Room and Control Area,” the applicant stated that the spent fuel storage system used at the HI-STORE CIS Facility is a passive system and, therefore, no control room is required to ensure safe operation. The staff notes that, because the spent fuel storage system is passive, operator actions are not needed to safely operate the facility. However, as stated in SAR section 3.4.1, instrumentation may be used to monitor VVM temperatures and data recorders and alarms are present in the Security Building, but the temperature monitors are not required for safe operation if there is visual surveillance of storage systems.

Based on its review, the staff finds the applicant’s description of the control room and control area acceptable because the 10 CFR 72.122 requirement is not applicable to the HI-STORE CIS Facility as the storage system is passive and requires no control room to ensure safe operation.

3.3.6 Analytical Sampling

As discussed in SAR section 3.6, “Analytical Sampling,” no analytical sampling is required for the safe operation of the HI-STORE CIS Facility or to ensure that operations are within prescribed limits.

Before opening a cask in the CTB, the applicant will sample the gas inside the transportation cask. The gas sample will be analyzed to determine whether the gas can be released to the atmosphere or whether the gas must be filtered and the proper radiological protection needed when removing the transportation cask closure.

In accordance with the operating procedures for the HI-STORE system described in SAR section 11.4.2, “Equipment, Instrumentation, and Facilities,” occupational workers are required to wear radiation monitors. This requirement in effect meets the regulatory requirements of 10 CFR 72.126(a).

Because there are no or very limited effluent releases that require an effluent monitoring system, the staff determined that the HI-STORE system is in compliance with the effluent monitoring requirements of 10 CFR 72.126(c).

3.4 Evaluation Findings

Based on information in the application, the staff concludes the following:

- The SAR includes acceptable descriptions and discussions of the projected operating characteristics and safety considerations, in compliance with 10 CFR 72.24(b).

- The SAR provides information that describes how the facility will be in compliance with the applicable regulations of 10 CFR 72.40(a)(13), which provides reasonable assurance that the activities to be authorized by the license can be conducted without endangering public health and safety.

- The operating procedures and schedule for operations for the CIS facility acceptably provide for control during storage operations to be accomplished from the security/monitoring/surveillance office facility and for control during loading, transfer, and unloading operations to be performed from temporary control facilities, for which the design includes acceptable provisions. This is considered to comply with 10 CFR 72.122(j) in an acceptable manner.

- The descriptions of the proposed CIS facility functions and operating systems with regard to the retrieval of stored radioactive material from storage in normal, off-normal, and accident conditions are acceptable and comply with 10 CFR 72.122(l).

- An acceptable capability to test and monitor important-to-safety components is provided in the design and procedures for the CIS facility, in compliance with 10 CFR 72.128(a)(1).

3.5 References

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NRC, NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," March 2000. ML003686776.

4 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA

In chapter 4, “Design Criteria for the HI-STORE CIS Structures, Systems and Components,” of Revision 0T of its safety analysis report (SAR), dated January 20, 2023, Holtec International (the applicant) described the principal design criteria used in the design of the proposed HI-STORE Consolidated Interim Storage (CIS) Facility.

4.1 Scope of Review

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the application related to the systems, structures, and components (SSCs) and design criteria in the SAR to ensure that the applicant accurately defined the limiting characteristics of the spent nuclear fuel (SNF) to be stored, the classification of SSCs according to their importance to safety, and the design criteria and design bases for the SSCs during normal and off-normal operations and accident conditions, including external conditions and natural phenomena events. The staff also reviewed applicable sections in SAR chapters 1, 2, 4, 5, and 7 and the proposed technical specifications (TS) that the applicant submitted with the application. For decommissioning considerations, the staff reviewed SAR section 13.6, “Decommissioning Plan,” as well as Holtec Report No. HI-2177558, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Decommissioning Plan,” Revision 0, dated February 23, 2018, and Holtec Report No. HI-2177565, “Holtec International & Eddy Lea Energy Alliance (ELEA) CIS Facility—Decommissioning Cost Estimate and Funding Plan,” Revision 1, dated November 16, 2022. The staff used NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” dated March 2000, to guide its evaluation.

4.2 Regulatory Requirements

The regulatory requirements relevant to the SSCs and the design criteria of the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.2, “Scope”
- 10 CFR 72.3, “Definitions”
- 10 CFR 72.6, “License required; types of licenses”
- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.30, “Financial assurance and recordkeeping for decommissioning”
- 10 CFR 72.103, “Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”

- 10 CFR 72.120, “General considerations”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.124, “Criteria for nuclear criticality safety”
- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”
- 10 CFR 72.130, “Criteria for decommissioning”
- 10 CFR 72.144, “Quality assurance program”
- 10 CFR 72.182, “Design for physical protection”

4.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed and evaluated the applicant’s description of the SSCs discussed in SAR Revision 0T as noted in section 4.1 of this safety evaluation report (SER), documents cited in or attached to the SAR, the applicant’s responses to the staff’s requests for supplemental and additional information, and other relevant literature.

The staff reviewed and evaluated the following topics in the application: materials to be stored, classification of SSCs, design criteria for SSCs important to safety (ITS), and design criteria for other SSCs not important to safety (NITS).

4.3.1 Materials to Be Stored

In SAR section 4.1, “Materials to Be Stored,” the applicant described the materials to be stored at the facility, which are limited to SNF that arrives onsite in sealed canisters that have been transported exclusively using the HI-STAR 190 transportation package with the total quantity of SNF and associated non-fuel hardware (pressurized-water reactor fuel only) specified in proposed TS 2.1, “Approved Contents, Fuel Specifications and Loading Conditions.” Additionally, proposed TS 4.2.2, “Storage Capacity,” and proposed license provisions 6, 7, and 8 specify the type of nuclear material, its chemical and physical form, and the maximum amount to be possessed at the HI-STORE CIS Facility.

Because SAR section 4.1, the proposed TS, and the proposed license provisions specify explicit characteristics and limits regarding the radioactive materials that can be received and stored at the facility, the staff found that the description of the material to be stored meets the requirements of 10 CFR 72.2(a)(1).

In SAR section 4.2, “Classification of Structures, Components, and Systems,” the applicant identified the SSCs classified as ITS, the NITS items, and the necessary quality assurance classifications. The applicant listed the ITS and NITS SSCs in SAR table 4.2.1, “ITS Classification of SSCs that Comprise the HI-STORE CIS Facility.” The applicant stated that each SSC classified as ITS is given a level of importance depending on the severity of consequence in the event of its failure or malfunction, consistent with NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” issued February 1996. SAR table 4.2.1 also identifies whether the safety classification of each component was defined in its native docket (i.e., the HI-STORM UMAX canister storage system certificate of compliance (CoC)) or by site-specific considerations for those SSCs not included in the HI-STORM UMAX system design.

4.3.2 Classification of Structures, Systems, and Components

4.3.2.1 Classification of Structures, Systems, and Components—Items Important to Safety

In SAR section 4.2, the applicant defined the quality categories for each of the ITS SSCs. The applicant stated that ITS Category A (ITS-A) items include SSCs whose failure or malfunction could directly result in a condition adversely affecting public health and safety. SAR table 4.2.1 lists the following components classified as ITS-A:

- multipurpose canisters (MPCs)
- HI-TRAC concrete-shielded (CS) transfer cask, transfer cask lift yoke, and transfer cask lift link
- cask transfer building (CTB) crane—main girder and all structural items in the direct load path
- HI-STAR 190 transportation package, transport cask horizontal lift beam, and transport cask lift yoke
- vertical cask transporter (VCT)—overhead beam and lifting towers
- MPC lift attachment
- slings
- MPC lifting device extension

ITS Category B (ITS-B) applies to SSCs whose failure could indirectly (i.e., in conjunction with the failure of another component) result in a condition adversely affecting public health and safety. Note 3 in SAR table 4.2.1 lists the following components classified as ITS-B:

- VCT cask restraint strap
- VCT MPC downloader system
- VCT load drop protection system

ITS Category C (ITS-C) items include SSCs whose failure or malfunction would not significantly reduce the effectiveness of the ITS SSCs and would not be likely to create a situation adversely affecting public health and safety. SAR table 4.2.1 lists the following components classified as ITS-C:

- cavity enclosure container (CEC), CEC closure lid, and CEC divider shell
- support foundation pad

- ISFSI pad
- controlled low strength material (i.e., engineered backfill)
- CTB and CTB slab
- canister transfer facility (CTF)
- transport cask tilt frame
- HI-PORT heavy haul trailer (drop deck)

In addition to the above items, SAR section 3.4.1, “Instrumentation and Control Systems,” states that, if temperature monitoring is used in lieu of visual inspections to verify the operability of the ventilated vertical module (VVM) heat removal system, that equipment shall be designated as ITS. The applicant also stated that CTB crane components not directly in the load path are treated as “augmented quality” under the Holtec quality assurance program. These items include the CTB crane main hoist, auxiliary hoist, and other electrical systems.

4.3.2.2 Classification of Structures, Systems, and Components—Items Not Important to Safety

NITS SSCs are defined as those SSCs whose failure or malfunction would not reduce the effectiveness of the system and would not create a situation adversely affecting public health and safety. The SAR does not include a specific list of NITS SSCs, but rather the information on NITS SSCs is distributed throughout the SAR. NITS SSCs include the following:

- fire protection and suppression systems—as described in SAR section 6.5.3, “SSCs Important to Safety Guidance for Fire Protection Program”

- security equipment—as described in SAR table 18.1.1, “Summary of SSCs Requiring Aging Management & Their Severity Index”

- NITS subcomponents of ITS SSCs—detailed in the bills of materials for each of the drawings in SAR section 1.5, “Drawings” (proprietary)

The staff notes that the SAR discusses various additional items on the site that are not included in the above list. For example, SAR section 1.1, “General Description of the Installation,” states that the facility includes an equipment building, a rail spur, and an administration building. Also, SAR section 1.2.7, “Ancillary Equipment Used at HI-STORE CIS,” discusses minor ancillary items such as “common rigging, ladders, platforms, equipment stands, service and mobile cranes for handling non-critical loads, etc.”

The applicant did not classify the radiation monitoring equipment as ITS. Although the staff recognizes that this equipment is ITS in terms of ensuring compliance with 10 CFR Part 20, “Standards for Protection against Radiation,” Subpart C, “Occupational Dose Limits,” the staff found that radiation monitoring equipment is different from other passive SSCs in that it will be tested and replaced on a regular basis in accordance with the site’s radiation protection program. For this reason, the staff found this classification to be acceptable.

4.3.2.3 Classification of Structures, Systems, and Components—Conclusion

The staff reviewed the classification of the SSCs at the proposed HI-STORE CIS Facility. The staff determined that the applicant’s classification of ITS SSCs is acceptable because the applicant generally followed, as applicable, the recommendations for storage systems in

NUREG/CR-6407, table 6. For some SSCs, the applicant conservatively applied a higher safety classification than that recommended in NUREG/CR-6407. The applicant's classification is also consistent with the safety classification of the HI-STORM UMAX and HI-STORM FW [flood and wind] storage systems incorporated by reference. The staff notes that NUREG/CR-6407 does not specifically address all the components of the HI-STORM UMAX storage system; however, the staff finds the applicant's classification methodology to be consistent with the guideline's recommendations. For those SSCs specific to the HI-STORE CIS Facility (i.e., CTB, concrete pads, and other items not associated with the storage modules), the staff concluded that the applicant had a reasonable basis for their classification as either ITS-A, ITS-B, ITS-C, or NITS.

For the reasons stated above, the applicant's SSC classification satisfies the requirements of 10 CFR 72.3, 10 CFR 72.24(n), and 10 CFR 72.144(a) and (c) for the identification of SNF handling systems to be covered by the quality assurance program.

4.3.3 Design Criteria for Structures, Systems, and Components Important to Safety

In SAR sections 4.3.1 through 4.3.5, the applicant discussed the principal design criteria for the following ITS SSCs: MPCs, VVM components and ISFSI structures, the HI-TRAC CS, the HI-STAR 190 transportation packaging system, and the CTF. For each of these SSCs, these SAR sections provide design criteria information for the technical disciplines of structural, thermal, shielding, confinement, and criticality control.

4.3.3.1 General Information

In SAR section 4.2, the applicant presented information on the classification of the SSCs in accordance with their ITS significance categories. The staff gives its evaluation of the classification of the SSCs for the HI-STORE CIS Facility in SER section 4.3.2.

In SAR section 4.3, "Design Criteria for SSCs Important to Safety," the applicant presented information on the design acceptance criteria and design bases for the ITS SSCs of the HI-STORE CIS Facility. The information addressed all operational modes, including normal and off-normal operations, short-term operations, man-made accident conditions, and extreme natural phenomena events to address the regulatory requirements of 10 CFR 72.98, "Identifying regions around an ISFSI or MRS site." SER section 4.3.3 presents the staff's evaluation of the design acceptance criteria.

SER section 4.3.4 includes the staff's review of the design acceptance criteria for all non-ITS SSCs.

4.3.3.2 Structural Design Criteria

This section presents the staff evaluation of the acceptance criteria used in the structural design of the ITS SSCs for the HI-STORE CIS Facility as delineated in SAR table 4.2.1. The HI-STORE CIS Facility uses storage systems, transportation casks, and SNF canisters that the NRC has certified under the HI-STORM UMAX canister storage system, the HI-STAR 190 transportation package, and the HI-STORM FW storage system, respectively. For SSC designs incorporated by reference, the staff in SER chapter 5 ensures that the site-specific design inputs are enveloped by the corresponding values of the certified design, and any site-specific operational load is evaluated and compared to the accepted safety margins. The staff evaluates the parametric values of the site-specific design input in SER chapter 2.

For SSCs that are designed specifically for the HI-STORE CIS Facility, the SAR adopted acceptance criteria from codes and standards used in industry for the associated level of safety performance or adopted in regulatory guidance documents.

Based on the staff findings of the review of the criteria in SAR chapter 4, the staff concludes that the proposed acceptance criteria are acceptable for ensuring compliance with the regulatory requirements of 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” and 10 CFR 72.122.

4.3.3.2.1 Multi-Purpose Canisters

The MPCs to be stored in the HI-STORE CIS Facility comprise a fuel basket enclosed in a welded canister. SAR table 1.0.4, “Canisters Allowed for Storage in HI-STORM UMAX at HI-STORE,” specifies that only MPC-89 and MPC-37 canisters are used at the HI-STORE CIS Facility to store boiling-water reactor and pressurized-water reactor SNF, respectively. SAR section 4.1 states that these canisters are qualified in the HI-STORM FW system final safety analysis report (FSAR), Revision 4, dated June 24, 2015, and further limitations on its design are imposed by SAR tables 4.1.1 through 4.1.4 for use at the HI-STORE CIS Facility. SAR table 4.0.1, “HI-STORM UMAX FSAR Material Incorporated in this FSAR by Reference,” identifies the design criteria and SSCs that the applicant incorporated by reference for use at the HI-STORE CIS Facility.

In SAR table 4.0.1, the applicant incorporated by reference the design criteria of the HI-STORM UMAX storage system that appear in section 4.3.2 of the HI-STORM UMAX storage system FSAR, Revision 3, dated June 29, 2016. The staff finds the incorporation by reference of the design criteria for the MPCs acceptable for use at the HI-STORE CIS Facility because the applicant is using the MPC for the same type of SNF as it was certified for in the HI-STORM FW storage system FSAR. The staff evaluated the function of the canister and its allowable contents and finds no notable change in the functional demand of the MPC design features specific to the HI-STORE CIS Facility. The staff notes the additional heat load and internal pressure requirements in SAR tables 4.1.1 through 4.1.4 limit the structural demand on the MPCs. The staff finds that the allowable demands for the certified MPC design envelope the demands at the HI-STORE CIS Facility except for some site-specific natural hazards that differ from those used for the certified design, and which the staff evaluated as part of this review, as noted below. Because the MPC certified by the HI-STORM FW FSAR meets all the demands of the HI-STORE CIS Facility except the site-specific design inputs, the staff concluded that enveloping the site-specific inputs results in appropriate acceptance criteria for using the same MPCs at the HI-STORE CIS Facility. SER section 5.3.1.1 discusses the staff’s further evaluation of the site-specific parametric values of the design criteria in reviewing the MPC structural design to ensure that the values used in the certified design envelope those for the HI-STORE CIS Facility.

4.3.3.2.2 Ventilated Vertical Module Components and Independent Spent Fuel Storage Installation Structures

The VVMs and the ISFSI slab as qualified in the HI-STORM UMAX FSAR are used for storage at the HI-STORE CIS Facility. The applicant incorporated by reference the acceptance criteria for this SSC in SAR table 4.0.1.

The staff finds the design acceptance criteria incorporated by reference acceptable for the structural design of the VVMs and ISFSI slab for use in the HI-STORE CIS Facility. The staff reviewed the construction material for the ISFSI, the surrounding controlled low-strength material, and the material of the subgrade as compared in SAR table 4.3.3, “Applicable Earthquake and Long Term Settlement data for the Certified HI-STORM UMAX System and the HI-STORE CIS Facility,” with the HI-STORM UMAX design. The staff finds that the structural design criteria for the HI-STORE CIS Facility require the materials to be identical or better for the HI-STORE CIS Facility. The staff finds some small change in the operational demands on the design incorporated by reference, such as the load associated with placing a loaded HI-TRAC CS above the VVM for transfer of an MPC (known as a stack-up), the load transmitted by the loaded VCT, and some difference in site-specific seismic hazards input. Because the VVM and the ISFSI as certified by the HI-STORM FW FSAR meet all the demands of the HI-STORE CIS Facility except the site-specific design inputs, the staff concluded that enveloping the site-specific inputs results as appropriate acceptance criteria for using the same VVMs and ISFSI at the HI-STORE CIS Facility. SER sections 5.3.3 and 5.3.4 discuss the staff’s further evaluation of the site-specific parametric values of the design acceptance criteria in reviewing the VVM and ISFSI structural design to ensure that the values used in the certified design envelope those for the HI-STORE CIS Facility.

4.3.3.2.3 HI-TRAC CS Transfer Cask

As described in SAR section 1.2.4, "HI-TRAC CS," the HI-TRAC CS is a concrete-shielded transfer cask, a variation of the HI-TRAC VW [variable weight] transfer cask that is licensed for use with the HI-STORM UMAX system. SAR section 4.3.3 states that the structural steel components of the HI-TRAC CS are designed to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsection NF, Class 3, at Level A stress limits for all operating modes. The embedded trunnions for lifting and handling of the transfer cask are designed in accordance with the guidance in NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980.

The staff reviewed the design criteria of the cask against the functional design requirements and finds the use of ASME BPVC, Section III, appropriate for the construction of this nuclear component. The ASME BPVC has a successful performance history in the construction of nuclear components and is cited in regulatory guidance documents for a similar application. The cask is metallic and fabricated with steel plates. The staff determined that it is appropriate to apply to this design the construction rules of ASME BPVC, Section III, Division 1, for plates.

The transfer cask does not provide a containment or pressure-retention function; hence, an ASME BPVC Class 3 design is acceptable. The staff accepts the use of subsection NF for design of a component using plates. As the operational conditions are under normal conditions, use of Level A stress limits is acceptable. For the lifting trunnion, the applicant cited the acceptance criteria included in NUREG-0162 in lieu of providing specific requirements. This NUREG refers to American National Standards Institute (ANSI) N14.6, “Radioactive Materials—Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More,” June 1993 edition, which refers to the ASME BPVC. The staff accepted the referenced ASME sections in the ANSI standard for the trunnion design stress limits, subsections NF-5340 and NF-5350, as they are consistent with the rest of the allowable limits for the stress analysis of the cask.

The staff considered that the HI-TRAC CS will always be handled using single-failure-proof devices and will only be used for short-term operations at the facility. The staff notes that the normal conditions of transport and hypothetical accident conditions of transport as specified in 10 CFR 71.71, “Normal conditions of transport,” and 10 CFR 71.73, “Hypothetical accident conditions,” do not apply in this type of application.

Based on the above findings, the staff concludes that for the limited functions of shielding and protection against natural hazards, the design acceptance criteria for the HI-TRAC CS meet the design requirements of 10 CFR Part 72 to support the transfer needs for short-term operations at the HI-STORE CIS Facility. The staff further evaluates the structural design of the HI-TRAC CS in SER section 5.3.2.2 using these design acceptance criteria.

4.3.3.2.4 HI-STAR 190 Transportation Package

SAR table 1.0.5, “Transport Casks Allowed for Receipt of Canisters at HI-STORE,” specifies that the HI-STORM CIS Facility will receive SNF only in HI-STAR 190 transport casks. SAR section 4.3.4, “HI-STAR 190,” further specifies that the HI-STAR 190 transport cask will be used to deliver the loaded canisters to the CTB. The NRC certified the HI-STAR 190 transportation system described in Holtec Report No. HI-2146214, “Safety Analysis Report on the HI-STAR 190 Package,” as a transportation package under 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.” In the SAR for the HI-STORE CIS Facility, the applicant referenced Revision 3, dated November 2, 2018, of the HI-STAR 190 transportation system SAR. In the CTB, the transfer cask is exposed to no additional demands other than those from canister transfer operations. The applicant specified the allowable stresses and stress intensities in SAR table 4.5.4 through table 4.5.9. These values ensure that the strength demand from the transfer operations is enveloped by those of the transportation demands in the incorporated design. Hence, the staff concluded that the generic acceptance criteria used for certification are acceptable. The staff compared the HI-STORE CIS Facility-specific usage in the structural evaluation in SER section 5.3.2.1 with the usage in the certified design to ensure that the design incorporated by reference is acceptable for use at the HI-STORE CIS Facility.

4.3.3.2.5 Canister Transfer Facility

The applicant described the CTF and its steel structure in SAR sections 4.3.5, “Canister Transfer Facility (CTF),” and 5.4.7, “Canister Transfer Facility Steel Structure.” The HI-STORE CTF is an underground cylindrical structure for transfer of SNF canisters from the HI-STAR 190 transportation cask to the HI-TRAC CS. The structural steel components of the CTF are designed to meet the stress limits of the 2010 Edition of ASME BPVC, Section III, Division 1, Subsection NF, Class 3, for normal, off-normal, and accident conditions, as applicable. The CTF reinforced concrete structures are designed to the American Concrete Institute (ACI) strength requirements of ACI 318-05, “Building Code Requirements for Structural Concrete and Commentary,” issued 2005. The applicant used the American Society of Civil Engineers (ASCE) guidance in chapter 8 of ASCE 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” issued 2017, to determine the seismic loads on the buried CTF structure.

The staff concludes that the code and design allowable limits used as design acceptance criteria for the concrete and steel portions of the CTF are acceptable. The CTF containing the MPC before transfer to the HI-TRAC CS performs the functions of a storage cask with the steel structure designed to the guidance prescribed for casks in NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” issued April 2020. The concrete structure meets the design performance of a standard building underground structure. The combined performances of the designed CTF using these codes are acceptable for an ITS-C structure. The staff further evaluates the structural design of the CTF to the design acceptance criteria in SER section 5.3.4.3.

4.3.3.2.6 Acceptance Criteria for Stress Limits Using ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF

SAR chapter 4 discusses how the HI-STORE CS and the CTF both perform the protective functions of a cask for short-term operations at the HI-STORE CIS Facility. Therefore, in SAR section 4.4, “Acceptance Criteria for Cask Components,” the applicant applied the stress limits of ASME BPVC, Section III, Division 1, Subsection NF, to the cask component design. SAR section 4.4 also discusses the allowable stresses under the code for the different load conditions, including thermal conditions. In SAR table 4.4.1, “Permissible Temperature Limits for HI-TRAC CS and CTF Materials,” the applicant specified the temperature limits for the HI-TRAC CS and the CTF. In SAR table 4.4.2, “Stress and Acceptance Limits for Different Loading Conditions for the Primary Load Bearing Structures in the Steel Weldments of Casks,” the applicant presented the stress and acceptance limits for different loading conditions for the primary load-bearing structures in the steel weldments of casks. In SAR appendix 4.A, “Stress Limits for ASME Section III, Subsection NF Linear Structures and Plate & Shell Type Structures,” the applicant cited the method used for computing the stress limit criteria pursuant to ASME BPVC, Section III, Division 1, Subsection NF, for performing stress analysis of the miscellaneous ancillaries at the HI-STORE CIS Facility.

The staff finds adherence to these design code criteria acceptable as the transportation cask design criteria meet structural demands that are much higher than those for the short-term operations of the HI-TRAC CS and the CTF, which functions as a cask, in the short-term operations at the HI-STORE CIS Facility. The staff used the allowable stress acceptance criteria in its structural evaluation of the HI-TRAC CS and CTF in SER sections 5.3.2.2 and 5.3.4.3, respectively.

4.3.3.2.7 Cask Transfer Building

SAR section 4.6, “Design Criteria for the Cask Transfer Building (CTB),” identifies the design basis for the CTB as “Other SSC subject to NRC approval.” SAR table 4.2.1 associates the CTB with an ITS-C classification. The applicant described the CTB as a concrete building with a load-bearing slab on grade. The concrete walls with columns support the overhead crane and the roof. The roof is a concrete slab on metal decking supported by a steel framing resting on the walls and columns. In addition, the CTB houses the CTF, which is integrated into the floor slab of the CTB. SAR section 4.6 identifies the following codes being used as acceptance criteria for the CTB structural design:

- American Institute for Steel Construction (AISC) 360-16, "Specification for Structural Steel Buildings," issued 2016, for the steel framework and connections design
- ASCE 7-10, "Minimum Design Loads for Building and Other Structures," issued 2013, for the minimum design loads to be considered for the CTB
- International Code Council, "International Building Code," issued 2015, for general design requirements
- ACI 318-05, "Building Code requirements for Structural Concrete and Commentary," issued 2005, for reinforced concrete slab, walls, and roof
- applicable portions of New Mexico's State and local building codes

The applicant stated that it used load combinations based on information provided in table 4-3 of NUREG-2215 and used section 9.2 of ACI 318-05 to demonstrate the structural integrity of the CTB.

The staff concludes that these design criteria are acceptable for the CTB design. The CTB does not provide any confinement or containment function and performs the function of a standard building providing protection from external hazards to the SSCs within it. Thus, the use of standard building codes supports the expected function of the building. The use of standard industry codes and standards provides for a design strength with adequate margin for a building with such classification. However, the staff notes that for a nuclear facility, the external hazards include protection against tornado missile impacts as characterized by Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, issued March 2007. The design criteria incorporates both localized and global loads resulting from events, as required by 10 CFR 72.122. SER section 5.3.4.4 discusses the staff's structural evaluation of the CTB, and SER section 15.3.2.3 discusses the staff's evaluation of the CTB for a missile impact event.

4.3.3.2.8 Load-Handling Devices of the HI-STORE CIS Facility

The two devices used for lifting heavy loads at the HI-STORE CIS Facility are the CTB overhead crane and the VCT. The following sections review the design criteria used for each of them.

4.3.3.2.8.1 Cask Transfer Building Overhead Crane

The applicant described the CTB overhead crane in SAR section 4.5.2, "Cask Transfer Building (CTB) Crane," as a top-running bridge crane with a trolley hoist. It is the principal load-handling device used to lift, up-end, down-end, and translocate the casks and other heavy loads inside the CTB. The CTB overhead crane is vendor supplied, designed as a single-failure-proof load-handling device, and built in accordance with the provisions of ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," issued 2015. SAR table 4.5.1, "Design Basis Loadings on the Cask Crane inside the CTB," specifies the design-basis dead and seismic loads, and SAR table 4.5.2, “Design Parameters for the CTB Crane,” specifies the crane design parameters.

The staff concludes that the specification for this vendor-supplied component is adequate to meet the design requirements for a single-failure-proof crane for the CTB. The design guidelines in ASME NOG-1 are comprehensive and supported by the guidance in NUREG-0612 for single-failure-proof design that allows the use of redundant devices or double the design factor of safety.

The staff further reviews the site-specific seismic design inputs provided to the vendor in SER section 5.3.4.4.4, under the structural evaluation of the CTB. The vendor-provided design reactions at the corbel level of the CTB are an inspection item to confirm that the design loads at the corbels envelope the vendor-supplied reactions at the crane support.

4.3.3.2.8.2 Vertical Cask Transporter

SAR section 4.5.3, “Vertical Cask Transporter,” describes the VCT as the principal load-handling system used at the HI-STORE CIS Facility. The following essential structural design requirements for the VCT appear in SAR section 4.5.3.3, “Structural”:

- All materials used in the design of the overhead beam and lifting towers are ASTM International or ASME approved.

- Prevention of a cask or canister drop is afforded by design conformance with NUREG-0612 and ANSI N14.6, where applicable, combined with enhanced safety margins and the use of redundant drop-protection features, such as hydraulic check valves, and a fail-safe electrical control system.

- The VCT vehicle frame is designed in accordance with applicable industry standards such as ASME BPVC, Section III, Division 1, Subsection NF, for Class 3 linear-type supports or equivalent, or AISC standards.

- The overhead beam, lifting attachments, and MPC downloader pulley/pins and other attachments are designed in accordance the applicable guidance of NUREG-0612, section 5.1.6, which includes ANSI N14.6 for those items meeting the definition of special lifting devices. For the special lifting devices, the safety factor shall be based on the lower of 1/6th the yield strength or 1/10th the ultimate strength. The overhead beam, MPC downloader pulley/pins, and other lifting devices shall be single-failure proof.

- Jacks are designed in accordance with ASME BPVC, Section III, Division 1, Subsection NF, for Class 3 linear-type supports, and ASME B30.1, “Jacks, Industrial Rollers, Air Casters, and Hydraulic Gantries—Safety Standard for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks, and Slings,” issued 2009, with design safety factors consistent with the guidance of NUREG-0612, section 5.1.6(1)(a), for the specific load lifted. Multistage jacks may have several rated capacities based on the extension stage. The jacks’ rated capacity shall be coupled with the load based on the jack configuration for the lift of the load.

- SAR table 4.5.3, “Design Basis Conditions and Loadings on the Vertical Cask Transporter,” lists the applicable design-basis dead weight and seismic loadings on the VCT. The VCT is shown not to tip over under any specified service condition. The vehicle's lateral and transverse center of gravity shall be lower than the HI-TRAC's lateral and transverse center of gravity while transporting a loaded HI-TRAC CS. Stability checks shall assume a 7 percent transverse grade in all modes for conservatism. A national consensus standard such as ASCE/Structural Engineering Institute (SEI) 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," issued 2005, shall be used for stability evaluation. The seismic restraints and their attachment points on the VCT frame shall be designed to meet the Level D stress limits of ASME BPVC, Section III, Division 1, Subsection NF.

The staff concludes that these specifications provide adequate criteria for the ITS function to be performed with the VCT at the HI-STORE CIS Facility. The staff's review identifies that the materials and the structural design of the lift beam and the supporting towers are appropriately constrained by the strength design requirements of ASME or AISC. The single-failure-proof function of the crane is supported by the adherence to NUREG-0612 redundancy in hydraulic and electrical systems. The tip-over analysis of the VCT allows adequate operating tolerance for the VCT to perform its intended function at the ISFSI pad.

4.3.3.3 Thermal

SAR chapter 4 discusses the thermal-related design criteria and acceptance criteria associated with the HI-STORE CIS Facility. For example, SAR table 4.0.1, table 4.1.1, table 4.1.2, table 4.1.3, table 4.1.4, figure 4.1.1, and figure 4.1.2 list thermal-related design criteria (e.g., content description and temperature limits), including those that were incorporated by reference from the HI-STORM UMAX FSAR. SAR table 4.2.1 provides the ITS designations of HI-STORE CIS SSCs; for example, table 4.2.1 lists the MPC content (SNF canisters) and HI-TRAC CS as ITS-A. In addition, SAR section 3.4.1 states that the continuous VVM air-temperature monitoring system used to ensure design-basis cooling of the canister in storage is designated as ITS if it is the sole means of surveillance (i.e., in lieu of direct visual inspections).

SAR section 4.3 discusses the thermal-related design and acceptance criteria for the ITS SSCs described in SAR section 4.2 and section 4.4. Design temperature limits appear in HI-STORM UMAX FSAR table 2.3.7, which the applicant incorporated by reference in SAR table 4.0.1. Thermal-significant loadings for the HI-TRAC CS cask and the HI-STAR 190 transportation package appear in SAR table 4.3.5, "Thermally Significant Loadings (TSL) for HI-TRAC CS," SAR table 4.3.6, "Governing Structural and Thermal Loadings for HI-STAR 190 during Short Term Operations," and SAR table 4.4.4, "HI-STAR 190 Materials Temperature Limits." SAR table 4.4.1 provides allowable temperatures of structural steel and shielding components of the HI-TRAC CS and CTF. SAR table 4.3.2, "Environmental Data for the Licensing Basis in the HI-STORM UMAX Docket and the HI-STORE Site for Different Service Conditions," lists the environmental data for normal, off-normal, and accident conditions. SAR chapter 6, "Thermal Evaluation," and the referenced calculation packages within that chapter contain additional details and discussion of the thermal loads, allowable temperatures, and site-specific environmental ambient conditions (e.g., temperature, insolation) as part of the descriptions of thermal analyses associated with normal, off-normal, natural phenomena, and accident conditions.

The staff presents its evaluation and findings regarding application of the above-mentioned design criteria to SSCs at the HI-STORE CIS Facility in SER chapter 6. The staff finds that the HI-STORE CIS Facility thermal design criteria are consistent with applicable guidance from section 4.4.3.3 of NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000, which states that the applicant must identify thermal design criteria and bases, and that these criteria and bases must recognize the site temperature range and the specific materials used in the ISFSI. As discussed above, the applicant has identified and characterized those criteria and, therefore, the staff finds the criteria and bases acceptable because the information meets the regulatory requirements of 10 CFR 72.122(a), (b)(1), (b)(2), (b)(3), (c), (h), and (i) and 10 CFR 72.128(a)(4) with respect to thermal considerations.

4.3.3.4 Shielding

The shielding design of the HI-STORE CIS Facility considers normal, off-normal, and accident conditions of the ISFSI operations which include: (1) transient operations (also known as short-term operations) and (2) storage operations after canisters are installed in the VVMs. Transient operations include receipt of canisters, transfer of canisters to the HI-TRAC CS transfer canister, movement of canisters to the VVMs, installation of canisters into the VVMs, and the corresponding reverse operations undertaken to remove canisters from storage and prepare them for shipment offsite. The storage operations of the ISFSI include storage of the MPCs in the VVMs under normal, off-normal conditions and potential accident conditions.

The shielding design for SSCs involved in transient operations at the ISFSI relies on the welded canister containment walls, the shielding design of the HI-STAR 190, the CTB, the HI-TRAC CS, the VVM wall, the closure lid of the VVMs, the backfill materials surrounding the VVMs, and the distance from the CTB and VVMs to the controlled area boundary. The applicant's ITS classification of these SSCs (from SAR table 4.2.1) is consistent with the shielding design criteria for both transient and storage operations.

The applicant described the shielding components in detail in SAR chapter 7, "Shielding Evaluation." The applicant performed shielding analyses for the different operations and provided the expected dose at the controlled area boundary and dose rates around the HI-STAR 190 and HI-TRAC CS and inside the CTB. Chapter 7 of this SER documents the details of the staff's review of the shielding design for the HI-STORE CIS Facility.

Based on its review, the staff finds that the applicant has correctly classified the SSCs important to shielding and used appropriate design criteria for the shielding design of the ISFSI. On these bases, the staff determined that the application meets the regulatory requirements of 10 CFR 72.24 and 10 CFR 72.122.

4.3.3.5 Confinement

In SAR section 4.3.1.4, "Confinement," the applicant described how the MPC-37 and MPC-89 canisters proposed for storage at the HI-STORE CIS Facility provide confinement for normal, off-normal, and accident storage conditions when inside the HI-STORM UMAX or HI-TRAC transfer cask. SAR section 4.3.1.4 notes that the confinement criteria are incorporated by reference from section 2.0.6 of the HI-STORM UMAX FSAR, Revision 3. The applicant stated in SAR section 9.2.1, "Storage Systems," that all normal, off-normal, and accident conditions relevant to confinement integrity for which the canister is certified in the HI-STORM UMAX storage system, Docket No. 72-1040, are equal to or less severe than the corresponding conditions at the HI-STORE CIS Facility. The applicant stated in SAR table 4.2.1 that the canisters (i.e., the MPC-37 and MPC-89 in the HI-STORM UMAX system) provide leaktight confinement, and the canister confinement boundary components are ITS-A.

The staff's evaluation of and findings for the HI-STORM UMAX canister storage system, with the MPC-37 and MPC-89 canisters proposed for storage at the HI-STORE CIS Facility, are in sections 9.3 and 9.4 of this SER, respectively. SER sections 9.3.2, 9.3.3, and 9.3.4 contain the staff's evaluation and findings on radionuclide confinement analysis, confinement monitoring, and protection of stored materials from degradation, respectively. The staff finds that the confinement design criteria for the HI-STORM UMAX canister storage system with the MPC-37 and MPC-89 canisters proposed for storage at the HI-STORE CIS Facility are consistent with applicable guidance from NUREG-1567, section 4.4.3.4, which states that the applicant must identify confinement design criteria and bases, and with NUREG-1567, section 4.5.3.4, which states that the reviewer should confirm that the method of sealing is defined and meets regulations for redundant seals and that the maximum leak rates are specified and do not result in exceeding dose requirements.

The applicant has identified and characterized those design criteria and bases related to confinement. For the reasons discussed above and in the evaluation in SER chapter 9, the staff finds the criteria and bases are acceptable because the information meets the regulatory requirements with respect to the design bases and criteria for shielding, confinement, radiation protection, and as low as is reasonably achievable (ALARA) considerations for 10 CFR 72.24(c)(1), (c)(2), (c)(4), and (n); 10 CFR 72.104(a), (b), and (c); 10 CFR 72.106(a) and (b); 10 CFR 72.122(a), (b), (c), (d), (e), (f), (g), (h), and (i); 10 CFR 72.126(c) and (d); and 10 CFR 72.128(a) and (b) with respect to confinement.

4.3.3.6 Criticality

As discussed in SAR section 4.3.1.5, "Criticality Control," the HI-STORE CIS Facility uses a combination of limiting the quantity of fissile materials to be stored per canister, neutron poison plates, and favorable geometric arrangement of the fissile materials to control criticality. As discussed in section 17.3.9 of this SER, the staff determined that the applicant has demonstrated that the neutron poison plates used in the canisters for criticality control will keep their effectiveness for the duration of the licensed storage time. On these bases, the staff found that this design feature is consistent with the regulatory requirements of 10 CFR 72.124(b), which states the following:

> *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 SNF storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

Therefore, the NRC staff concludes that the regulatory requirement for the continued efficacy of the neutron poison material is met as prescribed in 10 CFR 72.124(b).

4.3.3.7 Decommissioning Considerations

As part of its license application, on March 30, 2017, the applicant submitted a proprietary preliminary decommissioning plan (DP) (public version submitted on February 23, 2018). The staff reviewed the preliminary DP to determine whether the applicant's provisions for decommissioning the facility give reasonable assurance that decontamination and decommissioning of the facility at the end of its useful life will provide adequate protection of public health and safety.

In section 6.3, "Decontamination Tasks," of the preliminary DP, the applicant stated the following:

> Once all of the canisters stored at HI-STORE CIS Facility have been shipped off-site and the decommissioning period begins, a Historical Site Assessment will be performed to identify through record review and personnel interviews any incidents that may have caused contamination to an area of the site. Those areas in which a contaminated event may have occurred will be assessed in more detail to determine the extent of any residual contamination. Detailed characterization surveys will be performed to verify that the VVMs and the concrete pads are free of contamination and all other areas of the site determined to be non-impacted will have a confirmation survey conducted. It is anticipated that the VVMs and concrete pads will not be contaminated and will be left in place or removed as determined by HI-STORE CIS Facility. NRC limits for unrestricted release using conventional decontamination techniques will be used which minimize the volume of waste. Any waste generated will be sent to a licensed facility for disposal.

The staff recognizes that preliminary DPs submitted with the license application are prospective in nature and do not have the benefit of knowledge gained over the course of facility operation. Therefore, the staff conducted its review with a limited scope to determine whether the preliminary DP includes elements required by the regulations such as design and operational features that will facilitate decontamination and decommissioning, a decommissioning funding plan, a cost estimate, and a financial assurance mechanism.

The staff summarizes its review of the preliminary DP in SER chapter 13 and its review of the financial aspects of the preliminary DP and cost estimate in SER chapter 18. The staff finds that the SAR and docketed materials relating to the design criteria for decommissioning of the facility comply with 10 CFR 72.130 because the facility is designed for decommissioning and provisions have been made to facilitate decommissioning in the future.

4.3.3.8 Retrieval Capability

In SAR section 1.2.5.1, "Design Features," and section 5.4.1.2, "Design Criteria," the applicant stated that the design features of the HI-STORM UMAX storage system permit retrieval of the storage system contents. In the SER for the initial approval of the HI-STORM UMAX storage system CoC, dated April 2, 2015, the staff found that the HI-STORM UMAX storage system is adequately designed to support retrievability. The staff conducted its HI-STORM UMAX CoC review using 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," which provides the regulatory requirements for dry storage systems that may be used by general licensees. The CoC retrievability requirement included in 10 CFR 72.236(m) states, "To the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy." The design bases of the HI-STORM UMAX system reference NRC Interim Staff Guidance (ISG)-2, "Fuel Retrievability," Revision 0 (nonpublic), dated October 6, 1998, to demonstrate that the storage cask design is compatible with removal of the SNF from the reactor site. ISG-2 defined fuel retrievability as the ability to prepare an MPC for offsite transportation without having to handle individual fuel assemblies. The HI-STORM UMAX SAR, Revision 3, section 3.1.2, states that the structural analysis of the storage system demonstrates that no changes in the geometry of the storage modules occur under normal, off-normal, and accident conditions that would preclude retrieval of the MPC from the VVM cavity.

The retrieval regulation that is applicable to site licenses, 10 CFR 72.122(l), states that "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 staff notes that the applicant's basis for retrieval capability (consistent with ISG-2, Revision 0) is also consistent with one of the recommended approaches to demonstrate retrievability in the latest revision of NRC ISG-2, "Fuel Retrievability in Spent Fuel Storage Applications," Revision 2, dated April 26, 2016, which has been incorporated in the current standard review plan, NUREG-2215. To demonstrate the ability for retrieval, the guidance states that a licensee should demonstrate that it has the ability to perform any of the three options below.

(1) Remove individual or canned SNF assemblies from wet or dry storage.
(2) Remove a canister loaded with SNF assemblies from a storage cask/overpack.
(3) Remove a cask loaded with SNF assemblies from the storage location.

As stated in ISG-2, Revision 2, any of the three options described above is considered adequate to demonstrate retrievability to meet either 10 CFR 72.122(l) (applicable to the HI-STORE CIS Facility license) or 10 CFR 72.236 (applicable to the HI-STORM UMAX CoC). The applicant's demonstration of the capability to retrieve the MPC from the VVM cavity is consistent with option 2 above.

ISG-2, Revision 2, also states that the potential impact of aging-related issues on retrievability should be considered when site licenses are renewed beyond their initial storage term. The staff notes that, while the applicant is requesting only an initial 40-year license for the HI-STORE CIS Facility, the SAR nevertheless includes maintenance and aging management activities that address the ISG guidance for renewal, including those to ensure that the MPCs and VVM cavities are free of degradation. The staff documents its review of those activities in SER section 17.3.16.

Based on the information described above, including the HI-STORE CIS Facility-specific activities to manage long-term aging issues and the prior conclusion that the HI-STORM UMAX storage system met the regulatory requirements for 10 CFR 72.236, the staff finds that the applicant has adequately demonstrated compliance with the regulatory requirements in 10 CFR 72.122(l).

4.3.4 Design Criteria for Other Structures, Systems, and Components Not Important to Safety

The staff evaluated the design criteria for other NITS SSCs to verify that descriptions are sufficient to enable evaluation of facility compliance with the relevant regulatory requirements.

4.3.4.1 Fire Protection and Suppression Systems

In SAR section 6.5.3, the applicant stated that fire protection and suppression systems are classified as NITS due to the nonflammable nature of the materials of construction, other passive design features, and the limited fuel sources at the facility. Revision 5 of the site emergency response plan issued November 2022, states that fire protection systems are designed in accordance with applicable National Fire Protection Agency (NFPA) code requirements. Each operational area has hydrants, sprinklers, or both and is also equipped with portable extinguishers. The emergency response plan also states that NFPA and the National Fire Code will be followed with regard to location, testing, and maintenance of emergency equipment, standard hose connections, and sprinklers. The staff finds the design criteria and bases for the fire protection systems to be acceptable because the applicant is using appropriate NFPA code criteria for the systems’ design, testing, and maintenance.

4.3.4.2 Security Equipment

SAR table 18.1.1, “Summary of SSCs Requiring Aging Management & Their Severity Index,” indicates that security equipment at the facility is NITS. SAR section 1.1 states that security features of the site include the vehicle barrier system, passive and active physical barriers, turnstiles at security building entrances, a vehicle trap, and protected area fencing that is supplemented by intrusion detection and CCTV cameras. Revision 3 of the facility’s physical security plan (nonpublic), dated March 2, 2020, contains additional details on these security measures. The staff documents in SER section 10.3.6 its review to verify that the security plan meets the requirements of 10 CFR 72.180, “Physical protection plan,” to demonstrate adequate physical security of SNF. The staff finds the design criteria and bases for the security equipment to be acceptable because the staff’s review of the physical security plan verified that the applicant has adequately defined the design for physical protection, including tests, inspections, audits, and other means to be used to demonstrate compliance with physical security requirements.

4.3.4.3 Radiation Monitoring Equipment

SAR section 3.4.1 and section 11.2.5, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation,” state that radiation monitoring ensures that the doses continue to meet the regulatory requirements of 10 CFR 72.104, 10 CFR 72.106, and 10 CFR 72.44(c)(1)(ii). The applicant stated that thermoluminescent dosimeters are a backup for monitoring personnel radiation exposure, and they are located at the restricted area fence, the controlled area boundary, and various buildings where personnel normally work. The applicant also stated that local continuous air monitors warn operating personnel in the event of an airborne release, and these monitoring systems are designed with provisions for calibration and operability testing.

Further, SAR section 3.1.5.4, “Instrumentation,” states that operators are equipped with personnel dosimeters whenever they are in the protected (restricted) area. The staff finds the design criteria and bases for the radiation monitoring equipment to be acceptable because they ensure that the operation of the ISFSI system meets the regulatory requirements of 10 CFR Part 20, Subpart C.

4.3.4.4 Design Criteria for Other Structures, Systems, and Components Not Important to Safety—Conclusion

For the reasons stated above, the staff finds that the design criteria for NITS SSCs are acceptable because they meet the requirements in 10 CFR 72.24(a), (b), (c), (d), (e), (f), and (g) and the appropriate requirements as given in Subpart E, “Siting Evaluation Factors,” and Subpart F, “General Design Criteria,” of 10 CFR Part 72.

4.4 Evaluation Findings

Based on the information in the application, the staff concludes the following:

- The SAR and docketed materials adequately identify and characterize the SNF to be stored at the site in conformance with the requirements given in 10 CFR 72.2(a)(1). The description of the SNF is acceptable for demonstrating that the ISFSI design meets the requirements given in 10 CFR 72.120(b).

- The SSCs have been classified according to their function as ITS or NITS and meet the requirements given in 10 CFR 72.3, 10 CFR 72.24(n), and 10 CFR 72.144(a) and (c).

- The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.24(c)(1), (c)(2), (c)(4), and (n); 10 CFR 72.106(a); 10 CFR 72.120(a) and (b); 10 CFR 72.122(a), (b), (c), (d), (e), (f), (g), (h), (i), (j), (k), and (l); and 10 CFR 72.144.

- The SAR and docketed materials relating to the design bases and criteria for structures categorized as ITS meet the requirements given in 10 CFR 72.24(c)(1), (c)(2), (c)(3), and (n); 10 CFR 72.103; 10 CFR 72.120(a) and (b); and 10 CFR 72.122(a), (b)(1), (b)(2), (b)(3), (c), (d), (f), (g), (h), (i), (j), and (k).

- The SAR and docketed materials meet the regulatory requirements for design bases and the criteria for thermal consideration as given in 10 CFR 72.122(a), (b)(1), (b)(2), (b)(3), (c), (d), (f), (g), (h), and (i) and 10 CFR 72.128(a)(4).

- The SAR and docketed materials relating to the design bases and criteria for shielding, confinement, radiation protection, and ALARA considerations meet the regulatory requirements as given in 10 CFR 72.24(c)(1), (c)(2), (c)(4), and (n); 10 CFR 72.104(a), (b), and (c); 10 CFR 72.106(a) and (b); 10 CFR 72.122(a), (b), (c), (d), (e), (f), (g), (h), and (i); 10 CFR 72.126(a), (b), (c), and (d); and 10 CFR 72.128(a) and (b).

- The SAR and docketed materials relating to the design bases and criteria for criticality safety meet the regulatory requirements as given in 10 CFR 72.124(a) and (b).

- The SAR and docketed materials relating to the design criteria for decommissioning of the facility comply with the regulatory requirements given in 10 CFR 72.130.

- The SAR and docketed materials relating to the design bases and criteria for retrieval capability meet the regulatory requirements given in 10 CFR 72.122(a), (b)(3), (c), (f), (h), and (l).

- The SAR and docketed materials relating to the design bases and criteria for other SSCs not ITS but subject to NRC approval meet the general regulatory requirements as given in 10 CFR 72.24(a), (b), (c), (d), (e), (f), and (g) and the appropriate requirements as given in Subpart E and Subpart F of 10 CFR Part 72.

4.5 References

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5 INSTALLATION AND STRUCTURAL EVALUATION

In Chapter 4, “Design Criteria for the HI-STORE CIS Systems, Structures and Components,” of Revision 0T, dated January 20, 2023, of its safety analysis report (SAR), Holtec International (the applicant) provided the design criteria for the systems, structures, and components (SSCs) at the proposed HI-STORE Consolidated Interim Storage (CIS) Facility in Eddy and Lea counties of New Mexico and classification of the SSCs as important to safety (ITS) and not important to safety (NITS) but relied on for site operations. In SAR Chapter 5, “Installation and Structural Evaluation,” the applicant summarized the structural design safety analysis for the SSCs of the HI-STORE CIS Facility. The details of the analysis and design appear in Holtec Report No. HI-2177585, “Structural Calculation Package for the HI-STORE CIS Facility” (proprietary), Revision 5, dated January 20, 2023; Holtec Report No. HI-2210576, “Structural Analysis of the HI-STORE Cask Transfer Building” (proprietary), Revision 1, dated March 24, 2022; and Meloni Calculation No. 722118, “Static, Seismic Analysis and Mechanical Calculations for HI-STAR CRANE” (proprietary), Revision 0, dated March 16, 2022, submitted as part of the application.

5.1 Scope of the Review

In this section, the staff of the U.S. Nuclear Regulatory Commission (NRC) presents the safety evaluation of SAR chapter 5. The staff used the safety classification and the design acceptance criteria in SAR chapter 4 as a basis for conducting the regulatory safety evaluation of the structural design of each of the SSCs used in the operation of the HI-STORE CIS Facility, as described in SAR Chapter 3, “Operations at the HI-STORE Facility.” In addition to the information in the SAR, the staff used the calculations submitted as a part of the application to assess the applicant’s conclusions in the SAR.

In evaluating the design of an SSC, the staff recognized that the applicant’s quality assurance program ensures consistency and accuracy of the numerical values presented in the application. In addition, the staff recognized that the design used proceduralized design codes, which the applicant adhered to in meeting all the design limitations and allowable values for the strength of the structural members. The staff’s evaluation ensures that all the different operational conditions that may occur are considered in the design of the SSCs and that the design follows the regulatory safety requirements to ensure public health and safety in the operations of the HI-STORE CIS Facility.

The HI-STORE CIS Facility uses a dry cask storage system only. It does not use a pool and pool confinement facilities to conduct loading and unloading activities.

5.2 Regulatory Requirements

The regulatory requirements relevant to the installation and structural evaluation of the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: technical information”
- 10 CFR 72.40, “Issuance of license”

- 10 CFR 72.120, “General considerations”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

5.3 Staff Review and Analysis

In conducting its review, the staff used NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000. The staff also used applicable industry codes, which are cited in the text and reference list.

The HI-STORE CIS Facility uses storage systems, transportation casks, and spent nuclear fuel (SNF) canisters, the structural design of which NRC certified under the HI-STORM UMAX canister storage system, the HI-STAR 190 transportation package, and the HI-STORM FW storage system, respectively. In SAR table 5.0.3, “Material Incorporated by Reference in this Chapter,” the applicant incorporated by reference the structural design of the components from the previously certified HI-STORM UMAX and HI-STORM FW final SARs (FSARs). For all components incorporated by reference, the staff concludes that the design and associated documents depicting the design in the referenced FSARs) and in the HI-STORE CIS Facility SAR are applicable. In this chapter of the safety evaluation report (SER), the staff assesses the SSC design information incorporated by reference for the HI-STORE CIS Facility for site-specific differences by evaluating the site-specific value of each design parameter against those specified in the referenced certified designs. SAR chapter 4 also describes the design acceptance criteria for incorporated by reference components. SAR Chapter 2, “Site Characteristics,” presents the site-specific parametric values for natural phenomenon determined by the applicant, which are used for comparison in this chapter of the SER.

The objective of the staff’s review was to ensure compliance of the facility SSCs with the NRC’s regulatory requirements. The review assessed the structural integrity of the ITS SSCs, as well as the NITS SSCs, which are used to ensure the safety of operations at the HI-STORE CIS Facility. SAR chapter 3 provides information on operations at the HI-STORE CIS Facility. The SSCs may provide functions such as confinement, subcriticality, radiation shielding, and retrievability of the stored materials and must be appropriately maintained under all credible loads for normal, off-normal, and design-basis accident conditions, including natural phenomena events. The regulatory requirements specify these conditions and performance requirements. The loads and operational conditions considered in the design of storage units are driven by regulatory requirements and hence reviewed for compliance with 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.” The staff evaluates other buildings in the CIS Facility, such as the cask transfer building (CTB) as other SSCs subject to NRC approval in accordance with section 4.5.2.2 of NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” issued April 2020.

The design of the SSCs is based on the ITS safety or NITS classification in SAR chapter 4. In accordance with 10 CFR 72.3, ITS SSCs are SSCs that maintain the conditions required to store spent fuel, prevent damage to the spent fuel during handling or storage, or provide reasonable assurance that the facility can be operated without undue risk to public health and safety. NITS SSCs are SSCs that are relied upon to support facility operation. The ITS SSCs are further categorized based on their safety importance using an associated quality criterion that relates to the selected design criterion. The more important an SSC is to safety, the more stringent the quality and design criteria. By using this approach, the staff risk-informs the design review.

SAR table 4.2.1, “ITS Classification of SSCs that Comprise the HI-STORE CIS Facility,” lists components, the ITS classification of each component, and the source of the ITS determination. SSCs at the site include those used in initial receipt of SNF from off-site (components 1 and 2 below), those used for storage (components 3 through 8 below), and other SSCs used in short term operations at the HI-STORE CIS Facility:

(1) SNF canisters, along with their fuel basket design, provide leak-tight confinement and criticality control to the stored fuel. This SSC is designated as ITS-A, based on its classification in Holtec Report No. HI-2114830, “Final Safety Analysis Report on the HI-STORM FW System,” Revision 4, dated June 24, 2015.

(2) The HI-STAR 190 transport cask serves as the transportation cask for the SNF canisters. This SSC is designated as ITS-A and serves as a containment during initial receipt of canisters following offsite transport and later during final packaging prior to shipping canisters offsite.

(3) The cavity enclosure container (CEC) is a steel cylinder that defines the canister’s storage space and is classified as ITS-C.

(4) The CEC closure lid is a removable heavy structure placed atop the CEC that blocks sky shine from the stored canister and is classified as ITS-C.

(5) The CEC divider shell is a removable insulated shell that surrounds the stored canister, functioning as a partition between hot and cold air flow, and is classified as ITS-C.

(6) The support foundation pad (SFP) is a slab below the VVM that supports the weight of the VVM and the stored multipurpose canister (MPC). This SSC is classified as ITS-C.

(7) The interim spent fuel storage installation (ISFSI) pad is the top surface of the VVM. It supports the load from loading and unloading operations of the VVM and is classified as ITS-C.

(8) Controlled low-strength material (CLSM) occupies the subterranean space between the CECs and is classified as ITS-C. SAR Figure 4.3.1, “Sub-Grade and Under-Grade Space Nomenclature,” shows the subgrade and undergrade space nomenclature.

For functions not attributable to off-site transport or on-site storage, such as those conducted during short-term operations (STO), the loads and design are based on the HI-STORE CIS Facility-specific design acceptance criteria presented in SAR chapter 4. For SSCs that are of site-specific design in the HI-STORE CIS Facility, SAR chapter 5 provides the engineering rationale supported by summarized design results to demonstrate compliance with the design acceptance criteria in SAR chapter 4, using the design inputs presented in different sections of the SAR. The staff reviewed the structural design of SSCs contained in SAR chapter 5 and the results from the supporting calculations summarized in the SAR to determine whether the structural design of the HI-STORE CIS Facility met the regulatory requirements of 10 CFR Part 72.

The different subsections of the SER below discuss major categories of safety protection systems, which include confinement and containment SSCs, reinforced concrete structures, ITS SSCs, and NITS SSCs relied upon to support facility operation.

5.3.1 Confinement Structures, Systems, and Components

5.3.1.1 Multipurpose Canisters

The only confinement SSC used at the HI-STORE CIS facility is the MPC. Two types of MPCs are permitted to be stored at the HI-STORE site; namely, the MPC-37 and the MPC-89, both of which the NRC has licensed as part of the HI-STORM FW dry storage system (Docket No. 72-1032). Licensing Drawings No. 6505, sheets 1 through 4, Revision 17, and No. 6512, sheets 1 through 3, Revision 18 (both proprietary), in SAR section 1.5, “Licensing Drawings,” show the fabrication details of MPC-37 and MPC-89, along with the construction material.

In SAR table 5.0.3, “Material Incorporated by Reference in this Chapter,” the applicant incorporated by reference the MPC structural evaluation from the certified HI-STORM FW FSAR Revision 4. The staff determined that the only additional structural demands on the HI-STORE CIS Facility were because of site-specific environmental conditions, thermal demands from long-term storage, and natural hazard conditions.

The staff reviewed the comparison of the environmental conditions imposed on the MPC at the HI-STORE CIS Facility to that of the HI-STORM UMAX presented in SAR Table 4.3.2, “Environmental Data for the Licensing Basis in the HI-STORM UMAX Docket and the HI-STORE Site for Different Service Conditions,” and finds that, for all operating situations except the maximum 3 day average ambient temperature for short term operations, the environmental conditions considered for the HI-STORM UMAX envelope those of the HI-STORE CIS Facility. For the HI-STORE CIS Facility 3-day average ambient temperature for short term operations, the staff found the 91 deg F temperature (which is 1 deg F higher than the reported HI-STORM UMAX 90 deg F short term operations maximum temperature) acceptable because the 1 deg F difference in value was negligible relative to the margins of component temperatures with their allowable temperature values, as discussed in SER section 6.3.3.

In SAR section 5.1.4, “HI-STORM UMAX VVM,” the applicant stated that the fatigue evaluations for the HI-STORM FW and HI-STORM UMAX systems, which are found in section 3.1.2.5 of their respective FSARs, remain valid for the proposed 40-year storage term at the HI-STORE CIS Facility. The applicant states that this is because the passive nature and the large thermal inertia of these storage systems protect the MPC enclosure vessel from significant stress cycling. Moreover, as shown in SAR table 6.3.1, “Thermally Significant Parameters for the HI-STORM UMAX ISFSI at HI-STORE and Corresponding Certified Value in the System FSAR,” the maximum MPC heat loads and the ambient temperature conditions applicable to the

HI-STORE CIS Facility are less demanding than the corresponding values for which the HI-STORM UMAX system is certified. As indicated in SAR section 5.1.4, “Structural Analysis,” this reduces the stress amplitudes in the MPC at the HI-STORE CIS Facility and ensures that the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code)-required fatigue evaluations that were originally performed for the UMAX and FW systems remain valid for 40 years of storage at the HI-STORE CIS Facility. The staff finds this an acceptable assessment of the thermal demands for long-term storage.

The staff reviewed SAR table 4.3.1, “Loadings Excluded from Further Consideration in the Qualification of Storage System and Ancillaries at the HI-STORE SAR,” where the applicant compared the site-specific natural hazards evaluated in SAR chapter 2 and the evaluation of natural hazards in the HI-STORM UMAX FSAR. The staff confirmed that the MPC is an underground structure when in storage, there is minimal probability of a tornado missile strike, and high winds, lightning, and snow loads need not be considered as design-basis loads in the structural design as evaluated in SAR chapter 5. The staff finds acceptable the rationale presented for not considering tornado missile impact or the effect of tornadic winds, lightning, and snow loads when the MPC is in storage.

The staff reviewed the design seismic input in SAR table 4.3.3, “Applicable Earthquake and Long Term Settlement Data for the Certified HI-STORM UMAX System and the HI-STORE CIS Facility,” and observed that the list contains the comparison of the Newmark summation of zero period accelerations at grade between the HI-STORM UMAX FSAR design-basis earthquake and HI-STORE site design extension condition earthquake (DECE). This is acceptable, as the applicant conservatively used the DECE to inform the structural evaluation of the HI-STORE VVM and ISFSI system. The staff finds that the seismic inputs at the DECE level for the HI-STORE CIS Facility are lower than those considered for the structural evaluation in the Holtec Report No. HI-2115090, “Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System,” Revision 3, dated June 30, 2016.

The staff finds that the structural design for MPC-89 and MPC-37 is acceptably incorporated by reference without need for modification. Both MPCs can be used for long-term SNF confinement at the HI-STORE CIS Facility. The staff has reasonable assurance that the design parameters of the MPCs adequately bound the thermal loads and environmental conditions, and therefore, the MPC designs comply with the requirements in 10 CFR 72.122(b).

5.3.2 Containment Structures, Systems, and Components

5.3.2.1 Transportation Cask

The HI-STAR 190 transportation cask is used to deliver the loaded MPC-37 or MPC-89 to the HI-STORE CIS Facility and during STO. The safety analysis of the HI-STAR 190 as a transportation package, Holtec Report No. HI-2146214, “Safety Analysis Report on the HI-STAR 190 Package,” Revision 3, dated November 2, 2018, is certified under 10 CFR Part 71 regulations in NRC Docket No. 71-9373. SAR table 1.0.5, “Transport Casks Allowed for Receipt of Canisters at HI-STORE,” specifies that only the HI-STAR 190 transportation cask is to be used to deliver spent fuel to the facility.

To ensure that the load conditions that underlie the transportation certification of the HI-STAR 190 are not exceeded, the staff reviewed the CTB STO described in SAR chapter 3 to confirm that the operational configurations do not expose the HI-STAR 190 to conditions for which it was not qualified under 10 CFR Part 71. The review found that a single failure-proof device is always used to handle the cask. As an additional defense-in-depth measure, the cask remains equipped with its impact limiters during its handling from the rail car, the free-fall height of the cask is maintained below its certified limit in its 10 CFR Part 71 docket, and the cask is kept free of any wrappings that may inhibit its heat rejection function during STO.

The applicant identified two conditions that required additional evaluation when the cask performs functions under the requirements of 10 CFR Part 72. The applicant in SAR table 4.3.6, “Governing Structural and Thermal Loadings for HI-STAR 190 during Short Term Operations,” identifies these specific conditions as SSL-1 and TSL-2. SSL-1 occurs during an operating-basis earthquake when a HI-STAR 190 transportation cask containing a spent fuel canister is in the canister transfer facility (CTF), and TSL-2 is the thermal condition that may occur when a HI-STAR 190 transportation cask containing a spent fuel canister is confined in the CTF with limited air flow. The applicant addressed these two conditions as a part of its analysis in SAR chapter 5 and chapter 6, “Thermal Evaluation,” respectively. The review of the CTF in SER section 5.3.4.3 summarizes the staff’s evaluation of SSL-1. SER section 6.3.4.3 summarizes the staff’s evaluation of the thermal condition that may occur when a HI-STAR 190 transportation cask is confined in the CTF, which is described in SAR section 6.4.2.4, “HI-STAR 190 Thermal Model.”

In addition, the staff reviewed the applicant’s evaluation of the tipping and sliding potential of the HI-STAR 190 when parked on the CTB floor slab during a seismic event. The applicant analyzed this condition in Holtec Report No. HI-2177585, considering both empty and loaded conditions of the HI-STAR 190. The applicant computed the overturning and restoring moment of the cask when subject to a 0.25g seismic acceleration in all three orthogonal directions. The factor of safety against overturning was greater than 1.0, but the factor of safety against sliding was less than 1.0. The applicant used the procedure in American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 43-05, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” to compute the sliding distance using a friction coefficient of 0.3 for concrete to steel. The computed sliding distance with a safety factor (SF) of 2 was 2.0 cm (0.8 inch). The staff agrees with the applicant’s limitation, stipulated in SAR section 10.3.3.1, “Receipt and Inspection of Transportation Cask and Canister,” that if the HI-STAR 190 needs to be set down at any point during the indicated steps, it must be further than 2.5 cm (1 inch) from the edge of any slab to avoid any contact due to sliding. This allows 2.5 cm (1 inch) of edge distance when parking the HI-STAR 190 to ensure that it does not slide off the supporting surface.

Because the CTB has large openings, in Holtec Report No. HI-2177585, the applicant evaluated the HI-STAR 190 against the combined load of tornado wind and a large missile impact. By this analysis, the applicant determined that the maximum horizontal excursion of the cask midpoint (approximately equal to the cask center of gravity) under the given loading is less than 25 cm (10 inches). For a cask tip-over accident to occur, the centroid must undergo a horizontal displacement of 127 cm (50 inches). Therefore, the loadings from wind, tornado, and missile strike events will not result in the cask tipping over or cause excessive sliding of the HI-STAR 190. The applicant evaluated the effect of the impact of the other missiles identified in Regulatory Guide (RG) 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power

Plants," Revision 1, issued March 2007, and determined that the kinetic energy at impact was absorbed by the shell penetration depth, which is less than the thickness of the outer shell.

The staff finds that the structural design for HI-STAR 190 is acceptable for use during short-term storage operations and meets the regulatory requirements of 10 CFR Part 72. Specifically, the staff concludes that the HI-STAR 190 design complies with the requirements in 10 CFR 72.122(b) for the short-term storage function at the CTF.

5.3.2.2 HI-TRAC CS

The HI-TRAC CS functions as a transfer cask for the HI-STORE CIS Facility, providing short-term containment to the MPC while in transit from the HI-STAR 190 transportation cask to the dry storage location. SAR table 4.2.1 classifies the HI-TRAC CS as ITS-A and SAR section 4.2 indicates that it is designed for site-specific conditions. SAR section 5.4.2, "HI-TRAC CS," presents the applicant's structural analysis of the HI-TRAC CS.

The materials for the HI-TRAC CS are the same as those for the HI-STORM FW storage system, and their properties are obtained from summary tables in section 3.3 of the HI-STORM FW FSAR. The stress limits are from ASME Code, Section III, Subsection NF, as specified in SAR section 4.3.3.1. The embedded trunnions are designed to meet the guidance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," issued July 1980.

SAR Table 4.3.4, "Structurally Significant Loadings (SSL) for HI-TRAC CS," lists the loading scenarios for which the HI-TRAC CS is structurally qualified. SAR section 1.5, Licensing Drawing No. 10868, sheets 1 through 5, Revision PR2 (proprietary), shows the details of the HI-TRAC CS. Holtec Report No. HI-2177585 contains the calculations for the different loading scenarios. SAR Tables 5.4.2, "Key Results of HI-TRAC CS Lifting Analysis," (proprietary) and table 5.4.3, "Key Results of Tornado Missile Analysis for HI-TRAC CS," (proprietary) compare the demand to design allowable for the lifting analysis and tornado missile impact, respectively. SAR table 5.4.7, "Bounding Input Parameters and Final Results of HI-TRAC CS and HI-STAR 190 Seismic Stability Analyses," (proprietary) presents the results of the seismic stability analysis, and SAR table 5.4.8, "Fatigue Life of HI-TRAC CS," (proprietary) presents the assessment of the fatigue life of the HI-TRAC CS.

The HI-TRAC CS is equipped with four trunnions (two at the top and two at the bottom). The top two trunnions are used for lifting and must be capable of supporting a fully loaded HI-TRAC CS. The bottom two are used only for rotation during upending or downending the HI-TRAC CS. The bottom gate allows for the transfer of the MPC without exposure during stack-up operations.

The staff reviewed the following three critical elements of the HI-TRAC CS design:

(1) the vertical lift capability of the transfer cask, including the trunnions and the welds of the bottom gate

    The staff's review of the bending and shear stress in the trunnion indicates that there is an adequate SF against the demand. The staff finds that the design acceptance criteria are satisfied by the trunnion design shown in the design drawings and are consistent with the requirements of American National Standards Institute (ANSI) N14.6, "Radioactive Materials-Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) Or More," issued 1993, and those in ASME Code, Section III, Subsection NF, for heavy loads. The bottom trunnions are designed to be identical to those at the top end and are exposed to lesser demand during operation. Therefore, the staff finds the design acceptable.

The staff's review of the bottom shield gate indicates that the bending stress in the shield gate door is within the design acceptance stress allowable with an adequate margin of safety. The associated welds are sized appropriately using ASME Code, Section III, Subsection NF, and meet the code requirements with adequate margin.

(2) the capability of the cask to protect the MPC against tornado winds and tornado missile impacts

In SAR section 5.4.2.4.3, "Tornado Missile Analysis'" the applicant presented information on the safety analysis of the HI-STORE CS against tornado missile impact. The section provides information on the tip-over analysis involving a large missile and the deformation and stress analysis for smaller missiles. The missile spectrum is adopted from RG 1.76, Revision 1. SAR table 5.4.3 (proprietary) shows the results of the tornado missile analysis.

The staff reviewed the applicant's treatment of the effect of tornado wind and tornado-generated missiles and finds that the applicant evaluated the effect on a loaded HI-TRAC CS of a missile impact at the top edge of a free-standing cask with a constant wind force or an instantaneous pressure drop. This combination adds to the angular velocity of the cask caused by the missile strike. The applicant considered an automobile as the missile in this assessment. The analysis assumes that the energy of impact is dissipated through the translational and rotational components of the cask motion. The analysis uses the missile from RG 1.76 and a combination that is the most demanding for tip-over analysis. Through this assessment, the applicant demonstrated the stability of a free-standing HI-TRAC CS. The staff finds that the maximum horizontal excursion of the cask midpoint under the given loading is less than 25 cm (10 inches). A horizontal displacement of 127 cm (50 inches) is required for the cask to tip over.

The staff reviewed the applicant's treatment of smaller missiles with the potential for localized damage, which included a 2.5 cm (1-inch) solid sphere and a 15 cm (6-inch)-diameter pipe. For both missiles, the depth of penetration computed using an energy balancing methodology shows penetration depths are less than the thickness of the outer shell and the MPC lid. Hence, the staff concludes that there is no penetration of the impact zones caused by this missile. The staff reviewed the applicant's treatment of the global effect of the 15 cm (6-inch) pipe impacting the side of the transfer cask. The bending behavior of the cask from this impact and the resulting bending stress are well within the acceptance criteria of ASME Code Level D allowable stress.

(3) the effect of a seismic event during operations with the HI-TRAC CS

The applicant considered seismic effects on the HI-TRAC CS during vertical lifts in the design by scaling the SF under lift conditions.

SAR section 5.4.2.4.4, “Seismic Stability Analysis of Freestanding HI-TRAC CS,” presents the applicant’s assessment of the stability of the HI-TRAC CS, while both empty and fully loaded in a free-standing position subject to a seismic event. SAR table 5.4.7 provides the results of this analysis. Since this condition can occur in the CTB, which has large openings for the train and HI-PORT, the staff reviewed the stability analysis of the HI-TRAC CS on the CTB floor using the safe shutdown earthquake (SSE) seismic input. The staff finds that there is a factor of safety of 1.124 against overturning the loaded HI-TRAC CS. Since the center of gravity of the empty HI-TRAC CS is lower than that of the loaded HI-TRAC CS, the staff concludes that the SF will be higher for the empty HI-TRAC CS. Both the empty and the loaded HI-TRAC slide during the seismic event as the friction force is exceeded by the seismic force. Using an SF of 2, the displacement of the cask was computed as 2.06 cm (0.81 inch). The staff concluded that, if a minimum distance of 2.5 cm (1 inch) is maintained between the HI-TRAC and the edge of any slab, as specified in SAR section 10.3.3.5, “Placement of Canisters in the CEC,” then the cask will be in a stable condition. The staff finds this approach to be appropriate and finds the computed stress to be within the requirements of ASME Code, appendix F.

In addition, since the HI-TRAC CS will be reused continually in the CIS operations, a fatigue evaluation is necessary. The applicant, in SAR section 5.4.2.4.5, “Fatigue Evaluation,” provides information on its fatigue evaluation of the HI-TRAC CS. SAR table 5.4.8 (proprietary) includes the results of this analysis. The staff reviewed this analysis and finds that the applicant established the fatigue life by calculating the number of repeated stress cycles at a particular stress level using the fatigue life cycle curves defined in appendix I to ASME Code, Section III, and applying Miner’s rule, as applicable. The maximum stress is conservatively taken as the bounding stress in the load-bearing component during lifting and handling operations. The analysis used a minimum SF of 2. The results are presented as the number of MPC deployments. Each MPC deployment uses five load cycles. The staff finds that this approach to fatigue life computation acceptable, as it uses the stress limits of ASME Code, appendix I, in computing the maximum number of MPC deployments. The applicant presented the results of the fatigue analysis in Holtec Report No. HI-2177585, Supplement 10, pages 29 and 30.

Based on all the critical elements of its review, the staff finds that the HI-TRAC CS as a transfer cask can support the functional performance required under STO. The cask design criteria and materials are the same as those used for a transportation cask and, hence, are conservative, given the cask will only be handled by single-failure-proof lifting devices. The lifting trunnions and the vertical load-carrying components are designed with an adequate margin of safety. The safety of the cask against seismic and tornado events is adequate.

5.3.3 Ventilated Vertical Module Components

The configuration of VVMs with the ISFSI storage pad forms a typical storage unit for the HI-STORE CIS Facility, as shown in Licensing Drawing No. 10875, sheets 2 through 6, Revision PR2 (proprietary), in SAR section 1.5. This arrangement provides for storage of the MPC in a vertical orientation inside a subterranean cylindrical cavity entirely below the top-of-grade of the ISFSI. The following are the key constituents of a VVM and ISFSI pad:

- VVM components:
  - CEC
  - divider shell
  - closure lid
- ISFSI structures:
  - ISFSI pad
  - SFP
  - subgrade and undergrade

The subsections below give a brief description of each constituent part of the VVM, based on SAR section 1.2.2, “Constituents of the HI-STORM UMAX Vertical Ventilated Module and ISFSI Structures,” with the staff’s finding.

5.3.3.1 Cavity Enclosure Container

As described in SAR section 1.2.2, the CEC consists of a thick-walled shell integrally welded to a bottom plate. The top of the container shell is stiffened by a ring-shaped flange, which is also integrally welded. The constituent parts of the CEC are made of low-carbon steel plate. In its installed configuration, the CEC is interfaced with the surrounding subgrade for most of its height except for the top region, where it is encased in the ISFSI pad.

The CEC is a closed-bottom, open-top, thick-walled cylindrical vessel that has no penetrations or openings. Thus, ground water has no path for intrusion into the interior space of the CEC. Likewise, any water that may be introduced into the CEC through the air passages in the top lid will not drain into the ground water.

The CEC top contains an air plenum box that works in conjunction with the closure lid to channel incoming air into the downcomer flow region of the CEC. The air plenum box also contains anchors in rigid embedded locations for securing the HI-TRAC CS against movement during the stack-up for canister transfer operations. Licensing Drawing No. 10875, sheet 2, Revision PR2 (proprietary), provides a cutaway showing the different components of the VVM. The design of this SSC is incorporated by reference, as indicated in SAR table 5.0.3, and the staff finds no additional site-specific design evaluation is needed except for an evaluation of the loads during stack-up against tornado wind, a missile impact event, and a site-specific seismic event. The anchor bolts that hold the HI-TRAC CS in the stack-up position are welded to the CEC and, hence, transfer these operational loads to the CEC. As noted in SAR section 10.3.3.5, the applicant has opted to use administrative controls to prevent operations with the MPC during a tornado notification in lieu of evaluating the anchor bolts for tornado loading. For the seismic event, SAR section 5.4.1.4, “Structural Analysis,” and table 5.4.1, “Key Results of Stack-up Analysis at HI-STORM UMAX VVM/CTF,” describe and provide the results for the stack-up analysis. SER section 5.3.3.5 summarizes the staff evaluation and finding of the seismic event.

5.3.3.2 Divider Shell

The divider shell is important to the thermal performance of the VVM system. The divider shell, as its name implies, is a removable vertical cylindrical shell concentrically situated in the CEC that divides the CEC into an inlet flow downcomer and an outlet flow passage. As described in SAR section 1.2.2, the divider shell divides the radial space between the canister and the CEC cavity into two annuli. The bottom end of the divider shell has cutouts to enable movement of air from the downcomer to the up-flow region around the canister. The cutouts in the divider shell are sufficiently tall to ensure that, if the cavity were to be filled with water, the bottom region of the canister would be submerged to a depth of several inches. This design feature ensures adequate thermal performance of the system if flood water were to block air flow. The divider shell is not attached to the CEC, which allows its convenient removal for decommissioning or for any in-service maintenance or periodic inspection. The cylindrical surface of the divider shell is equipped with insulation to prevent significant preheating of the inlet air. The insulation material is selected to be water and radiation resistant, as well as nondegradable under accidental wetting.

The design of this SSC is incorporated by reference, as indicated in SAR table 5.0.3, and the staff finds no additional site-specific design consideration is required for its use in the HI-STORE CIS Facility.

5.3.3.3 Closure Lid

As described in SAR section 1.2.2, the closure lid is a steel structure filled with plain concrete that can withstand the impact of the design-basis missiles defined for the site. Both the inlet and outlet vents are located at grade level. The closure lid internals form segregated air channels for air inlet and outlet. A set of inlet passages located on top of the CEC provides maximum separation from the large outlet passage (which is in the center of the lid) and channels the inlet air into the CEC’s air plenum box. As depicted in the licensing drawings in SAR section 1.5 (proprietary), the geometry of the inlet and outlet ducts make the HI-STORE VVM insensitive to the direction and speed of the wind. The closure lid is physically constrained such that horizontal movement is precluded during a design-basis earthquake event or a tornado missile strike. The closure lid is also made of steel filled with shielding concrete to maximize the blockage of skyward radiation issuing from the canister. The geometric configuration ensures that the closure lid will not fall into the canister storage cavity space and strike the canister were it to drop accidentally during its handling. Because the closure lid is the only removable heavy load, the engineered design features to facilitate recovery from an accidental drop provide added assurance that a handling accident at the ISFSI will not lead to any radiological release.

The design of this SSC is incorporated by reference, as indicated in SAR table 5.0.3, and the staff finds no additional site-specific design consideration is required for its use in the HI-STORE CIS Facility.

5.3.3.4 Subgrade and Undergrade

As described in SAR section 1.2.2, the lateral space between each CEC, the SFP, and the ISFSI pad is referred to as the subgrade and is filled with CLSM. Alternatively, “lean concrete” may be used. CLSM is a self-compacted, cementitious material used primarily as a backfill in place of compacted fill. American Concrete Institute (ACI) 229R-99, “Controlled Low-Strength Materials,” uses terms such as flowable fill, unshrinkable fill, controlled density fill, flowable mortar, flowable fly ash, fly ash slurry, plastic soil-cement, and soil-cement slurry to describe CLSMs. ACI 116R-00, “Cement and Concrete Terminology,” defines lean concrete as a material with low cementitious content. CLSM and lean concrete are also referred to as self-hardening engineered subgrade (SES). The subgrade material must meet the shear velocity and density requirements in SAR table 4.3.3, “Applicable Earthquake and Long Term Settlement data for the Certified HI-STORM UMAX System and the HI-STORE CIS Facility.” The space below the SFP is the undergrade. Evaluations in SAR section 5.4.1.4, “Structural Analysis,” show that the SES provides a stable lateral support system to the ISFSI during a design-basis earthquake. The interface between the SES and the native subgrade defines the radiation protection boundary of the ISFSI. The applicable load-affected parts under each loading condition and the applicable structural acceptance criteria related to the HI-STORM UMAX VVM and ISFSI structures provide a complete framework for the required qualifying safety analyses in the SAR. The VVM storage system at the HI-STORE CIS Facility will be functionally identical to that certified in the HI-STORM UMAX docket.

The design of this SSC is incorporated by reference, as indicated in table 5.0.3, and the staff finds that the applicant’s specification that the CLSM’s density and shear wave velocity values, as stated in table 4.3.3, are the same as or bounded by the certified design values is acceptable.

5.3.3.5 Finding on the Storage Components

The only site-specific load that needs further evaluation is the stack-up load, described in SAR figures 3.1.1(l) and (s), which is not a part of the generic design. The applicant presented the methodology for this analysis in SAR section 5.4.1.4, “Structural Analysis.” The analysis used the design-basis earthquake established in SAR table 4.3.3 with the time histories developed from the soil-structure interaction analysis of the CTB. For the stack-up on the ISFSI pad, the HI-TRAC CS is bolted to the CEC with approximately 31.8 cm (12.5 inches) of free length. The applicant used LS-DYNA to perform the stack-up analysis to quantify the kinematic stability of the stack under the postulated earthquake and qualify all the structural elements in the load path. The applicant presented the results of the stack-up analysis in SAR table 5.4.1, which is based on Holtec Report No. HI-2177585. The staff’s review of the information provided in the SAR and supporting calculations indicates that the stability of the cask is maintained during the seismic event, the bolts are adequately sized to withstand the induced seismic loads, the load-bearing components of the shield gate weldment are adequate to withstand the imposed loads, and the concrete-bearing capacity of the ISFSI pad is not exceeded. The staff concluded that the MPC, fuel basket, and fuel assemblies will remain intact in this operation during and after the seismic event.

The tornado wind and missile impact are enveloped by the evaluation of a free-standing HI-TRAC CS on a concrete surface appearing in SER section 5.3.5.2.

The structural analysis of the VVM for all applicable normal, off-normal, and accident loadings is incorporated by reference from the HI-STORM UMAX FSAR. SAR table 5.0.1 compares the design basis and site-specific natural phenomena hazard loads for the HI-STORE CIS Facility. The staff review found that all the site-specific loads are bounded by the design-basis load. SAR table 4.3.3 compares the settlement, subgrade properties, and seismic input for the certified HI-STORM UMAX system and the site-specific HI-STORE CIS Facility. The staff found that the design-basis values envelope those of the CIS Facility. As a result of this enveloping of the site-specific design parameters, the staff finds that the HI-STORM UMAX VVM and ISFSI designs are acceptable for incorporation by reference at the HI-STORE CIS Facility and meet the requirements of 10 CFR 72.122(b).

5.3.4 Concrete Structures Important to Safety

SAR section 1.2.2 describes the ITS concrete structures.

5.3.4.1 Interim Spent Fuel Storage Installation Pad

The ISFSI pad is a monolithic reinforced-concrete structure that provides the load-bearing surface for the cask transporter. The ISFSI pad serves to augment shielding, to provide a sufficiently stiff riding surface for the VCT, as a barrier against gravity-induced seepage of rain or floodwater around the VVM body, and as a shield against a missile. SAR table 5.0.1, “Comparison of DBLs for HI-STORM UMAX System and Site-Specific Loads for HI-STORE CIS Facility,” compares the bounding design parameters for the ISFSI pad to those of the site-specific parameters.

The design of this SSC is incorporated by reference, as indicated in SAR table 5.0.3. Proposed Technical Specification (TS) 4.2.7, “Storage Pads,” provides high-level design features of the ISFSI pad and specifies that the pads shall be designed to support loaded VVMs and a cask transporter with a loaded HI-TRAC CS and to withstand a design-basis earthquake. The staff notes that, apart from the small increase in the weight of the HI-TRAC CS, the design parameters of the certified design include the site-specific values. The applicant justified incorporation by reference by considering that the seismic input for the reference design is much higher than the site-specific seismic design basis. The staff’s review confirmed this, and the staff finds this additional margin of safety adequately accounts for the small increase in weight of the HI-TRAC CS and the staff finds no additional site-specific design consideration is required for use of the ISFSI pad at the HI-STORE CIS Facility. The difference in size of the CIS Facility ISFSI from the generic design does not impact the finding of the staff, as the ISFSI is of the same quality and density of CLSM as that in the generic design.

5.3.4.2 Support Foundation Pad

The SFP is the underground pad that supports the HI-STORE VVM ISFSI. The SFP on which the VVM rests must be designed to minimize long-term settlement. The SFP and the undergrade material must have sufficient strength to support the weight of all the loaded VVMs during long-term storage and earthquake conditions. The generic design (from the HI-STORM UMAX FSAR) considered a 5 × 5 array of VVMs on the SFP. The HI-STORE CIS Facility deploys a 25 × 10 array of VVMs as stated in SAR table 1.1.1. This change from the generic design required a new estimate of the settlement below the SFP. The applicant provided the recomputed settlement in "HI-STORE Bearing Capacity and Settlement Calculations" (proprietary), Holtec Report No. HI-2188143, Revision 6, dated January 20, 2023. The difference in settlement between these two configurations is small (fraction of an inch) and the staff determined that this difference is well within the design margin of the pad.

The design of this SSC is incorporated by reference, as indicated in table 5.0.3, and the staff finds no additional site-specific design consideration is required for its use in the HI-STORE CIS Facility.

5.3.4.3 Canister Transfer Facility

The CTF is a below-grade structure within the CTB. Proposed TS 4.2.9, "Canister Transfer Facility (CTF)," describes high-level design features of the CTF. The top of the CTF structure is integrated into the CTB slab. The CTF is a cylindrical steel structure comprised of [ ]. The steel cylinder rests on a concrete pad termed the CTF foundation pad. The steel shell is surrounded by CLSM. The top of the shell is anchored by the surrounding CTB slab. The details of the CTF, along with the materials of construction, are shown in Licensing Drawing No. 10895, sheets 1 through 4, and SAR section 5.3.3, "Canister Transfer Facility Foundation." During stack-up, the HI-TRAC CS rests on the alignment ring by which it is connected to the CTB slab and the CTF shell. The anchor bolts that hold the HI-TRAC CS during a design-basis seismic event are evaluated in Holtec Report No. HI-2177585 in a stack-up configuration at the ISFSI pad. The stack-up analysis at the ISFSI bounds the condition at the CTF. For the transfer of the MPC to the HI-TRAC CS, the HI-STAR 190 cask is lowered into the CTF. The staff finds the steel shell of the CTF functions as a retainer for the CLSM, creating a shielded space for the HI-STAR 190 cask. The CTF foundation pad carries the weight of the loaded transportation cask. The CTF foundation pad is of the same design as the CTB slab, and the load per square inch is less than the load on the CTB slab below the alignment ring on which the HI-TRAC CS rests. The staff finds the design of the CTF foundation slab to be adequate because the CTB slab in SER section 5.5.4.5 was found to be acceptable for a higher load intensity. The alignment ring holds the HI-TRAC CS in position during the stack-up operation in the CTB. Four anchor bolts through the alignment plate hold the HI-TRAC CS in place during stack-up. SAR section 5.3.3.4, "Structural Analysis," item (e), addresses the acceptance criterion SSL-1 in SAR table 4.3.6, stating the design of the CTF includes restraint blocks at the top and bottom of the cavity space (as shown in the licensing drawings), limiting the movement of the HI-STORE 190 towards the cavity wall during a seismic event and preventing potential impact between the two parts.

The staff finds that the CTF foundation slab is adequately designed to support operational loads during the stack-up operation, consistent with the design acceptance criteria stated in SAR chapter 4. The stack-up loads under seismic conditions are enveloped by the anchor loads for the same condition at the ISFSI. Because of this, the staff considers the anchor bolt size to be adequate for the CTF. The staff recognizes that the CTF is embedded in CLSM and does not anticipate any significant seismic amplification between the CTF foundation slab and the CTB slab. The staff finds that the inclusion of seismic restraints in the CTF design prevents any adverse impact between the CTF and the HI-STAR 190 cask loaded with an MPC.

5.3.4.4 Cask Transfer Building

5.3.4.4.1 Description

The CTB is designed as ITS-C for site-specific conditions, as specified in SAR table 4.2.1. Proposed TS 4.3, “Cask Transfer Building (CTB),” provides the high-level design features of the CTB. The CTB is a reinforced concrete structure with steel framing supporting a reinforced concrete roof on metal decking. The entire roof system is supported vertically by a series of reinforced concrete columns, which are equally spaced around the perimeter of the CTB. The roof load is carried by the roof beams to the roof girders, which run parallel to the north and south walls and rest atop the vertical columns. The roof girders and vertical columns are hinged at their connection points. The space between columns is occupied by concrete shear walls, which resist in-plane lateral forces due to wind and seismic loads. The load handling operations described in SAR chapter 3 are conducted using the CTB overhead bridge crane. SAR section 4.5, “Lifting Equipment (CTB CRANE & VCT), Special Lifting Devices and Miscellaneous Ancillaries,” summarizes the CTB crane specifications; the CTB crane is a top-running bridge crane mounted on a set of rails supported by corbels in the reinforced concrete columns. The CTB walls and columns are fixed at their base where they terminate into a thickened slab haunch, which borders the CTB floor slab. Licensing Drawing No. 10912, sheets 1 through 7, Revision 0.2 (proprietary), shows the various design details of the CTB.

The concrete floor slab of the CTB serves as the processing area for inbound transport packages containing SNF canisters, their unloading, and the transfer of MPCs to the HI-TRAC CS transfer cask for transport to the storage units. The CTB floor slab would also be used for the reverse operations, which would transfer MPCs retrieved from the VVMs to transport casks to be shipped off site, as described in SAR section 10.3.3.6, “Removal of Canisters from the CEC.” From a structural standpoint, the CTB provides the following basic functions:

- a riding surface for the rail car and HI-PORT inside the CTB; the HI-PORT is the principal conveyance used to transport a loaded HI-TRAC CS between the CTB and the ISFSI pad
- a support system for the CTB overhead bridge crane
- a protective enclosure from external natural phenomena for personnel and equipment located within the CTB

As described in SAR section 5.3.2.1, the CTB has a large open interior to facilitate cask handling and conduct canister transfer operations. The east end of the CTB has two large door openings, which provide railcar access for the arriving transport packages. The overhead bridge crane runs the length of the building in the east-west direction, and across the building in the north-south direction, to offload the transport packages from the railcar upon their arrival. The CTB slab supports the cask tilt frame and the below-grade CTF, which is integrated into the CTB slab at the floor level. Once the canister transfer is complete, the loaded HI-TRAC CS is placed onto the stationary HI-PORT before it exits the CTB through the single door opening at the west end of the building for final delivery to the HI-STORE storage system on the ISFSI pad. Licensing Drawing No. 10940, sheets 1 and 2, Revision 0.2 (proprietary), show the haul path to the ISFSI pad and the general layout of the facility.

A more detailed description of the operations that are undertaken inside the CTB appears in SAR section 3.1, "Description of Operations," and is illustrated in SAR Figure 3.1.1, "Cask Handling Summary Illustrations." The CTB has a thick, reinforced concrete slab whose essential design data are summarized in SAR table 4.6.2, "Reference Design Data for the CTB Slab." In SAR table 4.6.3, the applicant listed the load combinations obtained from the different codes, which are from the design acceptance criteria presented in SAR section 4.6.2, "General Design Requirements," and SAR table 4.6.4, "Bounding Load Combinations Used for CTB Analysis," which lists the bounding load combinations used in the analysis.

The staff review of this facility is based on the requirements of 10 CFR 72.122(b)(2), as the CTB is a protective enclosure against natural events and is an ITS structure for the MPC transfer STO. The following SER sections summarize the staff's review to provide assurance that the functional performance of the CTB will meet the regulatory safety requirements of the STO associated with the MPC transfer between a transportation cask and a transfer cask. For this review, the staff used the design acceptance criteria in SAR section 4.6.2 and the CTB crane design parameters in SAR table 4.5.2, "Design Parameters for the CTB Crane."

5.3.4.4.2 Cask Transfer Building Superstructure

The staff reviewed the load path described by the applicant in SAR section 5.3.2.1, "Description of Structural Design," and finds that the reinforced concrete members of the CTB provide an adequate path for the applied loads to reach the foundation. The walls of the CTB, acting as shear walls, transfer the lateral forces that are imposed on the building by the external events and crane operations. The columns are the vertical load-carrying elements pinned at the roof connections and fixed at the bottom where they meet the thickened part of the slab. The columns transfer the roof load and the crane load to the base slab. A vendor supplies the crane girder, crane bridge, and all components of the crane. The building is designed for natural hazards, including wind loads from tornadoes. However, the CTB is not designed to protect against tornado missile impacts. The casks used in the CTB provide tornado missile protection to the MPC during STO. The staff finds this configuration of the structural members adequate for the transfer from the superstructure loads to the building foundation. The CTB has been analyzed for a strike from a large tornado missile with a nonbuilding collapse criterion. The analysis is presented in Holtec Report No. HI-217758. The staff's evaluation of that analysis appears in SER section 5.3.4.4.5.

The staff reviewed SAR tables 4.6.3 and 4.6.4 and finds that the bounding load combinations used in the analysis are consistent with the design codes used for the design of the CTB structural components. SAR table 5.3.1, "Material Properties for CTB & CTF Foundation," summarizes the properties of the concrete. Based on the licensing drawings, the structural steel properties conform to American Society for Testing and Materials (ASTM) A572 or A992 Grade 50. The staff finds these material properties acceptable, as they are consistent with those specified for normal industrial facilities. The staff finds acceptable that the casks within the CTB provide the needed protection to the MPC against tornado missile impact and that the building is not designed as a barrier against tornado-driven missiles.

5.3.4.4.3 Analysis and Design

The staff reviewed the structural analysis presented in SAR section 5.3.2.4, “Structural Analysis,” in which the applicant used different analytical methods to incorporate the effect of the site soil in the building seismic response. The applicant modeled the CTB using the ANSYS finite element (FE) code to determine the building dynamic response under seismic loads and a demand analysis under the different static load combinations. Two separate models were used, one to conduct the dynamic response analysis and the second for the static demand analysis. The staff noted that the applicant used different mesh sizes to perform the dynamic and demand analyses. To provide assurance that the FE models consistently represent the dynamic response characteristic of the structure, the applicant compared the mass and modal frequencies using alternative methods for the two FE models, showing them to be close. For the change in mesh size, the auto meshing feature was used, and no change was made to other inputs. However, the applicant did not perform a mesh size sensitivity check when selecting the mesh size for the dynamic analysis. The staff accepts that, given the input motion and the simplicity of the analytical approach, the mesh size accuracy would not be a dominant factor in the results of the analysis. The staff agrees with the applicant’s rationale that a smaller mesh size in the static model would better capture the demand in the members. For the dynamic analysis, the use of a larger mesh size reduces the computational time without appreciable loss in numerical accuracy. The number of modes computed over the wide frequency range captures the frequencies of interest in the building response. The FE model used for dynamic analysis included the crane located at the middle of the CTB with a suspended loaded cask. This allowed for the development of input motion at the corbel level for the design of the crane girder and crane component by the vendor.

To capture the effect of the foundation soil in the dynamic response of the CTB, the applicant used the guidance in ASCE/SEI 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” issued 2017, to develop strain-compatible soil springs and their corresponding dampers. The SHAKE routine is used to compute strain-compatible soil properties of the in situ soil columns. SAR table 5.3.4, “HI-STORE CTB Soil Column Properties,” presents the soil column properties used in the analysis. These are further processed in ANSYS using the ASCE/SEI 4-16 procedure to establish the translational and rotational soil springs for each of the six degrees of freedom. SAR table 5.3.7, “Soil Spring Stiffnesses Used in CTB Dynamic Model,” presents the resulting soil spring stiffness used in the dynamic analysis. A mode superposition analysis is performed in ANSYS to establish the response of the CTB under the applied input motion. In SAR figures 5.3.4 through 5.3.6, the applicant showed that the response spectra in RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1, issued December 1973, envelope the site-specific response using the site soil columns. The resulting acceleration responses are used in the lateral force calculated according to the analysis methodology of ASCE 7-10, “Minimum Design Loads for Buildings and Other Structures,” issued 2010, to determine the seismic loads imposed at different elevations of the CTB. These seismic loads are used in combination with the other design loads to compute the load demand in the structural elements using the static analysis model in ANSYS.

The staff reviewed the analysis of the loads resulting from different natural phenomena, including missile impact, presented in FE models in SAR figures 5.3.1 (proprietary), 5.3.2, 5.3.3, 5.3.7, 5.3.15, and 5.3.16. The applicant presented, in figures, the stress contours as results of the FE model static analysis in SAR figures 5.3.17 through 5.3.28. The staff reviewed the static load analysis process used to compute the input loads in the demand analysis using the ANSYS static analysis model. The applicant characterized the natural phenomena loads using the provisions of ASCE 7-10. The staff reviewed the methodology used to convert the characterized loads into design loads for use in the demand analysis using the static FE model. The FE model includes the seismic loads as three-directional shear loads distributed using the equivalent static shear distribution methodology of ASCE 7-10, and the effect of the load in three orthogonal directions was combined using the 100-40-40 rule. The applicant presented the detailed calculations in support of the CTB analysis and design in HOLTEC Report No. HI-2210576.

The staff finds the approach to include soil-structure interaction effects in the seismic response analysis of the CTB and the base shear distribution approach of ASCE 7-10 acceptable for demand computation in the CTB structural members. The staff finds the loads and their combinations used to determine the demands in the structural members using the static FE model appropriate and consistent with the design criteria.

SAR table 5.3.3, "Summary of Minimum Safety Factors for CTB Structural Analysis," provides the results of the static analysis, along with the load combinations associated with these demands and the minimum factors of safety for each type of load. The stress contours shown in SAR figures 5.3.17 through 5.3.28 demonstrate that the maximum shear and bending stress in the members did not occur at the same location and no other location was stressed beyond those noted in the table. The staff reviewed these results, along with the explanations provided in SAR section 5.3.2.4.3, "Analysis Results." The staff finds that the results show there is adequate margin in the capacities because the maximum bending and shear demands do not occur at the same location. Even with this conservatism, the interaction ratios and SFs are within the code limits. The staff concludes that the design of the structural framing members is acceptable and includes the loads imposed by the external natural events, as required by the regulations, except for the tornado missile impact loads. In addition, the staff reviewed some elements of the CTB in greater detail because they influence the operational safety within the CTB, as discussed in the following sections.

5.3.4.4.4 Cask Transfer Building Crane

The applicant described the CTB crane in SAR sections 1.2.7, 4.5.2, 5.3.2, and 5.4.3 and in Licensing Drawing No. 12404, Revision 0, in SAR section 1.5. The applicant also proposed TS 4.2.6, "Cask Crane," to establish the applicable design code, maximum lifted load, and operating-basis earthquake. The CTB crane is an electric bridge crane with a single-failure-proof main hoist used to move casks containing the MPC with a main hook lift capacity of 185 metric tons and a secondary hook lift capacity of 20 metric tons. The crane is designed to requirements in ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," issued 2015, as a single-failure-proof crane. The crane is vendor-supplied and requires the applicant to provide site-specific seismic input to qualify the crane components. Using the dynamic FE model, the applicant developed the spectra at 5 percent damping at elevation 44′-4″. Licensing drawings show this elevation on the column to be the top of the corbel. The acceleration results in all directions for nodes at elevation 44′-4″ from all the columns were used to derive the envelope spectral input shown in figures D4-7 to D4-9 in the HI-STORE CTB structural calculation, Holtec Report No. HI-2210576. SAR figures 3.3.12 through 3.3.14 present the same information.

. The vendor used this information in the design of the CTB overhead crane. In Meloni

Calculation No. 722118 (proprietary), the applicant's vendor modified the crane design from another project that envelopes the crane demands for the HI-STORE CIS Facility. In the calculation, the structural response to the earthquake was calculated using the response spectra method. After obtaining the natural frequency and modal participation factors, a generic response spectrum was applied at the crane rail level. The modal responses for each direction were then combined using the square root of the sum of the squares method. The effect of residual masses was evaluated in the cases in which the sum of the participating masses considered in the combination did not reach 90 percent of the total mass.

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]. Page 162 of the calculation presents the site-specific response spectra at the crane runway level. The staff in their review noted that there are exceedances of the site-specific spectra over the assumed design spectra. The vendor calculation has addressed these exceedances to show that at the frequencies of exceedance the stresses are within acceptable limits for the crane member.

The staff finds the process of using a consistent spectrum for the design of the CTB crane to be appropriate for adequate transfer of the seismic load to and from the building to the crane. The vendor demonstrated that the crane design envelopes the stress demands of the HI-STORE CIS Facility.

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]. In SAR table 5.3.5, "Maximum Loads on Crane Runway Corbels," the applicant provided the maximum vertical and shear loads for which the corbels are designed. The crane reactions at the crane rail level from a similar crane shown in Meloni General Arrangement Drawing 3138-007-1, which is referenced on page 9 of the Meloni calculation, include the horizontal and the vertical reactions from the crane analysis. In SAR section 5.3.2.4.3, the applicant has committed to confirm that the corbel design loads envelope the reactions at the corbel after the crane runway beam is designed by the supplier to reconcile the crane. This reconciliation will further ensure that the CTB columns are designed with adequate capacity to comply with the requirements of 10 CFR 72.122(b)(2).

5.3.4.4.5 Tornado Missile Impact on Cask Transfer Building

The staff reviewed tables H3-1 and H3-2 of Holtec Report No. HI-2210576, which list all the load combinations from the referenced design codes and those used in the analysis, respectively, which are the same as those appearing in SAR tables 4.6.3 and 4.6.4. The staff noted that the load combinations include all the applicable loads except those from the tornado missile impact. The applicant analyzed the missile impacts on the casks used in STO and, therefore, did not include them in the CTB design. The applicant concluded that the two smaller missiles (spherical ball and the schedule 40 pipe) did not have sufficient kinetic energy to adversely impact the CTB, even if they penetrated or entered the CTB through the large openings. The impact of a tumbling automobile is accounted for in the applicant's building analysis to demonstrate that the building will not collapse under such an impact.

In addition to qualifying the building under normal load conditions, the applicant assessed the CTB design for a no-collapse condition under the tornado-generated automobile missile impact. Table 2 of RG 1.76 specifies three types of tornado missiles. Among them, schedule 40 pipe and the solid steel sphere can penetrate through the building wall but will not cause the collapse of the building because of their small size and associated energy. The applicant evaluated the automobile missile to determine whether it can cause damage that may lead to a building collapse. In Holtec Report No. HI-2177585, the applicant evaluated the kinetic energy associated with such an impact. The applicant considered two strike zones, as shown in SAR figures 5.3.15, “Large Missile Impact at Center of Long Wall,” and 5.3.16, “Large Missile Impact at Corner Column Location.” The first location is a missile strike near the center of the CTB long wall, and the second is a missile strike at a corner column location. The applicant reanalyzed the demand with these portions of the wall removed in the FE model. Holtec Report No. HI-2177585, table K6-31, shows that all the SFs are greater than 1.0. Therefore, the structural components are still able to support the external loads when certain portions of the wall and the column are damaged by the automobile missile. To illustrate this behavior, figures K6-5 to K6-7 in Holtec Report No. HI-2177585 show the force and moment contours of the impacted wall. The SFs for all CTB structural components are greater than the minimum of 1.0, showing that the automobile missile impact will not cause the concrete wall to fail, and, therefore, the CTB will not collapse due to a tornado missile event.

The staff concludes that the CTB can withstand the impact of a tumbling automobile without collapse, and the casks protect the MPCs against the smaller missiles. Thus, this combination provides reasonable assurance of adequate protection of the MPC from tornado-driven missiles in compliance with 10 CFR 72.122(b)(2)(B)(ii).

5.3.4.4.6 Cask Transfer Building Slab

The CTB floor slab transfers all the superimposed loads to the soil below, and at the haunched portion of the slab column connections, it transfers the moments at the column base to the slab. The applicant analyzed the loads using the static ANSYS model. In Holtec Report No. HI-2210576, the applicant evaluated the maximum demand loads in the 40-inch-thick section and showed that the designed slab thickness has adequate factors of safety. The ANSYS static model addresses the analysis of the slab at the haunched 160 cm (63-inch) thick portion of the slab. The staff concluded that the described load path is reasonable for transferring the loads imposed by the building superstructure to the supporting subsurface under all operating conditions. The slab and building structural members are designed for load combinations required by the design codes specified in the acceptance criteria. In addition, the CTB slab provides a working space for conducting STO. This exposes the slab to short-term operational loads from rail car movement, HI-PORT operation, loaded HI-TRAC CS, and loaded and empty HI-STAR 190 and HI-TRAC CS storage, as concentrated loads. In addition, the slab is integrated with the CTF, where the stack-up operation is performed, subjecting the slab to stack-up operational loads.

In Holtec Report No. HI-2177585, attachment 11, the applicant evaluated the slab as a beam on an elastic foundation for the above loads. The applicant evaluated the slab as designed for each of the above operational load conditions using the SSE seismic condition. SAR figures 5.3.9 through 5.3.11 show the in-structure response spectra at the CTB slab used for the slab design.
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The staff finds that the CTB slab, including the area around the CTF, was designed to include all the loads transmitted by the superstructure and the operational loads arising from the cask transfer and load handling equipment. In addition, the staff finds that the loads from the stack-up at the CTF were considered, as the stack-up analysis for the ISFSI slab bounds the stack-up loads and anchor design for the CTF. The staff finds that the lift-off and overturning of both the HI-STAR 190 and HI-TRAC CS casks, which could be free standing during operations at the CTB, was considered in their respective analyses for seismic and tornado missile impact.

5.3.4.4.7 Conclusion

The staff review finds that the methodologies used to convert the characterized external events into loads for the static FE model analysis for demand calculation are consistent with the industry practice and support the functional needs of the CTB. In addition, the CTB slab has been evaluated for the loads that arise from STO, including punching shear and bearing capacity.

The staff reviewed the calculation on design capacity and noted that the design philosophy and methodology remain unchanged among ACI 318-05 and ACI 318-14, both titled "Building Code Requirements for Structural Concrete," and ACI 349-06, "Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349 06) and Commentary." Even though the capacities computed follow ACI 349-06, they also satisfy ACI 318-14 and ACI 318-05. Based on a review of the different elements of the CTB design, the staff concludes that the CTB design will support its functional requirements and satisfies the applicable portions of 10 CFR 72.122 related to design criteria and safety analysis.

5.3.4.5 Finding on Concrete Structures Important-to-Safety

Based on its review, the staff concludes that the applicant has adequately described the reinforced concrete overpack structures to meet the system description structural requirements in 10 CFR 72.24(a) and (b).

The staff reviewed applicable portions of the documents incorporated by reference to ensure that the design criteria listed for the concrete structure ITS were adequate. Based on its review, the staff concludes that the applicant has adequately discussed the structural design criteria of each cask system's reinforced concrete structures important to safety and meets the requirements in 10 CFR 72.24(c)(1), (c)(2), and (c)(4); 10 CFR 72.40(a)(1); 10 CFR 72.120(a) and (b); 10 CFR 72.122(a), (b), (c), (d), (f), (g), (h), (i), (j), (k) and (l); and 10 CFR 72.128(a) and (b).

5.3.5 Other Important-to-Safety Cask Handling Components

5.3.5.1 HI-PORT

The HI-PORT, described in SAR sections 4.5.4 and 5.5.3, both titled, “HI-PORT,” is the principal conveyance used to transport a loaded HI-TRAC CS from the CTB to the HI-STORE CIS ISFSI pad. The applicant proposed TS 4.2.5, “Cask Transporter,” which describes the structural specifications applicable to the HI-PORT, including the vehicle main frame, overhead beam, lifting attachments, MPC downloader, lifting towers, redundant drop protection, and maximum lifted load. The applicant provided the essential design requirements that the HI-PORT procured for the HI-STORE CIS Facility must fulfill to comply with the SAR. The detailed specification of the HI-PORT is contained in attachment 15 to Holtec Letter 5025071 as Holtec Record No. PS-5025-0001, “Purchase Specification for the HI-STORE HI-PORT” (proprietary), Revision 0, dated March 16, 2022. The constituent parts of the HI-PORT are shown on Licensing Drawing No. 12481, Revision 0, in SAR section 1.5 (proprietary). The HI-PORT consists of two self-powered transport trailers with a center drop deck between the trailers. During transport, the transfer cask is oriented vertically and mechanically fastened to the drop deck at four tie-down locations. The elevated drop deck is supported at both ends by the front and back trailers. The HI-PORT is categorized as ITS-C and is supplied by a vendor to the specifications reflecting the design criteria in SAR section 4.5.4.1, “General Design Requirements.” SAR section 5.5.3.3.1, “Seismic Stability Analysis,” presents the seismic stability analysis of the HI-PORT carrying a loaded HI-TRAC CS. The detailed analysis appears in Holtec Report No. HI-2177585. In addition, Supplement 18 of Holtec Report No. HI-2177585 specifies, as a requirement, a fatigue assessment with a capability of 500 canister deployments. SAR section 5.5.3.3.2, “Tornado Wind and Missile Evaluation,” (proprietary) presents the stability of the HI-PORT with the HI-TRAC under automobile tornado missile impact.

The staff’s review of the information in the SAR, the purchase specification, and Holtec Report No. HI-2177585 finds that there is adequate information for the custom design of the HI-PORT. The staff’s review of the calculations shows that the HI-TRAC remains attached to the HI-PORT under seismic conditions when attached to the drop-deck with]. The HI-PORT assembly will not slide off the edge of the haul path if a minimum clearance of 7.6 cm (3 inches) is maintained between the edge of the HI-PORT and edge of the haul path in the transverse direction. In SAR section 10.3.3.5 the applicant identified this as an operational condition. The staff reviewed the information presented on the analysis of tornado missile impact and agrees that the HI-TRAC cannot slide off the HI-PORT, as it is secured to the HI-PORT deck. The combined center of gravity of the HI-PORT and HI-TRAC is lower than the center of gravity of the free-standing HI-TRAC, and the significant increase in the restoring moment caused by the large base of the HI-PORT ensures that the missile impact is unable to topple the HI-TRAC CS loaded on the HI-PORT. Additionally, the vendor committed to demonstrating a service life equivalent to deployment of 500 canisters considering high and low stress cycles. Based on the above, the staff finds that the description of the HI-PORT in the SAR and docketed materials satisfies the requirements of 10 CFR 72.24, and the design of the HI-PORT satisfies applicable portions of 10 CFR 72.122 related to design criteria and safety analysis.

5.3.5.2 Vertical Cask Transporter

The VCT is the principal load handling equipment used outside the CTB for moving the MPC from the transfer cask into the VVM. The SAR provides information about the VCT in sections 1.2.7, 3.1.4.3, 4.5.3, and 5.5.2 and in Licensing Drawing No. 12432 (proprietary) in SAR section 1.5. The VCT is a vendor-supplied item with an ITS-A classification. The VCT design classification is ITS-A. It is bounded by the NRC-certified design described in the HI-STORM FW system FSAR. The design-basis requirements and loadings on the VCT appear in SAR section 4.5.3.3, “Structural,” and table 4.5.3, “Design Basis Conditions and Loadings on the Vertical Cask Transporter,” respectively. Used in conjunction with the HI-TRAC CS lift links, the VCT provides the critical lifting and handling functions associated with the canister transfer operation at the ISFSI pad. Licensing Drawing No. 12432 shows the licensing information for the VCT as custom-designed equipment on a set of caterpillar tracks with a robust gear train and transmission powered by a diesel engine. The VCT is raised or lowered by a set of hydraulically activated towers with redundant drop protection features to prevent accidental load drops in the event of a hydraulic or power failure. These features are designed under the single failure criterion.

The staff evaluated the acceptance criteria and established that, apart from the site-specific demands from seismic, tornado wind, and missile impact and any change in transfer cask weight, the HI-STORE CIS Facility imposes no other site-specific or operational demands on the VCT design. In SAR section 5.5.2.3, “Structural Analysis, the applicant discussed its evaluation of the seismic stability of the VCT for the site-specific DECE seismic loads using the prescribed 7 percent grade for the path. The applicant evaluated overturning in the transporter’s transverse direction, which is shorter than the longitudinal direction, for two bounding cases: one with an empty VCT and the other VCT with a loaded HI-TRAC CS. The HI-TRAC CS is seismically restrained to the VCT, preventing any out-of-plane motion. [

. For sliding, the applicant used a friction coefficient of 0.3 between concrete and steel. The SFs for sliding in both the empty VCT and the VCT with a loaded HI-TRAC CS are less than 1.0, which is presented in Holtec Report No. HI-217758; therefore, the VCT will slide under the bounding seismic event. The applicant computed the maximum sliding distance using an SF of 2.0, and the estimated sliding distance is approximately 2.5 cm (1.0 inch). The impact of the tornado wind and associated missile impact is compared to the analysis of the free-standing HI-TRAC CS under the same loading. The applicant found that the restoring moment of the VCT configuration is much higher than that of the free-standing HI-TRAC CS. Thus, this loading condition is enveloped by the analysis of the free-standing HI-TRAC CS.

The staff finds that the empty VCT and loaded VCT will remain stable and will not lift under the bounding design-basis seismic event when staged or traveling on the approach apron or on the ISFSI pad. The staff noted that the analysis is valid only to a nominal HI-TRAC CS lift height of 30 cm (12 inches). To prevent the VCT from sliding off the ISFSI pad or the approach apron, the VCT operations need to maintain a minimum edge distance of 2.5 cm (1.0 inch) from the edge. To ensure stability related to lift height and sliding, the applicant imposed in SAR section 10.3.3.5 a 30 cm (1-foot) maximum lift height for a HI-TRAC CS and a limitation that, if the HI-TRAC needs to be set down at any point, it must be further than 2.5 cm (1 inch) from the edge of any slab. In addition, the applicant also imposed a limitation in SAR section 10.3.3.5 that the loaded HI-PORT and VCT must be kept 7.6 cm (3 inches) from the edge of the haul path and pad. The staff notes that SAR table 5.0.2, “Bounding Weights for Cask Components and Ancillary Equipment,” clarifies that the specifications pertaining to the HI-TRAN 225 in Licensing Drawing No. 12432 apply to the VCT. The staff accepts the applicant’s justification for the restoring effect of the large mass of the VCT and HI-TRAC CS, along with the low center of gravity and the wide space between the tracks. The staff finds that the VCT with the HI-TRAC CS will remain unaffected by the tornado missile impact and tornado wind, as they are enveloped by the free-standing analysis. Based on the above, the staff finds that the description of the VCT in the SAR and docketed materials satisfies the requirements of 10 CFR 72.24 and the design of the VCT satisfies applicable portions of 10 CFR 72.122 related to design criteria and safety analysis.

5.3.6 Special Lifting Devices

Special lifting devices is a category of devices that are designed to ensure that the load path of a lift is single failure proof. They are designed to special requirements as stated here. The facility uses several of these in conducting operations. The SER sections below address each device individually. These components transmit loads from lifting attachments, which are structural parts of either the HI-TRAC CS transfer cask or the HI-STAR 190 transportation overpack, to the hooks of an overhead hoisting system and are classified as special lifting devices. For the HI-STORE CIS Facility, special lifting devices are discussed in SAR section 5.4.6, “Other Special Lifting Devices,” and the requirements are specified in TS 4.2.8, “Special Lifting Devices.” The design of the special lifting devices is based on ANSI N14.6. The stress criteria follow those of ASME Code, Section III, NF-2300, and the limitations stated in NUREG/CR-1815, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick,” issued June 1981, as discussed in SAR section 4.5.1.2, “Stress Compliance Criteria Applicable to Special Lifting Devices (SLDs).” The materials used for the fabrication of these devices are listed in their respective drawings or are adopted from ASME Code, Section II, part D, as stated in SAR section 5.4.6.3, “Material Properties.” The single-failure-proof criterion for special lifting devices is met by doubling the SF or by using redundancy.

5.3.6.1 HI-TRAC CS Lift Yoke

The HI-TRAC CS lift yoke is shown in Licensing Drawing No. 10900, sheet 1, Revision P2 (proprietary), in SAR section 1.5. The lift yoke attaches to the two top trunnions of the HI-TRAC CS during the lifting operation. The lift yoke, in addition to the weight of the HI-TRAC CS and the shield gate, experiences the weight of the loaded MPC as it is moved into the HI-TRAC CS during stack-up at the CTF. The detailed stress calculations for the lift yoke appear in Holtec Report No. HI-2177585. The lift yoke is classified as ITS-A and is designed to site-specific demands.

The staff confirmed that the applicant reviewed each element of the lift yoke, including the strong back plate, the lift arm, the main pin, the actuator plate, and the actuator pin for the HI-TRAC CS, against its code allowable for all failure modes and loads, including seismic. A review of these SFs shows that the minimum SF is 1.20 at the actuator pin. The staff finds all the parts of the lift yoke are designed to the acceptance criteria in SAR section 4.5.1, “Design

Requirements Applicable to Lifting Devices and Special Lifting Devices," and meet the allowable limits with an SF greater than 1.0.

5.3.6.2 Transport Cask Horizontal Lift Beam

The transport cask horizontal lift beam is shown in Licensing Drawing No. 10894, sheets 1 through 3, Revision PR2 (proprietary), in SAR section 1.5. The horizontal lift beam is used to lift the HI-STAR 190 cask in the horizontal orientation from the rail transport to the tilt frame. The applicant evaluated the horizontal lift beam against criteria in SAR section 5.4.6 and table 5.4.6, "Minimum Calculated Safety Factors for Other Special Lifting Devices," and SSE seismic loading. The lift beam slings are sized for a design factor of 10 relative to the maximum lifted load. The fabrication material is specified in the drawing and is designed to its safety category ITS-A. The stress evaluation of the horizontal lifting beam appears in Holtec Report No. HI-2177585, Supplement 9.

The staff's review of the information presented in the SAR, supported by the calculations submitted as part of the application, indicates that the applicant evaluated the stress in each of the fabrication parts of the horizontal lifting beam. The staff finds that all lift arm components are evaluated and within the allowable stress limits by an SF of greater than 1.0 ([ ]). Because of this, the staff determined that the horizontal lift beam is consistent with the design acceptance criteria, and the design is in accordance with the performance level expected of an ITS-A component that conforms to NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," issued February 1996. The calculated minimum SF for special lifting devices is compiled in SAR table 5.4.6. For these reasons, the horizontal lift beam is qualified as a special lifting device in accordance with NUREG-0612 and ANSI N14.6 for loads up to the dynamic load factor of 1.15 times 418.7 kips.

5.3.6.3 Transport Cask Lift Yoke

The HI-STAR 190 lift yoke is shown in Licensing Drawing No. 10902, sheet 1, Revision P2 (proprietary), in SAR section 1.5. The lift yoke attaches to the two top trunnions of the HI-STAR 190 during the lifting operation. The lift yoke experiences the weight of the HI-STAR 190 as it is moved to the CTF. The detailed stress calculations for the lift yoke appear in Holtec Report No. HI-2177585. The lift yoke is classified as ITS-A and is designed to site-specific demands.

The staff's review of the information presented in the SAR, supported by the calculations submitted as a part of the application, indicates that the applicant evaluated the stress in each of the fabrication parts of the HI-STAR 190 lift yoke. The staff finds that the HI-STAR 190 lift yoke components are all within the allowable stress limits by an SF of greater than 1.0]. As a result, the staff determined that the HI-STAR 190 lift yoke is consistent with the design acceptance criteria, and the design is in accordance with the performance level expected of an ITS-A component that conforms to NUREG/CR-6407. Therefore, the HI-STAR 190 lift yoke is qualified as a special lift device in accordance with NUREG-0612 and ANSI N14.6.

5.3.6.4 HI-TRAC CS Lift Link

The HI-TRAC CS lift links are used during STO with the VCT. The links experience direct tensile loads from the weight of the loaded HI-TRAC CS and are evaluated as special lifting devices under normal and seismic conditions. Design details of the lift link appear in Licensing Drawing No. 10901, Revision PR2 (proprietary), in SAR section 1.5. Holtec Report No. HI-2177585 contains the details of the lift analysis.

The staff review finds that the HI-TRAC CS lift links have an SF of greater than 1.0 under the given loads and load conditions. [ ]. The staff concludes that the design of the HI-TRAC lift links is adequate to meet the performance requirements of STO at the HI-STORE CIS Facility because the SF for all the components is above 1.0.

5.3.6.5 Multipurpose Canister Lift Attachment

The MPC lift attachment is a special lifting device that is used to lift the MPC into the HI-TRAC CS and subsequently to lower the MPC into the CEC of the HI-STORE VVM. The MPC lift attachment is shown in Licensing Drawing No. 10891, Revision P2 (proprietary), in SAR section 1.5. The drawing shows the material used in the fabrication of the MPC lift attachment. The MPC lift attachment is designed for site-specific conditions as a special lifting device in accordance with NUREG-0612 and ANSI N14.6 and is classified as ITS-A. The stress is limited by the requirements in ASME Code, Section II, appendix F. Holtec Report No. HI-2177585 presents the design details.

The staff's review of the design finds that the SF is greater than 1.0 for all modes of failure of the components that form the MPC lift attachment. The lift pin stress was limited in accordance with NUREG-0612. [ ]. The staff finds the design of the MPC lift attachment structurally adequate for its designed purpose because the SF for all the components is above 1.0.

5.3.6.6 Multipurpose Canister Lifting Device Extension

The MPC lifting device extension is used in conjunction with the MPC lifting attachment. Licensing Drawing No. 10889, sheets 1 through 3, Revision PR3 (proprietary), in SAR section 1.5 show the fabrication components of the lifting extension. The materials of construction are listed on the drawings, and the MPC lifting device extension is categorized as an ITS-A with a site-specific design. The seismic input used in the analysis is the design-basis earthquake for the HI-STORE CIS Facility. The extension is designed to the requirements of a special lifting device. Holtec Report No. HI-2177585 presents the design details.

The staff's review of the calculations shows that the minimum SF for the assembly is [ ]. The staff finds the design of the MPC lifting device extension structurally adequate for its designed purpose because the SF for all the components is above 1.0.

5.3.7 Transport Cask Tilt Frame

The transport cask tilt frame is used in conjunction with the CTB crane and its special lifting devices to upend or downend the HI-STAR 190 transport cask between the vertical and horizontal orientations. Proposed TS 4.2.10, “Tilt Frame,” provides the high-level design features of the tilt frame. The transport cask tilt frame consists of a set of trunnion support stanchions and a cask support saddle. The trunnion support stanchions engage the cask’s rotation trunnions and provide a low-friction rotation point for cask tilting. The cask support saddle contacts the upper portion of the cask when the cask reaches the horizontal orientation. The trunnion support stanchion assembly is bolted to the CTB slab at its base while in use. Licensing Drawing No. 10899, sheets 1 through 3, Revision RP2 (proprietary) show the different elements of the tilt frame and saddle. The transport cask tilt frame has a safety classification of ITS-C and is designed as a site-specific SSC. The drawings identify the construction materials, the properties of which are taken from ASME Code, Section II, part D. The applicant performed a detailed analysis for the tilt frame in Holtec Report No. HI-2177585.

The staff agrees with the applicant that the transport cask tilt frame is not a lifting device, and any operational load drop is prevented as the CTB crane is attached to the cask during the drop limiter removal process. Thus, the transport cask tilt frame is only evaluated for the stress limits in accordance with design acceptance criteria. The applicant performed an FE-based stress analysis, using the ANSYS code, under seismic and normal load conditions. The tilt frame assembly is evaluated for the following conditions using a dynamic load factor of 15 percent:

- upending the HI-STAR 190 at 90 degrees from the horizontal
- upending the HI-STAR 190 at 45 degrees from the horizontal
- HI-STAR 190 on the tilt frame with the impact limiters

For normal and seismic conditions, the staff finds that the analysis shows all the stresses in the transport cask tilt frame and saddle elements as having an SF greater than 1.0, and the stresses in the welds are within the allowable limits. As a result, the staff concludes that the transport cask tilt frame and saddle design under all load conditions are acceptable.

The analysis assumes that the transport cask tilt frame is anchored to the CTB floor slab. The staff’s review of the loads in the anchor bolts using the results of the seismic analysis of the tilt frame and the saddle shows that the interaction ratio for both the transport cask tilt frame and the saddle tee-anchor bolts is less than 1.0. The tee-anchor bolts hold the transport cask tilt frame and saddle to a grid plate, which is embedded in the CTB slab and restrained in it with J-hooks. Holtec Calculation No. HI-2177585, attachment 13A, page 13A-61, establishes the bending and shear capacity of the grid slates. The staff will further affirm this and conduct a detailed evaluation of the entire anchoring assembly when the as-built configurations accounting for the different cask systems and field conditions become available.

5.3.8 Slings

SAR section 4.5.1.1, “Stress Compliance Criteria Applicable to Lifting Devices (LDs),” provides the compliance criteria for those devices that do not qualify as special lifting devices but are used for lifting, such as slings. All slings used at HI-STORE CIS Facility conform to the stress limits of ANSI B30.9, “Slings.”

The staff finds that the use of slings at the HI-STORE CIS Facility is acceptable because the slings conform to ANSI B30.9, which uses a high level of conservatism in its stress limits and is an acceptable code to use, as it is an industry practice for slings.

5.3.9 Fatigue

The evaluation of fatigue failure modes of primary structural members whose failure may result in the uncontrolled lowering of the load is necessary for many of the SSCs used at the HI-STORE CIS Facility. The applicant evaluated the potential of fatigue failure of SSCs and presented its detailed analysis in Holtec Report No. HI-2177585. SAR table 5.4.8 evaluates the fatigue life of the HI-TRAC CS and its components, and SAR table 5.4.9, “Fatigue Life of Lifting Ancillaries,” presents the fatigue life of the lifting ancillaries.

The applicant used the number of MPC deployments as a measure of the fatigue life of lifting and transportation equipment used at the CIS Facility. The staff concluded this was a very practical means of measuring the useful life of SSCs undergoing repeated loading and unloading cycles. The staff found that the Hi-TRAC CS is the SSC that has the lowest fatigue life, which means that the HI-TRAC will need to be refurbished sooner than any of the lifting or transportation SSCs. The staff based its review on Holtec Report 2177585, Supplement 10, which captures the computation of the Fatigue Life of the HI-TRAC CS and Lifting ancillaries. The staff’s review concludes that the fatigue life of the SSCs have been adequately defined.

5.3.10 Structures, Systems, and Components Not Important to Safety

In SAR table 4.2.1, “ITS Classification of SSCs that comprise the HI-Store CIS Facility” the applicant provided the ITS classification of the SSCs using the guidance of NUREG/CR-6407. In this classification, the VCT is classified as ITS-A, because that is the highest safety classification of its subcomponents. However, the VCT also includes some subcomponents at lower safety classifications. Since the VCT is a single failure proof lifting device, the applicant has classified all components directly in the line of the load as ITS-A. These are the overhead beam and the lifting towers. Other subcomponents that support the lift are classified as ITS-B, and the rest of the subcomponents are classified as NITS. The staff finds that graded approach to classification acceptable as it follows the guidance of NUREG/CR-6407 as components are classified in accordance with their safety function and maintenance of integrity of the MPC during the lifting and transportation process.

5.4 Evaluation Findings

Based on the information in the application and staff’s review as discussed above, the staff concludes the following:

- The SAR and docketed materials relating to the description of confinement SSCs, reinforced concrete structures, and other ITS SSCs meet the requirements of 10 CFR 72.24(a) and (b).

- The SAR and docketed materials relating to structural design criteria, including applicable codes and standards, meet the requirements of 10 CFR 72.24(c)(1), (c)(2), and (c)(4); 10 CFR 72.40(a)(1); 10 CFR 72.120(a) and (b); 10 CFR 72.122(a), (b), (c), (d), (f), (g), (h), (i), (j), (k), and (l); and 10 CFR 72.128(a) and (b).

- The SAR and docketed materials relating to suitable material properties for use in the design and construction of the confinement SSCs, reinforced concrete structures, and other ITS SSCs meet the requirements of 10 CFR 72.24(c)(3).

- The SAR and docketed materials provide adequate analytical and test reports ensuring that the structural integrity of the confinement SSCs, reinforced concrete structures, and other ITS SSCs meet the requirements of 10 CFR 72.24(d)(1), (d)(2), and (i) and 10 CFR 72.122(b)(1), (b)(2), and (b)(3), (c), (d), (f), (g), (h), (i), (j), (k), and (l).

Therefore, the staff concludes that the regulatory requirements that pertain to the structural design of the SSCs for the HI-STORE CIS Facility are met.

5.5 References

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ACI, ACI 318-05, “Building Code Requirements for Structural Concrete,” 2005.

ACI, ACI 349-06, “Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-06) and Commentary,” 2006.

ACI, ACI 116R-00, “Cement and Concrete Terminology,” 2007.

ACI, ACI 318-14, “Building Code Requirements for Structural Concrete,” 2014.

ASCE 7-10, “Minimum Design Loads for Buildings and Other Structures,” 2010.

American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI), ASCE/SEI 43-05, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” 2005.

ASCE/SEI, ASCE/SEI 4-16, “Seismic Analysis of Safety Related Nuclear Structures,” 2017.

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American National Standards Institute (ANSI) N14.6, “Radioactive Materials Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) Or More,” 1993.

Code of Federal Regulations, Title 10, Part 71, “Packaging and Transportation of Radioactive Material.”

Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.”

Holtec International, Holtec Report No. HI-2114830, “Final Safety Analysis Report on the HI-STORM FW System,” Revision 4, June 24, 2015. Agencywide Documents Access and Management System Accession No. ML15177A338.

Holtec International, Holtec Report No. HI-2115090, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Revision 3, June 29, 2016. ML16193A339.

Holtec International, Holtec Report No. HI-2146214, "Safety Analysis Report on the HI-STAR 190 Package," Revision 3, November 2, 2018. ML18306A911.

Holtec International, Holtec Record No. PS-5025-0001, "Purchase Specification for the HI-STORE HI-PORT" (proprietary), Revision 0, March 16, 2022. ML22108A131.

Holtec International, Holtec Report No. HI-2210576, "Structural Analysis of the HI-STORE Cask Transfer Building" (proprietary), Revision 1, March 24, 2022. ML22108A129.

Holtec International, "Attachment 2 - Proposed HI-STORE License/Technical Specifications," November 23, 2022. ML22331A005.

Holtec International, "HI-STORE Bearing Capacity and Settlement Calculations" (proprietary), Holtec Report No. HI-2188143, Revision 6, January 20, 2023. ML23020A134.

Holtec International, Holtec Report No. HI-2177585, "Structural Calculation Package for the HI-STORE CIS Facility" (proprietary), Revision 5, January 20, 2023. ML23020A135.

Holtec International, Holtec Report No. HI-2167374, "Licensing Report on the HI-STORE CIS Facility, Revision 0T, January 20, 2023. ML23025A112.

Meloni Tecno-Handling s.r.l., Meloni Calculation No. 722118, "Static, Seismic Analysis and Mechanical Calculations for HI-STAR Crane" (proprietary), Revision 0, March 16, 2022. ML22108A132.

U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, December 1973. ML003740207.

NRC, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980. ML070250180.

NRC, NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick" (nonpublic), June 1981. ML062710437.

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NRC, Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, March 2007. ML070360253.

NRC, NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities," April 2020. ML20121A190.

6 THERMAL EVALUATION

In Revision 0T of safety analysis report (SAR), dated January 20, 2023, chapter 6, “Thermal Evaluation,” Holtec International (the applicant) presented information to demonstrate that the proposed Holtec HI-STORE Consolidated Interim Storage (CIS) Facility structures, systems, and components (SSCs) important to safety (ITS) and spent nuclear fuel material temperatures remain within allowable values or criteria for normal, off-normal, and accident conditions.

6.1 Scope of Review

Unless otherwise stated, the staff evaluated the HI-STORE CIS Facility system thermal analyses by reviewing the SAR, documents cited in the SAR, the applicant’s responses to the requests for additional information (RAIs), and other relevant literature. In particular, SAR chapter 6 provided information to demonstrate the proposed HI-STORE CIS Facility ITS SSCs and spent nuclear fuel cladding temperatures remained within allowable values as well as criteria for normal, off-normal, and accident conditions. In addition, the staff’s review considered information in SAR chapter 1, “General Description”; SAR chapter 2, “Site Characteristics”; SAR chapter 3, “Operations at the HI-STORE Facility”; SAR chapter 4, “Design Criteria for the HI-STORE CIS Systems, Structures, and Components”; SAR chapter 10, “Conduct of Operations”; SAR chapter 15, “Accident Analysis”; and SAR chapter 16, “Technical Specifications.”

6.2 Regulatory Requirements

The thermal evaluation requirements for the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.92, “Design basis external natural events”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related greater than Class C waste, and other radioactive waste storage and handling”

6.3 Staff Review and Analysis

The areas of review included decay heat removal systems; material temperature limits; thermal loads and environmental conditions; and thermal-related analytical methods, models, and calculations. Section 6.3.5 of this safety evaluation report (SER) provides the staff’s evaluation associated with fire protection.

The following sections describe the staff’s evaluation to determine whether the decay heat removal system is capable of reliable operation so that the temperatures of materials used for ITS SSCs and fuel assembly cladding material remain within the allowable limits under normal, off-normal, and accident conditions. In addition, the staff reviewed and evaluated the dry fuel assembly transfer systems for adequate decay heat removal under normal, off-normal, and accident conditions. Lastly, the staff reviewed the information in the application to confirm that the thermal design of the HI-STORE CIS Facility has used acceptable analytical methods.

6.3.1 Decay Heat Removal Systems

6.3.1.1 Dry Storage Systems

According to SAR section 6.1, “Decay Heat Removal Systems,” and section 6.4, “Applicable Systems, Analytical Methods, Models, and Calculations,” the storage system to be deployed at the HI-STORE CIS Facility is based on the previously certified HI-STORM UMAX storage system, as described in HI-STORM UMAX final safety analysis report (FSAR) (Holtec Report No. HI-2115090, Revision 3) and, therefore, many of the thermal analyses in the HI-STORM UMAX FSAR are relevant to the HI-STORE system. The staff confirms that the ventilated system, whereby ambient air is drawn through the vertical ventilated module (VVM) plenum to cool a multipurpose canister (MPC), is designed with passive heat removal capabilities under normal, off-normal, and accident conditions. SAR table 10.3.1, “Maintenance and Inspection Activities for the HI-STORM UMAX VVM Systems,” notes that VVM plenum inspections would occur annually or following a severe weather event that may introduce foreign materials, in order to visually verify inlet and outlet plenums are free of significant material and air passages are not degraded. Likewise, SAR section 10.3.3.5, “Placement of Canisters in the CEC,” states that openings from air vent extensions temporarily removed from surrounding VVMs to aid in transporting the HI-TRAC CS transfer cask are covered by a low-profile temporary cover screen assembly that prevents debris from entering the vent opening without blocking air flow. The proprietary drawings in SAR section 1.5, “Licensing Drawings,” provide the specific design information for the HI-STORE system. Differences between the HI-STORE system and the HI-STORM UMAX system, described in the referenced HI-STORM UMAX FSAR, include the inlet and outlet vent arrangement (shown in SAR Drawing 10875, “HI-STORM UMAX Vertical Ventilated Module (Version C),” Revision PR2, proprietary). Thermal analyses specific to the HI-STORE CIS Facility were presented in the SAR, RAI responses, and HI-STORE calculation package reports, as described in this SER chapter.

Based on the above, the staff concludes that the applicant described the heat removal characteristics of the dry storage system in sufficient detail for the staff to determine that the thermal requirements of 10 CFR 72.122(h)(1) and 10 CFR 72.128(a) have been adequately satisfied.

6.3.1.2 Dry Transfer Systems

According to SAR chapters 1, 3, 4, and 6, the major operations associated with the HI-TRAC CS transfer cask include transfer of the loaded MPC from the HI-STAR 190 transportation cask to the HI-TRAC CS transfer cask using the canister transfer facility (CTF) located inside the cask transfer building (CTB), transfer of the HI-TRAC CS transfer cask from the CTB to the HI-STORM UMAX subterranean independent spent fuel storage installation (ISFSI) using the HI-PORT conveyance and vertical cask transporter (VCT), and transfer of the loaded MPC from the HI-TRAC CS to the HI-STORM UMAX storage system. SER section 6.3.4 includes the summary description of the dry transfer thermal analyses. The staff confirmed that the thermal description of the transfer process showed that the system is designed with passive heat removal capabilities.

Based on its review, the staff concludes that the applicant described the dry transfer system in sufficient detail for it to determine that the thermal requirements of 10 CFR 72.122(h)(1) and 10 CFR 72.128(a) have been adequately satisfied.

6.3.2 Material Temperature Limits

6.3.2.1 General Considerations

SAR chapter 4, including SAR table 4.0.1, “HI-STORM UMAX FSAR Material Incorporated in this FSAR by Reference”; table 4.4.1, “Permissible Temperature Limits for HI-TRAC CS and CTF Materials”; and table 4.4.4, “HI-STAR 190 Materials Temperature Limits,” provide temperature limits of the material of construction for storage and transfer components of the HI-STORE system and the HI-STAR 190 transportation package components. According to SAR table 4.0.1, the MPC (which includes the MPC-37 or MPC-89 fuel assembly content and the backfilled inert helium gas that has a high thermal conductivity) and VVM temperature limits are found in table 2.3.7, “Temperature Limits,” of the HI-STORM UMAX FSAR, which is incorporated by reference. The staff confirmed that the MPC and VVM material temperature limits are provided in the HI-STORM UMAX FSAR. The materials evaluation (chapter 17 of this SER), section 17.3.4, “Properties of Metallic Materials,” further discusses the material temperature limits.

6.3.2.2 Fuel Cladding

As noted in SAR table 4.0.1, the fuel cladding temperature limits (e.g., normal, short-term, off-normal, and accident conditions) for the fuel assemblies within the MPC-37 and MPC-89 were provided in HI-STORM UMAX FSAR table 2.3.7, which the applicant incorporated by reference. The materials evaluation in chapter 17 of this SER, section 17.3.17, “Spent Nuclear Fuel,” further discusses the material temperature limits.

6.3.2.3 Concrete

SAR table 4.4.1 provided the concrete allowable temperature limits for the HI-TRAC CS transfer cask. The materials evaluation in chapter 17 of this SER, section 17.3.8, “Radiation Shielding Properties,” further discusses the concrete performance and temperature criteria.

6.3.2.4 Extreme Low Temperatures

SAR table 2.3.1, “Lea County Regional Airport Station Temperature Data (09/01/1941–06/09/2016),” includes a summary of recorded maximum and minimum temperatures at Lea County Regional Airport weather station for the period from 1941 to 2016; this location is within approximately 30 miles of the HI-STORE CIS Facility according to SAR section 2.3.3, “Onsite Meteorological Measurement Program.” Specifically, SAR table 2.3.1 lists monthly maximum and minimum as well as daily maximum and minimum temperatures. The lowest daily minimum temperature was listed as -11 degrees Fahrenheit (ºF) (-24 degrees Celsius (ºC)). The staff finds that the environmental conditions for the HI-STORE storage system at low temperature are bounded by the extreme minimum -40ºF (-40ºC) temperature associated with the design of the HI-STORM UMAX system, as reported in HI-STORM UMAX FSAR table 2.3.6, “Environmental Temperatures.”

Based on the review of the application, the staff finds that the HI-STORE CIS Facility meets the thermal regulatory requirements of 10 CFR 72.122(h)(1) and (l) and 10 CFR 72.128(a) for safe storage of spent nuclear fuel.

6.3.3 Thermal Loads and Environmental Conditions

SAR table 4.0.1 summarizes information that the applicant incorporated by reference for the HI-STORE storage system, including the spent fuel and MPC to be stored at the HI-STORE CIS Facility, MPC internal design pressures, MPC and content temperature limits, UMAX VVM temperature limits, wind and flood design conditions, and HI-STORM UMAX VVM and ISFSI design criteria. For example, the MPC, spent fuel, and HI-STORM UMAX VVM temperature limits (e.g., normal, short-term, off-normal, and accident conditions) are based on HI-STORM UMAX FSAR table 2.3.7, which the applicant incorporated by reference. SAR table 4.1.1, table 4.1.2, table 4.1.3, table 4.1.4, figure 4.1.1, and figure 4.1.2 provide the MPC-37 and MPC-89 fuel assembly decay heat limits, loading pattern, and helium backfill requirements; these also appear in technical specification table 2-1, table 2-2, table 2-3, table 2-4, figure 2-1, and figure 2-2. SAR table 4.4.1 and table 4.4.4 provide HI-TRAC CS, CTF, and HI-STAR 190 component material temperature limits. The applicant also summarized the bounding nature of the analyzed content in its RAI responses dated November 20, 2020. Finally, SAR table 6.3.1, “Thermally Significant Parameters for the HI-STORM UMAX ISFSI at HI-STORE and Corresponding Certified Value in the System FSAR,” also provided thermal loads and environmental conditions, and SAR table 4.3.2, “Environmental Data for the Licensing Basis in the HI-STORM UMAX Docket and the HI-STORE Site for Different Service Conditions,” summarized some site environmental data. SAR chapter 2 provides additional details (e.g., monthly ambient temperatures, site elevation, wind direction, and speed conditions in SAR table 2.3.2, “Lea County Regional Airport Station All Wind Data (12/01/1948–12/31/2014)”).

With regards to allowable internal MPC pressure, table 2.3.5 of the HI-STORM UMAX FSAR indicates that the normal condition design pressure is 100 pounds per square inch, gauge (psig) (689 kilopascals (kPa)), off-normal and short-term condition design pressure is 120 psig (827 kPa), and the accident condition design pressure is 200 psig (1,379 kPa). In addition, the applicant’s RAI responses dated August 16, 2021, state that the structural qualification of the MPC is also based on the analysis results from the HI-STORM FW FSAR (Holtec Report No. HI-2114830, Revision 6, dated June 18, 2019), which indicates, in section 3.4.4.1.5 and table 2.2.1, a bounding short-term normal internal pressure of 120 psig (827 kPa). Holtec Report No. HI-2177597, “HI-STORE CTF Thermal Evaluation,” Revision 2, dated August 13, 2021, states that MPC pressure calculations were based on using the MPC’s maximum backfill pressure. SAR section 6.4.3.2, “MPC Cavity Pressures,” and table 6.4.4, “MPC Cavity Pressure During Normal Long-Term Storage in HI-STORM UMAX VVM,” indicate that the MPC pressures associated with normal condition (assuming 1 percent rod rupture with 100 percent fill gleas and 30 percent fission gas released from the ruptured rods), off-normal condition (assuming 10 percent rod rupture with 100 percent fill gas and 30 percent fission gas released from the ruptured rods), and accident condition (assuming 100 percent rod rupture with 100 percent fill gas and 30 percent fission gas released from the ruptured rods) are 89.2 psig (615 kPa), 98.3 psig (678 kPa), and 188.7 psig (1,302 kPa), respectively, which are below the corresponding allowable pressures mentioned above. SAR section 6.4.3.2 indicates that the calculations did not credit the potential for increased heat transfer due to increased helium density and additional internal convective thermosiphon flows associated with the 100 percent rod rupture.

SAR table 2.7.1, “Site Specific Data for Thermal and Structural Analysis,” and section 6.3, “Thermal Loads and Environmental Conditions,” indicate that the normal condition storage thermal analyses assume an average ambient temperature of 62ºF (17ºC). However, SAR table 2.3.1 indicates the site’s average monthly maximum temperature for June, July, and August is approximately 92ºF (33ºC); this temperature would more accurately represent a maximum normal ambient temperature, considering that, for part of the year, the storage system would have to transfer the content’s decay heat at ambient temperatures greater than 62ºF (17ºC). The staff considered the model and results described in section 6.1.1 of the calculation package, Holtec Report No. HI-2177591, Revision 2, “Thermal Evaluations of HI-STORM UMAX at HISTORE CIS Facility,” September 27, 2021, which dealt with a steady-state thermal evaluation at a high ambient temperature of 94ºF (34ºC), which approximates the 92ºF (33ºC) normal ambient temperature discussed above. The staff review of output files from this analysis indicates a cladding temperature of 647ºF (342ºC), which is 34ºF higher than the 613ºF (342ºC, which is 19ºC higher than the 323ºC) peak cladding temperature (PCT) reported in SAR table 6.4.3, “Normal Long-Term Storage Temperatures for MPC-37 in HI-STORM UMAX at HI-STORE CIS,” for the normal condition analysis that assumes a 62ºF ambient temperature. These results show that a 32ºF increase in ambient temperature results in a 34ºF (18ºC results in a 19ºC) increase in PCT, which is approximately a 1:1.1 ratio. The staff finds that the 613ºF (323ºC) and 647ºF (342ºC) PCT values are less than the 752ºF (400ºC) allowable limit with PCT margins over 100ºF (56ºC).

SAR table 4.3.2 and table 6.3.1 indicate the off-normal storage condition maximum temperature criterion of 91ºF (33ºC), defined in SAR table 4.3.2 as the highest 72-hour average ambient temperature. The applicant provided the site-specific calculation of the highest 72-hour average temperature of 90.7ºF in SAR section 2.3.1, denoted as 91ºF in SAR table 6.3.1, which points to SAR table 2.7.1. The site’s short-term operations maximum temperature criterion is defined as the maximum 3-day average temperature, which the applicant reported as 91ºF for the CIS Facility in SAR table 4.3.2 and table 2.7.1. The staff notes that this is similar to the 3-day average temperature by comparing the temperatures and temperature designations in SAR table 4.3.2 and table 2.3.6 of HI-STORM UMAX FSAR. SAR table 2.3.1 shows the average monthly maximum temperature during June, July, and August near the site is approximately 92ºF (33ºC). Although the 91ºF site short-term operations maximum temperature is slightly greater than the 90ºF UMAX 3-day maximum average short-term temperature reported in HI-STORE UMAX FSAR table 2.3.6, the staff notes that the applicant’s short-term operations thermal analysis, discussed in SER section 6.3.4.2, indicates that an 83ºF (46ºC) cladding temperature margin and 29ºF (16ºC) concrete temperature margin are sufficient to show a 1ºF difference has no bearing on performance. In addition, SAR table 4.3.2 indicates the accident condition criterion temperature of 108ºF (42ºC) is defined as the maximum average ambient temperature over a 24-hour period; this temperature is reported as the site’s extreme maximum temperature in SAR table 2.3.1. The staff notes that the HI-STORE thermal results, margins, and steady-state evaluations associated with ambient temperatures higher than the site’s 62ºF (16.7ºC) annual average ambient temperature (e.g., 94ºF (34.4ºC) steady high ambient, 91ºF (32.8ºC) short-term operation, and 108ºF (42ºC) extreme ambient accident temperature) were based on applying these temperatures as constant, steady-state value boundary conditions (i.e., constant peak value rather than time-varying diurnal). The staff finds that the environmental conditions for the HI-STORE UMAX storage system at high off-normal and accident temperatures are bounded respectively by the 100ºF (38ºC) and 125ºF (52ºC) temperatures associated with the design of the HI-STORM UMAX system, as reported in SAR section 6.5.1, table 4.3.2, and HI-STORM UMAX FSAR table 2.3.6.

Other environmental considerations are discussed in SAR section 6.4.3.5; the applicant's RAI responses dated January 31, 2019 (proprietary, Attachment 1); the applicant's RAI responses dated August 16, 2021 (proprietary, Attachment 1); and Holtec Report No. HI-2177591, which discusses the impact of ambient wind on a HI-STORE UMAX VVM and the impact of an array of loaded HI-STORE UMAX VVMs on system temperature. SAR section 2.3.1 indicates that winds at the site are moderate and blow 84 percent of the time. The licensee analyzed the impact of different wind speeds on a VVM's inlet and outlet vent openings using a half-symmetric model of the HI-STORM UMAX Version C, which is based on the HI-STORM UMAX model described in the HI-STORM UMAX FSAR. SAR section 6.4.3.5 and table 6.4.7 show that the HI-STORE UMAX system PCT increased due to reduced air flow entering the VVM annulus for a range of wind speeds but that the increased temperature was bounded by the temperatures for the HI-STORM UMAX system set forth in Chapter 4 of the HI-STORM UMAX FSAR.

In addition, SAR section 6.4.3.5 and Holtec Report No. HI-2177591, state that the impact of wind on 1 x 12 array and 1 x 25 array (to account for effects at the end of the array) FLUENT models (based similarly on a previous 1 x 8 UMAX array) of loaded UMAX VVMs, such as due to preheating the inlet air from a neighboring outlet vent's heated air, was analyzed using the half-symmetric array from HI-STORM UMAX FSAR section 4.4.9, which the applicant incorporated by reference in SAR table 6.0.1. The RAI response dated August 16, 2021, indicates a 42.35-kilowatt (kW) VVM decay heat load for this model, which is nearly 10 kW greater than the loaded VVM at the HI-STORE CIS Facility. The model considered the effects of neighboring transverse arrays by employing reflecting lateral wall symmetric boundary conditions along the model's half-pitch planes. This boundary condition prevents the lateral dissipation of heated air from leaving the system and, in addition, redirects radiation heat transfer back into the model's domain. Model results show that wind and array effects can increase inlet air temperature by more than 10ºF (5.5ºC), compared to the ambient air temperature.

SAR table 6.4.8 indicates that the combined impact of wind and array effects could increase peak cladding temperatures by nearly 35ºF (19ºC). It was noted above that the 94ºF (34ºC) ambient temperature model results in a PCT of 647ºF (342ºC). Therefore, adding 35ºF due to the impact of wind and array effects to the 647ºF cladding temperature shows that PCT would continue to be below the cladding's normal condition 752ºF (400ºC) allowable temperature with margin of over 65ºF (36ºC).

Finally, Holtec Report No. HI-2177591, notes that insolation values (according to 10 CFR 71.71(c)) were applied to the thermal model surfaces exposed to ambient conditions. The staff finds that the insolation boundary conditions are reasonable because the 10 CFR 71.71(c) insolation values have a conservative basis and were used in the previously reviewed HI-STORM UMAX subterranean system.

Based on the above, the staff has reasonable assurance that the design parameters of the proposed storage systems bound the thermal loads and environmental conditions of the HI-STORE CIS Facility site and therefore meet the requirements of 10 CFR 72.92(a) and 10 CFR 72.122(b).

6.3.4 Analytical Methods, Models, and Calculations

SAR chapter 6 discusses the analytical methods, models, thermal material properties, and calculations associated with the HI-STORE system thermal analyses; additional details are also provided in Holtec Report No. HI-2177591; Holtec Report No. HI-2177597, Revision 2; and Holtec Report No. HI-2177553, Revision 3, “Thermal Analysis of HI-TRAC CS Transfer Cask,” dated August 13, 2021. SAR table 6.0.1 indicates that numerous aspects of the HI-STORE thermal analysis, including thermal properties, MPC-37 and MPC-89 thermal model and methodology, HI-STORM UMAX VVM thermal model and methodology, minimum temperatures, engineered clearances, evaluation of sustained wind, off-normal environment temperature, extreme environment temperature, and flood, are based on that reported in the HI-STORM UMAX FSAR, and the applicant incorporated by reference those aspects discussed in the HI-STORM UMAX FSAR. Likewise, as noted in SAR section 6.4.1, “Applicable Systems,” and section 6.4.2, “Analysis Methodology,” the methods and models used are similarly described in the previously certified HI-STORM UMAX FSAR. SAR section 6.4.2.1, “Computer Code,” states that the thermal analyses used the FLUENT computational fluid dynamics code and that the FLUENT solutions reported in the SAR follow the numerical stability and grid sensitivity criteria (e.g., convergence parameters, under-relaxation factors, numerical residuals and behavior) according to the HI-STORM UMAX FSAR. The text below summarizes the thermal analyses for the new systems or aspects associated with the HI-STORE CIS Facility and not previously evaluated in the HI-STORM UMAX FSAR. Finally, the SAR glossary defines the section average temperature (i.e., a thermal-related term) as the lineal average temperature through the thickness of a component; the staff notes that this is calculated by taking the average of the temperatures through a component’s cross-section.

6.3.4.1 HI-STORE System

As noted above, SAR table 6.0.1 indicates that the MPC and HI-STORM UMAX VVM thermal models and methodology are incorporated by reference. In addition, SAR sections 6.4.1 and 6.4.2.3 indicate that the thermal analyses reported in the previously certified HI-STORM UMAX FSAR are bounding because of their higher decay heat and generally higher ambient temperature compared to the HI-STORE system. Nonetheless, a HI-STORE thermal analysis is discussed in SAR sections 6.4.1, 6.4.2, and 6.4.3 to describe the impact of the HI-STORM UMAX Version C Type SL with a decay heat defined by HI-STORE CIS Facility technical specification table 2-1 for Pattern 1 (approximately 32.09 kW) at an ambient temperature of 62ºF (17ºC); additional model details appear in Holtec Report No. HI-2177591. SAR section 6.4.2.2, “MPC Thermal Model,” states that the MPC thermal analysis model is incorporated by reference from section 4.4 of the HI-STORM UMAX FSAR. Likewise, SAR section 6.4.2.3, “HI-STORM UMAX VVM Thermal Model,” states that the thermal modeling of the HI-STORM UMAX VVM is incorporated by reference from section 4.4 of the HI-STORM UMAX FSAR.

The results of the analyses appearing in SAR table 6.4.4 indicate that the 88.2 psig (608 kPa) MPC pressure, based on no rod rupture (MPC pressure increased to 89.2 psig (615 kPa), assuming 1 percent rod rupture) was below the 100 psig (689 kPa) design pressure reported in table 2.3.5 of the HI-STORM UMAX FSAR, which the applicant incorporated by reference in SAR table 4.3.1, “Loadings Excluded from Further Consideration in the Qualification of Storage System and Ancillaries at the HI-STORE SAR,” and that temperatures are below the design allowable values reported in HI-STORM UMAX FSAR table 2.3.7. For example, SAR table 6.4.3 shows that the 613ºF (323ºC) fuel cladding temperature is well below the 752ºF (400ºC) allowable value. In addition, results show that the difference between the average air outlet vent temperature of 153ºF (67ºC) and the 62ºF (17ºC) ambient temperature boundary condition is 91ºF (50ºC); the staff finds that this supports the 91ºF (50ºC) air temperature increase setpoint specified in HI-STORE CIS Facility technical specification surveillance requirement (SR) 3.1.2. Based on the above, the staff finds that the HI-STORE CIS Facility storage system thermal analysis demonstrates that the HI-STORE CIS Facility storage system can passively transfer the bounding analyzed content decay heat such that ITS component temperatures are below allowable values and MPC pressures are below design limits.

In addition, SAR section 6.5.1, “Off-Normal Events,” discusses the condition of 50 percent blockage of the UMAX system’s inlet and outlet vent screens. According to section 6.7.1 of Holtec Report No. HI-2177591, except for the partial blockage of the inlet and outlet opening, the thermal model used in this partial blockage steady-state analysis is the same as the long-term, steady-state storage condition in SER section 6.3.4. The computational run was continued until the heat balance and mass balance through the domain were approximately 100 percent and the PCT, mass-flow rate, and numerical residuals were stabilized. According to SAR table 6.5.5, “Maximum Temperatures for MPC-37 in HI-STORM UMAX at HI-STORE CIS During 50% Inlet and 50% Outlet Vent Blockage,” the results indicate that the 626ºF (330ºC) fuel cladding temperature and the UMAX system component temperatures had over 125ºF (69ºC) margin with normal-condition allowable temperatures. Likewise, the MPC pressure of 92.9 psig (640 kPa) is below the 100 psig (689 kPa) normal condition design pressure. The staff finds that these results support the technical specification limiting condition for operation (LCO) 3.1.1, which states that the HI-STORE system is in an operable condition when 50 percent or more of inlet and outlet vent areas are unblocked and available for flow.

6.3.4.2 HI-STORE HI-TRAC CS Transfer Cask

According to SAR section 3.1.4, “Operations at HI-STORE,” and SAR section 6.1, the HI-TRAC CS is the dry use transfer cask that is used to transfer the MPC to the HI-STORE storage system after its receipt in the HI-STORE CIS Facility CTF. As noted in SER section 6.3.4, the MPC model within the HI-TRAC CS and the subsequent model’s solution convergence (i.e., stability, grid sensitivity) is based on the previously reviewed HI-STORM UMAX model and thermal analyses. Specifically, the HI-TRAC CS thermal analyses are based on the most limiting MPC-37 canister (pressurized-water reactor 14x14 short fuel) with heat load pattern 1 described in HI-STORE CIS Facility technical specification table 2-1. SAR section 6.4.2.5, “HI-TRAC CS Transfer Cask Thermal Model,” describes the HI-TRAC CS transfer cask thermal three-dimensional quarter-symmetric model, which is based on Holtec Drawing No. 10868, “HI-TRAC CS Licensing Drawing,” Revision 1 (proprietary), cited in Holtec Report No. HI-2177553, Revision 3, dated March 18, 2022. The SAR notes a number of its important features, [

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Likewise, the calculation package indicates that the MPC's axial decay heat profile was adopted from the HI-STAR 190 Package SAR (Holtec Report No. HI-2146214, Revision 0.D, dated January 25, 2017).

SAR section 6.4.3.7, "Evaluation of Normal Onsite Transfer in HI-TRAC CS," and SAR table 6.4.6, "Normal On-Site Transfer Temperatures and MPC Cavity Pressure in HI-TRAC CS," provide the results from the steady-state analysis and show the fuel cladding temperature of 669ºF (354ºC) is below the 752ºF (400ºC) allowable short-term temperature limit for high-burnup fuel. The staff notes that the 1,058ºF (570ºC) short-term cladding temperature limit (reported in SAR chapter 6; Holtec Report No. HI-2177591; and Holtec Report No. HI-2177597, Revision 2) for an MPC containing all low-burnup fuel is dependent on the cladding hoop stress being less than or equal to 90 MPa (13,053 psig). The temperature margin associated with high-burnup fuel cladding is 83ºF (46ºC) and the margin associated with the HI-TRAC CS concrete temperature is 29ºF (16ºC). The staff finds the 83ºF (46ºC) cladding temperature margin and 29ºF (16ºC) concrete temperature margin indicate that diurnal conditions with a daily maximum temperature higher than 91ºF (33ºC) and less than the 108ºF (42ºC) site extreme maximum temperature (reported in SAR table 2.3.1) would continue to result in cladding and concrete temperatures below allowable values. The results also show the MPC pressure is 96 psig (662 kPa), which is less than the 120 psig (827 kPa) pressure limit indicated in table A.7.2 of Holtec Report No. HI-2177553, Revision 3. Likewise, thermal expansion calculations are presented in Holtec Report No. HI-2177553, Revision 3. The results show that there are no interferences between the fuel basket-to-MPC radial gap, fuel-basket-to-MPC axial gap, MPC-to-HI-TRAC CS radial gap, and MPC-to-HI-TRAC CS axial gap. Based on the above, the staff finds that the HI-STORE HI-TRAC CS transfer cask thermal analysis demonstrates that the HI-TRAC CS transfer cask can transfer the bounding analyzed content such that ITS component temperatures are below allowable values and MPC pressures are below design limits.

6.3.4.3 HI-STAR 190 SL Transportation Cask within the Canister Transfer Facility

The CTF, which is located within the CTB, is used when transferring the MPC from the HI-STAR 190 SL transportation cask to the HI-TRAC CS transfer cask. The applicant described the thermal analysis associated with this operation in SAR section 6.4.2.4, “HI-STAR 190 Thermal Model,” and Holtec Report No. HI-2177597, Revision 2. As noted above in SER section 6.3.4.2, the MPC model and the CTF model’s solution convergence (i.e., stability, grid sensitivity) is based on the previously reviewed HI-STORM UMAX model and thermal analyses. Specifically, for this steady-state analysis, the quarter symmetric three-dimensional thermal model included the vertical-oriented HI-STAR 190 SL transportation cask (which included the containment shell, Holtite, enclosure shell, closure lid, containment bottom and top forgings) placed inside the CTF cavity. SAR section 6.4.2.4 mentions that the limiting operating scenario is when the annulus between the MPC and HI-STAR 190 cask is backfilled with nitrogen because it is the least conductive gas. The staff finds that of the three backfilled-gas scenarios, the scenario with nitrogen in the annulus would be limiting because nitrogen’s thermal conductivity is lower than air and helium. According to SAR section 6.4.2.4, the flow in the annulus space between the HI-STAR 190 transportation cask and the CTF was modeled as turbulent using the k-ω model with the transitional option enabled, flow within the MPC was modeled as laminar, the nitrogen within the HI-STAR 190 was conservatively modeled by not including convective flow (i.e., stationary), and the ambient temperature was 91ºF (33ºC). The HI-STAR 190 impact limiters are removed before being placed within the CTF and, therefore, were not modeled. However, the lid remained in place during the steady-state thermal analysis. The HI-STAR 190 SL rests on a pedestal placed within the CTF, as described in HI-STORE Drawing No. 10895, “Canister Transfer Facility (CTF)” (proprietary, see SAR section 1.5). In addition, in RAI responses dated October 9, 2020, the applicant indicated that the ribbed pedestal (specifications in the RAI response included a volumetric ratio of ribs to the air inside the pedestal greater than 5%), which is fabricated from carbon steel, is an ITS component that ensures adequate heat dissipation from the HI-STAR 190 SL. Further, the applicant’s RAI responses dated November 20, 2020, indicate the thermal model conservatively underpredicted the pedestal’s thermal conductivity. The MPC content includes the bounding pressurized-water reactor short fuel content, as described earlier using the heat load pattern 1 provided in technical specification 2.1 and SAR table 4.1.1. According to SAR section 6.4.2.4 and Holtec Report No. HI-2177597, Revision 2, there are conservative aspects of the thermal model, including the use of high friction loss assumptions for the CTF inlets (e.g., understated areas). SAR table 6.4.5, “Maximum Component Temperatures and MPC Cavity Pressure for HI-STAR 190 in CTF Short-Term Operation,” indicates a cladding temperature of 716ºF, which is 36ºF below the 752ºF (380ºC, which is 20ºC below 400ºC) normal-condition allowable limit. In addition, ITS component temperatures associated with the MPC and the HI-STAR 190 transportation package are below allowable values. The MPC pressure of 102.3 psig (705 kPa) is below the 120 psig (827 kPa) short-term limit described above in SER section 6.3.3. Based on the above, the staff finds that the HI-STAR 190 SL transportation package and CTF thermal analysis demonstrate that the content, package, and MPC ITS component temperatures are below allowable values and MPC pressures are below design limits.

Based on the discussion above, the staff has reasonable assurance that the HI-STORE CIS Facility is adequately designed to protect ITS SSCs from all postulated normal, off-normal, and accident thermal loads and environmental conditions and, therefore, meets the requirements of 10 CFR 72.122(h)(1) and 10 CFR 72.128(a).

6.3.4.4 Off-Normal and Accident Conditions

SAR chapter 15 and section 6.5.2, "Accident Events," lists thermal-related analyses for off-normal and accident scenarios, including the HI-STORM UMAX fire accident, HI-TRAC CS fire accident, HI-STAR 190 fire accident, off-normal environmental temperature, extreme environmental temperature, 100 percent blockage of HI-STORM UMAX inlet and outlet air vents, partial blockage of MPC basket vent holes, and 100 percent rod rupture accident coincident with accident events. In addition, SAR section 6.4.3.8, "Evaluation of HI-TRAC CS to HI-STORM UMAX Transfer," presents a thermal analysis for the MPC transfer condition of an empty HI-TRAC CS not being able to be removed away from the loaded UMAX storage system. The staff notes that the preceding thermal SER sections discussed "partial vent blockage condition" and "hypothetical non-quiescent wind condition" thermal analyses.

Regarding the HI-STORM UMAX fire accident, SAR section 6.5.2.1, "Bounding Fire Event," states that this accident condition was bounded by an engulfing fire thermal analysis performed for the HI-STORM FW system ("Evaluation of Effects of Tracked VCT Fire on HI-STORM FW System," Holtec Report No. HI-2135677) with the same quantity of combustibles (430 gallons) (1,628 liters) cited in SAR table 6.5.1 and proposed Technical Specification 4.2.5.2. According to SAR section 6.5.2.1, results of the HI-STORM FW model indicate that both fuel and MPC confinement integrity are maintained. Therefore, the applicant states that the contents and MPC integrity would be maintained for an engulfing HI-STORM UMAX system fire accident at the HI-STORE CIS Facility. The SAR identifies three reasons for concluding that the MPC and its contents are safe in the HI-STORM UMAX system at the HI-STORE CIS Facility: (1) the HI-STORM UMAX system's initial PCT and MPC component temperatures are lower than for the HI-STORM FW system, (2) the HI-STORM UMAX decay heat is lower than for the HI-STORM FW system, and (3) the HI-STORM UMAX underground system has much less surface exposed to fire than the aboveground HI-STORM FW system. The staff finds that, between the HI-STORM FW system and the HI-STORM UMAX system with a given fire source, a system such as the HI-STORM UMAX system, having lower initial temperatures, less decay heat, and less exposed surface area, would result in lower temperatures. Based on the above, the staff concludes that the MPC and its contents remain below temperature and pressure allowable values during the fire accident scenario.

SAR section 6.5.2.1(b), "HI-TRAC CS Fire Accident," also discusses the HI-TRAC CS fire accident condition in which the transfer cask with the bounding MPC-37 located within the CTB is engulfed within a fire from the VCT. SAR section 6.5.2.1 describes the details associated with a HI-TRAC CS fire accident condition thermal analysis; additional details are presented in Holtec Report No. HI-2177553, Revision 3. According to section A.2 of Holtec Report No. HI-2177553, the thermal analysis principles and methodology are based on those previously described in the HI-STORM FW FSAR (Holtec Report No. HI-2114830, Revision 4) and HI-STORM UMAX FSAR (Holtec Report No. HI-2115090, Revision 3). The applicant used the FLUENT Version 14.5 code to perform the HI-TRAC CS thermal modeling. The transient HI-TRAC CS fire accident thermal analysis assumed 430 gallons (1,628 liters) of combustible fluid from transport vehicles (e.g., the VCT), combustion of solid tires begins after the combustible fluids are consumed, flame emissivity of 1, HI-TRAC CS surface emissivity of 0.85, average flame temperature of 1,475ºF (802ºC) applied to all exposed HI-TRAC CS surfaces, maximizing the fire duration by using a lower bound fuel consumption rate and by assuming the fuel source extends at 1 meters (m) (but not more than 3 m) beyond the external surface of the cask, and a forced convection fire heat transfer coefficient of 4.5 Btu/(hr ft^2 ºF). The transient thermal analysis was based on applying the above-mentioned boundary conditions to a FLUENT model of the HI-TRAC CS. The transient period included the fire duration and the postfire period for the transfer cask temperatures to reach maximum values and then recede. During the postfire period, a natural convection correlation was used to determine the surface's heat transfer coefficient. The analysis results appearing in SAR table 6.5.2, "HI-TRAC CS Fire and Post-Fire Accident Results," indicate that the maximum fuel cladding temperature of 701ºF (372ºC) was well below the 1,058ºF (570ºC) accident-condition allowable temperature and that the MPC pressure of 100.2 psig (691 kPa) also was well below the 200 psig (1,379 kPa) accident design pressure. As mentioned in note 2 of SAR Table 6.5.2, although "an extremely small area" of the HI-TRAC CS concrete exceeds temperature limits, the RAI response dated August 16, 2021, indicates that the shielding analysis accounts for reduced shielding performance by the affected concrete.

Likewise, SAR section 6.5.2.1(b.2), "Combined VCT and HI-PORT Fire Accident," discusses the thermal analysis related to the combined effects of a VCT and HI-PORT fire accident, stating that the analysis adopted the same methodology and thermal model as the preceding VCT fire thermal analysis. The combined VCT and HI-PORT fire accident duration was longer due to the increased liquid combustibles (528 gallons) (1,999 liters, as specified in proposed Technical Specification 4.2.5.5) and the solid combustibles of rubber tires (4,479 pounds) (2,032 kg), in which an equivalent volume of liquid as the heating content of the solid combustibles was added to the liquid combustibles to maximize the engulfing nature of a liquid fire. The analysis results appearing in SAR table 6.5.4, "HI-TRAC CS Combined VCT and HI-PORT Fire and Post-Fire Accident Results," indicate that the maximum fuel cladding temperature of 730ºF (388ºC) was well below the 1,058ºF (570ºC) accident-condition allowable temperature, and that the MPC pressure of 103.9 psig (716 kPa) also was well below the 200 psig (1,379 kPa) accident design pressure. The staff discusses the effect of the above-mentioned fires on the concrete in SER chapter 7, regarding shielding, and SER chapter 17, regarding materials.

Based on the above, the staff finds that the HI-STORE HI-TRAC CS transfer cask thermal analysis demonstrates that the MPC and its content remain below accident design criteria during the fire accident scenario.

Regarding a HI-STAR 190 fire accident condition, SAR section 6.5.2.1(c), "HI-STAR 190 Fire Accident," indicates that the loading and lifting operations occur inside the CTB using a crane and that there are no combustibles during these operations. Nonetheless, the SAR (section 15.3.1.1, "Fire Analysis," item (c), "HI-STAR 190 Fire") incorporated by reference the 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," 30-minute fire hypothetical accident condition described in section 3.4 of the HI-STAR 190 SAR (Holtec Report No. HI-2146214, Revision 3, dated November 2, 2018) to demonstrate that the HI-STAR 190 package and its contents can survive a long-duration fire event. The staff notes that the 30-minute fire duration is nearly twice as long as the fire duration calculated for HI-TRAC CS loaded with an MPC. The staff finds that the 30-minute fire analyzed in the HI-STAR 190 SAR

(Holtec Report No. 2146214, Revision 3) would be a bounding analysis, considering that the loading and lifting operations do not include combustible material, and therefore, the staff concludes that the MPC and its contents remain below temperature and pressure allowable values during the fire accident scenario.

SAR section 6.5.2.3, “Burial Under Debris,” indicates that there are no structures over the HI-STORE system at the HI-STORE CIS Facility that could cause a loaded UMAX VVM to be buried by debris, and therefore, burial under debris is not credible. However, SAR section 6.4.3.8, “Evaluation of HI-TRAC CS to HI-STORM UMAX Transfer,” analyzes the condition for determining the time that the HI-TRAC CS must be removed from atop the UMAX VVM system after lowering the loaded MPC within the VVM by assuming a bounding burial under debris event analyzed in section 4.6.2.4 of the HI-STORM UMAX FSAR that relied on a thermal mass capacity equation. SAR section 6.4.3.8 also mentions that an alternative evaluation could be used based on a transient analysis of the computational fluid dynamics (CFD) models and methodologies described in chapter 6 of the SAR. The analysis in SAR section 6.4.3.8 found that the cladding temperature could reach its normal-condition allowable temperature if the HI-TRAC CS were not removed from the UMAX VVM within 6 hours, after which the situation would be considered an accident condition such that the remedial time to remove the HI-TRAC CS would be based on the 100 percent vent blockage event described below. The staff notes that this analysis was referenced in the caution box found in SAR section 10.3.3.5. The staff finds that the equation in SAR section 6.4.3.8 or use of the CFD models and methodologies described in SAR chapter 6 to determine relevant time limits for a burial under debris analysis are adequate, as these methods have been approved in earlier HI-STORM UMAX system evaluations.

SAR section 6.5.2.5, “100% Blockage of Air Vents,” and section 6.7.2 of Holtec Report No. HI-2177591, discuss the accident condition of complete blockage of inlet and outlet ducts, in which the thermal model described earlier for long-term steady-state storage conditions was modified by making the inlet and outlet ducts impervious to air flow. The model was run as a transient analysis for 32 hours with the initial condition based on the temperatures calculated during the steady-state normal storage condition. According to SAR table 6.5.6, “Maximum Temperatures for MPC-37 in HI-STORM UMAX at HI-STORE CIS During 32-hour 100% Inlet and 100% Outlet Vent Blockage,” the fuel cladding temperature had a margin greater than 248ºF (138ºC) from the allowable accident temperature (i.e., 810ºF (432ºC) compared to the limit of 1,058ºF (570ºC) from Interim Staff Guidance (ISG)-11, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” Revision 3, dated November 17, 2003. In addition, the metal components of the UMAX MPC system had temperatures with margins of at least approximately 300ºF (167ºC from the allowable accident temperatures (i.e., 759ºF compared to 1,058ºF (404ºC compared to 570ºC)) and the closure lid concrete temperature had a margin over 125ºF (70ºC) from the allowable accident temperature. Likewise, the MPC pressure of 116.2 psig (801 kPa) was well below the 200 psig (1,379 kPa) allowable accident pressure. Based on these results, the staff finds that there is reasonable assurance for the premise of the 32-hour time-for-action blockage accident period associated with technical specification LCO 3.1.1, especially considering the site topography, which is not in proximity to mountains or prone to landslides, and limited precipitation totals (i.e., potential for snow) described in SAR chapter 2.

SAR section 6.5.1 incorporated by reference section 4.6.1 of the HI-STORM UMAX FSAR (Holtec Report No. HI-2115090, Revision 3), which presents the off-normal results of a thermal analysis based on steady-state conditions (i.e., not time-varying diurnal) of an ambient temperature of 100ºF (38ºC), in accordance with HI-STORM UMAX FSAR table 2.3.6. Specifically, HI-STORM UMAX FSAR table 4.6.5 indicates an MPC pressure of 96.2 psig (663.3 kPa) (which is less than the 100 psig normal condition design pressure). Likewise, HI-STORM UMAX FSAR table 4.6.1 indicates a PCT of 713ºF, which is 39ºF less than the 752ºF (378ºC, which is 22ºC less than 400ºC) normal-condition allowable temperature. The pressure and temperatures would tend to be less for the HI-STORE CIS Facility system because, as noted in SAR table 6.3.1, the HI-STORE CIS Facility aggregate heat load of 32.09 kW is less than the 37.06 kW aggregate heat load reported in the HI-STORM UMAX FSAR.

SAR section 6.5.2.4, “Extreme Environmental Temperature,” considers the extreme environmental temperature accident condition analyzed in HI-STORM UMAX FSAR section 4.6.2.2, which indicates that a steady-state thermal analysis of a 125ºF (52ºC) constant daily temperature (according to HI-STORM UMAX FSAR table 2.3.6) results in an MPC pressure of 99 psig (according to HI-STORM UMAX FSAR table 6.4.10) and a 738ºF (392ºC) PCT, which is below the 752ºF (400ºC) normal-condition allowable temperature. SAR section 6.5.2.4 states that the situation at the HI-STORE CIS Facility is bounded by the analysis described in section 4.6.2.2 of the HI-STORM UMAX FSAR for a number of reasons. First, the SAR notes that not only are the PCT and MPC component temperatures for the HI-STORE system at normal conditions lower than the normal condition PCT temperature and MPC component temperatures reported in the HI-STORM UMAX FSAR, but also the extreme environmental temperature (108ºF, 42ºC) at the HI-STORE CIS site is lower than the extreme environmental temperature (125ºF, 52ºC) described in the HI-STORM UMAX FSAR. The staff agrees and finds that the results described in the HI-STORM UMAX FSAR for the extreme environmental temperature accident condition are bounding for the HI-STORE CIS site because the decay heat, the initial component temperatures, and the extreme environmental temperature at the HI-STORE CIS site are lower than those defined in the HI-STORM UMAX FSAR.

According to SAR section 15.3.2, the analysis associated with the partial blockage of MPC basket vent holes described in UMAX FSAR section 12.2.2 is incorporated by reference. Likewise, HI-STORM UMAX FSAR section 12.2.2 incorporates by reference the partial blockage of the MPC basket vent hole analysis in HI-STORM FW FSAR section 12.2.5. The staff has previously accepted the HI-STORM FW FSAR description of the accident as not being credible.

SAR section 15.3.20 comments on an accident condition with 100 percent fuel rupture, including postulating that NUREG-1536, “Standard Review Plan for Spent Fuel Dry Cask Storage Systems at a General License Facility,” Revision 1, issued July 2010, describes 100 percent fuel rupture as nonmechanistic, and therefore, it would be analyzed as a stand-alone event. However, HI-STORM UMAX FSAR table 2.6.5 states that the MPC vessel is designed to withstand maximum internal pressures considering 100 percent fuel rod failure and maximum accident temperatures; the staff finds this is consistent with NUREG-1536, section 2.5.2.2(3)(d).

SAR section 6.5.2.2, “Explosion Event,” states that, although there is no credible explosive event at the HI-STORE CIS Facility, a design-basis external pressure event for the MPC was evaluated in chapter 3 of the HI-STORM FW FSAR. SER section 15.3.2.8 further discusses external pressure.

SAR section 6.5.2.6, “Flood,” states that the flood accident scenario at the HI-STORE CIS Facility is bounded by the analysis in the HI-STORM UMAX FSAR, and, therefore, the applicant incorporated by reference section 4.6.2.5 of the HI-STORM UMAX FSAR as noted in SAR section 6.5.2.6 and SAR table 6.0.1. SER section 15.3.2.4 further discusses flooding.

Based on the discussion above, the staff has reasonable assurance that the HI-STORE CIS Facility is adequately designed to protect SSCs ITS from the postulated off-normal and accident thermal loads and environmental conditions and, therefore, meets the requirements of 10 CFR 72.122(h)(1) and 10 CFR 72.128(a).

6.3.4.5 Thermal Stresses

Multiple documents describe thermal stress calculations associated with the HI-STORM UMAX at the HI-STORE CIS Facility, the content and MPC within the HI-STAR 190 transportation package inside the CTF; and the content and MPC within the HI-TRAC CS transfer cask. These descriptions are included in SAR section 6.4.3.4; section 4.4.6 of HI-STORM UMAX FSAR, incorporated by reference; Holtec Report No. HI-2177591; Holtec Report No. HI-2177597, Revision 2; and Holtec Report No. HI-2177553, Revision 3.

Holtec Report No. HI-2177591, discusses differential thermal expansion calculations during normal, off-normal, and accident conditions for fuel basket-to-MPC radial growth, fuel basket-to-MPC axial growth, MPC-to-divider shell radial growth, and MPC-to-VVM closure lid axial growth; the results showed restraint-free thermal expansion. Section 6.3 and table 6.10 of this Holtec report also indicate that the calculations were based on conservative assumptions and thermal expansion coefficients, which still resulted in thermal expansion margins and restraint-free thermal expansion. Holtec Report No. HI-2177597, Revision 2, discusses HI-STAR 190 differential thermal expansion calculations for fuel basket-to-MPC radial growth, fuel basket-to-MPC axial growth, fuel-to-MPC axial growth, MPC-to-Cask radial growth, MPC-to-cask axial growth, and cask-to-CTF pedestal radial growth; the results showed restraint-free thermal expansion. Holtec Report No. HI-2177553, Revision 3, discusses differential thermal expansion calculations for fuel basket-to-MPC radial growth, fuel basket-to-MPC axial growth, MPC-to-cask radial growth, and MPC-to-cask axial growth; the results showed restraint-free thermal expansion. The staff finds thermal stresses would be limited based on restraint-free thermal expansions. SER section 5.3.1.1 further discusses thermal stresses involving the MPCs.

6.3.4.6 Thermal-Related Technical Specifications

The applicant’s proposed technical specification LCO 3.1.1 describes the operability and surveillance requirements of the spent fuel storage cask heat removal system. LCO 3.1.1 and Surveillance Requirement (SR) 3.1.2 mention that the loaded HI-STORE spent fuel storage cask heat removal system is operable when 50 percent or more of the inlet and outlet vent areas are unblocked and available for flow or when air temperature requirements are met when temperature monitoring equipment is installed. LCO 3.1.1 also indicates that blockage to the inlet and outlet vents would be removed within 8 hours and, if unable to be completed within that time frame, within another 24 hours either the blockage would be removed or the MPC transferred into a transfer cask. Likewise, surveillance requirement bases found in section SR 3.1.2 of SAR chapter 16 indicate that corrective actions are to be taken promptly to remove vent blockages (including slight obstructions) and restore full flow. The staff finds a spent fuel storage cask heat removal system having 50 percent or more of the inlet and outlet vent areas unblocked and available for flow can be considered operable based on information in SAR section 6.5.1 and section 6.7.1 of Holtec Report No. HI-2177591. These sources describe a 50 percent partial blockage inlet vent and outlet vent CFD model (mentioned above in SER section 6.3.4.1) having steady-state results demonstrating that ITS component temperatures would remain below their normal-condition allowable values, and MPC pressures (89.7 psig (618 kPa)) would remain below the 100 psig (689 kPa) normal-condition allowable pressure.

According to technical specification SR 3.1.2 for VVMs with installed temperature monitoring equipment, air temperature requirements are met when the difference between the average VVM air outlet duct temperature and the ISFSI ambient temperature is 91ºF (50ºC) or less. The staff finds that the 91ºF air temperature difference requirement is reasonable because SAR table 6.4.3 shows that the difference between the normal long-term storage average air outlet temperature (153ºF, 67ºC) and the 62ºF (17ºC) ambient temperature boundary condition is 91ºF (50ºC). In addition, the staff notes that SAR section 3.4.1 classifies the temperature monitoring equipment as ITS when it is used as the sole means of surveillance.

The applicant's proposed design feature Technical Specification 4.2.4, "Site Temperature Limits," states that pretransfer operations, transport operations, and shipment operations are to be conducted when the working area ambient temperature is greater than or equal to 0ºF (-18ºC) and when the working area 3-day average ambient temperature is less than or equal to 91ºF (33ºC). As noted above in SER section 6.3.3, SAR table 4.3.2 showed that the HI-STORM UMAX system short-term operations maximum and minimum temperatures reported in the HI-STORM UMAX FSAR (Holtec Report No. HI-2115090, Revision 3) are similar to the site-specific data for the HI-STORE CIS Facility. The staff finds that the 0ºF (-18ºC) minimum temperature is the same minimum temperature described in the HI-STORM UMAX FSAR table 4.3.2, which the applicant incorporated by reference according to SAR section 6.4.3.3, "Minimum Temperatures." Likewise, regarding an environmental temperature of 91ºF (33ºC) (which is approximately the average maximum monthly temperature in June, July, and August and is similar to the corresponding condition in the previously reviewed HI-STORM UMAX FSAR), the staff finds that the steady-state thermal analysis discussed in SAR section 6.4.3.7, SAR table 6.4.1, and SAR table 6.4.6 supports an environmental temperature of 91ºF for a short-term operation. This is because thermal analyses using that temperature as a constant, steady-state boundary condition showed cladding (e.g., 669ºF (354ºC)) and ITS component temperatures were below normal-condition allowable temperatures with margin (e.g., 83ºF (46ºC) PCT margin relative to the 752ºF (400ºC) normal-condition allowable value) and the 96 psig (662 kPa) MPC pressure was below the 100 psig (689 kPa) normal condition design pressure.

Technical specification 4.2.3, "HI-STORM UMAX VVMs Spacing," lists a 15 foot, 6-inch minimum center-to-center pitch between adjacent VVMs. The staff finds that this pitch dimension was specified in the previously approved HI-STORM UMAX system and, according to section 4.4.1 of the HI-STORM UMAX FSAR, was a geometric parameter used in the thermal analyses.

Finally, technical specification 4.2.5.2 states that the cask transporter (that handles from above) is procedurally limited to less than or equal to 430 gallons (1,628 liters) of combustible fluids. Likewise, technical specification 4.2.5.5 states the cask transporter (that supports from below) is limited to less than or equal to 528 gallons (1,999 liters) of combustible fluid and the mass of combustible solids is limited to less than or equal to 4,479 pounds (2,032 kgs). The staff finds that these quantities are acceptable because they form the basis of the HI-STORE fire accident condition analyses, according to SAR section 6.5.2 and Holtec Report No. HI-2177553, Revision 3, which the staff discussed previously in SER section 6.3.4.4, which found temperatures and pressures were within allowable values.

Based on the discussion above, the staff has reasonable assurance that the HI-STORE CIS Facility is adequately designed through its thermal-related technical specifications to protect SSCs ITS from all postulated normal, off-normal, and accident thermal loads and environmental conditions and, therefore, meets the requirements of 10 CFR 72.122(h)(1) and 10 CFR 72.128(a).

6.3.5 Protection from Fire and Explosions

The SAR and supplemental documents, including Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility – Emergency Response Plan (Holtec Report No. HI-2177535, Revision 5)) and HI-STORE CIS Facility Fire Protection Plan (Holtec Report No. HI-2177938, Revision 1), consider the impact of, and protection from, fires and explosions. In accordance with 10 CFR 72.122(c), SSCs classified as ITS must be designed and located to continue to perform their safety functions effectively under credible fire and explosion exposure conditions.

In addition to evaluating the HI-STORE CIS Facility decay heat removal systems under normal, off-normal, and accident conditions in the preceding SER sections, the staff reviewed the fire protection measures for the HI-STORE CIS Facility to determine whether the applicant provided a basis for the ISFSI design and location of ITS structures and systems to minimize the probability and effect of fires and explosions.

With regard to the potential impact of fire associated with the CTF and ISFSI, SAR section 6.5.3 indicates that all fixed locations with combustible materials at the HI-STORE CIS Facility (e.g., charcoal filters, diesel storage tanks, diesel engines, and back-up diesel generators as mentioned in HI-STORE CIS Facility Environmental Report (Holtec Report No. HI-2167521) sections 1.4.2.3, 4.10.2, and 6.5) are not located in areas that could impact radioactive materials. Likewise, Holtec Report No. HI-2177938, section 7.1.1.2 states that all permanent (e.g., fixed) fire loads shall be located, designed, operated, and maintained in accordance with applicable NFPA code guidelines. SAR section 6.5.3 indicates that CTF construction materials do not support combustion, there are no combustible materials stored within the ISFSI, and the materials at the ISFSI are compatible with operating environments, thereby precluding explosions. Even though SAR section 14.1 states the MPC is protected from explosive effects while in the HI-TRAC boundary and in a HI-STORM UMAX system, SAR section 6.5.2.2 indicates that a design-basis external pressure event for the MPC was evaluated in the HI-STORM FW FSAR and found to be within limits. Fire-prone materials are limited to diesel fuel and hydraulic fluid associated with transfer vehicles. Proposed technical specification 4.2.5.2 procedurally limits the quantity of fire-prone materials in the cask transporter to less than or equal to 430 gallons (1,628 liters) of combustible fluids for a transporter that handles from above, and proposed technical specification 4.2.5.6 limits the quantity of fire-prone materials in the cask transporter to 528 gallons (1,999 liters) of combustible fluids and 4,479 pounds (2,032 kg) of combustible solids for a cask transporter that handles from below). SAR section 6.5 discusses the complete combustion (i.e., no fire suppression) fire impacts on ITS SSCs (e.g., MPC within VVM storage, HI-TRAC CS, HI-STAR 190) due to the flammable material (e.g., transport fuel, hydraulic fluid, tires), which, as noted in SER section 6.3.4.4, indicates that the temperatures of ITS SSCs would remain within allowable limits. The impacts of any fire would be mitigated if fire accident corrective actions were initiated, as noted in SAR section 15.3.1.2. SAR section 6.5.3 states that the small amounts of low-level solid radioactive waste temporarily located at the site are stored within flammable storage cabinets and containers (which are fire protection-related systems) and, therefore, would not be affected by nor contribute to fire events.

With regard to the potential impact of fire associated with other areas of the site, SAR section 6.5.2.1 notes that the HI-STORE CIS Facility is designed and operated as a vegetation-free storage area within the controlled area boundary with vegetation cleared land around the controlled area boundary, such that hazards associated with range-land fire and fire jump hazards are not credible. Likewise, SAR section 6.5.3 indicates that a gravel-covered fire break to control vegetation surrounds the storage pads and CTF. SAR section 2.2.2 indicates that analyses determined there would be no damage to critical operations at the HI-STORE CIS Facility from nearby postulated pipeline ruptures. For example, the analyses showed overpressures at the facility from pipeline incidents would be less than 1 psi (6.9 kPa), which is less than the 10 psi overpressure design value for HI-STORM UMAX storage canisters. In addition, SAR table 8.3.1 and SAR section 15.3.8 evaluate accident information incorporated by reference from the UMAX FSAR section 12.2.8 and note that its bounding. Although there is one active gas and distillate well, identified as Hanson State #001, within the HI-STORE CIS Facility owner-controlled area, SAR section 2.1.2 states that potential impacts from this well are bounded by the previously mentioned pipeline analysis. Similarly, SAR section 2.1.2 notes the potential fire risk from the three closest oil wells is bounded by the facility fire risk evaluation provided in SAR section 6.5.2, which indicates that, because of the large distance of the oil well and oil recovery facility from the CIS facility storage pad and haul path, the fire impacts would be limited, and emergency response would mitigate the fire.

SAR section 6.5.3 indicates that Holtec Report No. HI-2177535 describes measures for fire prevention, fire detection, and fire suppression. Additional information regarding fire prevention, fire detection, and fire suppression appears in Holtec Report No. HI-2177938. As noted in these documents, a fire protection plan consists of equipment, procedures, and operational controls that support prevention and mitigation of fire-related accidents. Regarding procedures and operational controls, Holtec Report No. HI-2177535 section 5.3.1 indicates that controls are to be in place to prevent combustible materials in the protected CIS facility area. Likewise, Holtec Report No. HI-2177938 indicates that site procedures include control of ignition sources (e.g., welding) in the HI-STORE CIS Facility protected area.

Holtec Report No. HI-2177535, section 5.3.1 indicates that fire protection systems associated with the HI-STORE CIS Facility, which is a facility that handles radioactive materials, are designed in accordance with requirements of applicable National Fire Protection Association

(NFPA) codes. The staff notes relevant fire protection requirements that include applicable NFPA codes are found in NFPA 801 "Standard for Fire Protection for Facilities Handling Radioactive Materials." According to Holtec Report No. HI-2177535, sections 5.3.1 and 6.3, and HI-STORE CIS Facility Environmental Report, section 1.4.2.3, the site's fire protection systems include standpipes (wharf hydrants), hoses, sprinklers, portable extinguishers (foam, inert gases, dry chemicals), fire pump diesel engine, and water of adequate volume to supply the water hoses, sprinklers, water spray systems. Likewise, Holtec Report No. HI-2177938, section 7.2.5 (proprietary) indicates that the site will include fire barriers, fire suppression, and fire detection systems (i.e., fire and smoke alarms). In addition, Holtec Report No. HI-2177535, section 6.3 and Holtec Report No. HI-2177938, sections 7.2.4 (proprietary) and 7.2.5 indicate the presence of internal and external communication and alarm systems. Holtec Report No. HI-2177535, section 6.3 states that all applicable OSHA regulations, NFPA codes, National Electric Code, and National Fire Code will be followed regarding location, testing, and maintenance of emergency equipment. SAR section 3.1.4.7 also indicates that fire protection systems would be maintained.

With regards to training, SAR section 10.3.1 and section 10.4.2 indicate that personnel would be trained on the Fire Protection Plan. Likewise, Holtec Report No. HI-2177938, sections 7.2.3 and 7.5 (both proprietary), and Holtec Report No. HI-2177535, section 7.2.2 (proprietary) provide information related to personnel responsibilities and the training of site personnel related to emergency response, including fire protection. In addition, Holtec Report No. HI-2177535, sections 5.3.1 and 7.2.3 and Appendix D and Holtec Report No. HI-2177938, Appendix 8.0 (proprietary) provide information related to the availability of local off-site emergency response assistance. Finally, Holtec Report No. HI-2177938, section 7.3 (proprietary) states that, at a minimum, annual fire protection audits will be performed by qualified fire safety personnel who meet NFPA code guidelines. The audits would review the Fire Protection Plan, site modifications, fire protection procedures, fire hazards analyses, design of fire protection equipment, training activities, fire protection equipment and supplies, and records associated with off-site support agency interfaces.

Based on its review of the information and evaluations in the application as described above, the staff has reasonable assurance that the HI-STORE CIS Facility is adequately designed to protect ITS SSCs from postulated fires and explosions to meet the requirements of 10 CFR 72.122(c).

6.4 Evaluation Findings

Based on its review of the SAR and supporting documents, the staff determined the following:

- SSCs ITS are described in sufficient detail in the SAR to enable an evaluation of their heat removal effectiveness. The cask SSCs ITS remain within their operating temperature ranges in accordance with 10 CFR 72.122 and 10 CFR 72.92(a).
- The HI-STORE CIS Facility is designed with a heat removal capability having testability and reliability consistent with its ITS classification as required by 10 CFR 72.128.
- The spent fuel cladding is protected against degradation that leads to gross ruptures by maintaining the cladding temperature for 40 years below 400°C in an inert environment.

Protection of the cladding against degradation will allow ready retrieval of spent fuel assemblies for further processing or disposal as required by 10 CFR 72.122.

- The staff concludes that the site-specific fire and explosion hazard evaluations are acceptable and that the fire protection program meets the requirements of 10 CFR 72.122(c). This conclusion is based on the applicant meeting the guidelines of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” section 9.5.1.1, “Fire Protection Plan,” as well as applicable industry standards. In meeting these guidelines, the applicant has (1) provided an acceptable basis for the design and location of safety related structures and systems to minimize the probability and effect of fires and explosions, (2) used noncombustible and heat resistant materials whenever practical, and (3) provided fire detection and firefighting systems of appropriate capacity and capability to minimize adverse effects of fire on ITS systems.

6.5 References

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Holtec International, Inc., “Final Safety Analysis Report on the HI-STORM FW MPC Storage System,” Holtec Report No. HI-2114830, Revision 6, June 18, 2019, NRC Docket 72-1032. ML19177A171.

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Holtec International, Inc., “Thermal Analysis of HI-TRAC CS Transfer Cask,” Holtec Report No. HI-2177553, Revision 3, Docket 72-1051, March 18, 2022. ML22108A130 (proprietary).

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Holtec International, “Attachment 2 - Proposed HI-STORE License/Technical Specifications,” November 23, 2022. ML22331A005.

Holtec International, Inc., “Licensing Report on the HI-STORE CSI Facility,” Revision 0T, Holtec Report No. HI-2167374, NRC Docket 72-1051, January 20, 2023. ML23025A112.

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NRC, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” section 9.5.1.1, “Fire Protection Program,” Revision 0, March 2009. ML090510170.

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7 SHIELDING EVALUATION

Holtec International (the applicant) described the shielding design for the proposed HI-STORE Consolidated Interim Storage (CIS) Facility in Chapter 7, “Shielding Evaluation,” of the safety analysis report (SAR), Revision 0T, dated January 20, 2023.

7.1 Scope of Review

The staff evaluated the applicant’s shielding design for the HI-STORE CIS Facility by reviewing the information provided in the SAR, proposed technical specifications (TS), the applicant’s responses to the staff’s requests for additional information, and information that the applicant incorporated by reference into the SAR. The staff also considered relevant information in SAR chapters 1, 2, 3, 4, 6, 17, and 18.

7.2 Regulatory Requirements

The regulatory requirements relevant to the shielding evaluation of the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”
- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”
- 10 CFR 20.1301, “Dose limits for individual members of the public”
- 10 CFR 20.1302, “Compliance with dose limits for individual members of the public”

These requirements define the information to be submitted in an application; the applicable radiation dose limits; and the requirements for structures, systems, and components (SSCs) and operations to control radiation exposures to personnel.

The applicant proposed to use the cask systems identified in table 1-1 of this safety evaluation report (SER), each of which the NRC staff has reviewed and approved under certificates of compliance (CoCs) issued under the provisions of 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” or 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

7.3 Staff Evaluation

7.3.1 Objectives of the Review

The HI-STORE CIS Facility is a centralized interim spent nuclear fuel (SNF) storage facility. It is an independent spent fuel storage facility (ISFSI) pursuant to the regulations of 10 CFR Part 72.

The objective of the shielding review is to ensure that the shielding design of the ISFSI is sufficient to meet the regulatory requirements of the dose limits to the public and occupational workers as prescribed in 10 CFR 72.104 and 10 CFR 72.106.

Unless otherwise stated, the staff evaluated the HI-STORE CIS Facility system shielding design by reviewing the SAR, the applicant's responses to the staff's requests for additional information (RAIs), and other relevant literature.

The staff performed its review following the guidance in NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000. This chapter documents the staff's review and conclusions on the shielding design of the HI-STORE CIS Facility.

7.3.2 General Shielding Design Features

The HI-STORE CIS Facility is designed to store pressurized water reactor (PWR) and boiling-water reactor (BWR) SNF assemblies that are loaded in multipurpose canisters (MPCs) MPC-37 and MPC-89. There are 37 PWR SNF assemblies in an MPC-37 canister and 89 BWR SNF assemblies in an MPC-89 canister. Nonfuel hardware (NFHW) discharged from PWRs can also be loaded together with the SNF in the MPC-37 canister.

The MPCs are welded to prevent releases of radioactive materials from the contents. In the HI-STORE CIS Facility, the MPCs are stored in inground vertical ventilated modules (VVMs). The TS for the HI-STORE CIS Facility defines the specifications of the allowable SNF and NFHW.

The HI-STORE CIS Facility is designed primarily based on the HI-STORM UMAX SNF dry cask storage system that has been certified by the NRC under Amendment 2 of CoC No. 1040, Docket No. 72-1040, dated January 9, 2017. As discussed in this chapter, the applicant incorporated by reference certain information from Revision 3 of the UMAX system final SAR (FSAR), Holtec Report No. HI-2115090, dated June 29, 2016, pursuant to the regulation of 10 CFR 72.18, "Elimination of repetition." However, the HI-STORE CIS Facility design introduced several significant variations from the certified UMAX system.

The four major features related to shielding design at the HI-STORE CIS Facility that are not part of the certified UMAX storage system are (1) a revised closure lid for the VVMs, (2) the HI-TRAC CS [concrete shielded] transfer cask, (3) exclusive use of the HI-STAR 190 transportation package as the packaging system to transport MPCs to the facility, and (4) the cask transfer building (CTB).

The applicant provided a detailed description of the HI-STORE CIS Facility design in SAR chapter 1, "General Description." SAR chapter 7.0, "Shielding Evaluation," discusses the following six major SSCs of the HI-STORE CIS Facility design that are important to the safety function of radiation shielding:

(1) weld-sealed canisters for storage of SNF assemblies and NFHW (PWR only)

(2) VVMs for storage of the MPCs

(3) storage pads which contain the VVMs

(4) the controlled area boundary within which the facility owner and operator can control access to the facility as needed

(5) the HI-TRAC CS transfer cask, which is used to receive, transfer, and load the MPCs into the VVMs

(6) the CTB to facilitate the receipt and transfer of the MPCs from the HI-STAR 190 transportation package to the HI-TRAC CS transfer cask

Each of these SSCs provides a shielding function for the facility at various stages of the facility operations during transfer or storage of the SNF and NFHW (i.e., the authorized contents as specified in SAR section 4.1.1, “Spent Fuel Canisters.”

In SAR figures 2.1.6(a) through (d), the applicant provided a clear layout of the controlled area boundary of the facility. The staff reviewed the layout (including the controlled area boundary) and finds that the applicant has included sufficient information for the staff to perform a detailed review of the shielding design of the facility. On this basis, the staff finds that the information provided in the SAR for the HI-STORE CIS Facility is acceptable for review.

The clear definition of the controlled area boundary provides the basis for the staff to determine that the shielding design of the facility meets the regulatory requirements of 10 CFR 72.104 and 10 CFR 72.106 with respect to the dose limits applicable to a real individual at or beyond the controlled area boundary and to occupational workers who perform the daily operations of the facility.

The VVM is a subgrade storage module designed to hold the MPCs that contain the SNF and NFHW (i.e., the authorized contents as specified in SAR section 4.1.1) during the period of licensed operation. The MPCs will be moved from the CTB to the ISFSI pad and lowered into the designated VVMs for storage.

After an MPC is loaded into a VVM cavity, the closure lid is installed on the top of the VVM. The new VVM design used at the facility, known as Version C, uses a square lid with modified air outlet path design, whereas the HI-STORM UMAX system Amendment 2 uses a round closure lid. Sheets 1, 2, 3, 4, and 6 of licensing drawing 10875 (proprietary) in the SAR show design details of the closure lid Version C.

The staff reviewed the licensing drawings for the closure lid and finds that the applicant has provided sufficient details for the staff to review and perform confirmatory shielding calculations. On this basis, the staff finds that the SAR has provided a detailed description for the closure lid design and the description meets the regulatory requirements of 10 CFR 72.24(c) which requires: “The design of the ISFSI or MRS in sufficient detail to support the findings in § 72.40 for the term requested in the application.” On these bases, the staff found description of the HI-STORE CIS Facility to be acceptable.

In accordance with the design, the HI-STORE CIS Facility will store only the PWR SNF and NFHW in the MPC-37 canisters or BWR SNF in the MPC-89 canisters. In addition, SAR table 1.0.5, “Transport Casks Allowed for Receipt of Canisters at HI-STORE,” specifies that the loaded canisters must be delivered to the site using the HI-STAR 190 transportation packaging system that has been approved under CoC No. 9373, Revision 1, Docket 71-9373, with SAR Revision 3.

The staff reviewed this requirement and finds that it provides a bounding condition for defining the source terms for the authorized contents. Section 7.3.4 of this SER provides more details on the source term calculation and why the source terms calculated for the HI-STAR 190 are more conservative.

The HI-TRAC CS transfer cask is designed to receive the MPCs that are delivered to the facility. The MPCs are transported to the ISFSI pad in the HI-TRAC CS transfer cask where they are loaded into VVMs. There is no operation to load or unload the SNF into or out of the canisters at the facility.

The HI-TRAC CS transfer cask is different from the transfer cask approved for the UMAX in CoC No. 1040, amendment 2. The difference between the HI-TRAC CS design and the transfer cask for the UMAX system is that the new transfer cask uses a concrete shield encased in two concentric steel shells, whereas the transfer cask for the UMAX system uses a steel, lead, and water jacket shielding design. The dimensions between the two transfer casks also differ to provide adequate radiation shielding for the HI-TRAC CS because concrete is less effective for shielding than the lead, steel, and water shielding design used in the transfer cask approved for the UMAX system.

The applicant provided the structure, dimensions, and bill of materials for the HI-TRAC CS in SAR licensing drawings 10868, Revision PR2 (proprietary). The staff reviewed the licensing drawings and the bill of materials and finds that the applicant has provided sufficient information to permit a review of the shielding design of the transfer cask. On this basis, the staff finds that the information in the SAR for the HI-TRAC CS transfer cask is sufficient and acceptable.

The CTB is a reinforced concrete structure that houses the canister transfer facility and the cask-receiving area. It also provides storage space for ancillary equipment used in short-term operations. The canisters transported to the site using the HI-STAR 190 transportation package will be unloaded and transferred to the HI-TRAC CS transfer cask inside the CTB. The HI-TRAC CS will transfer the canisters to and install the MPC in the VVMs. In accordance with SAR section 10.3.3.6, “Removal of Canisters from the CEC,” the HI-TRAC CS will also be used for the reverse operation, moving MPCs from the VVMs to the CTB for subsequent transport of the canisters off site.

In the SAR, the applicant provided detailed design dimensions and a bill of materials for the CTB in licensing drawing 10912, Revision PR1 (proprietary). The staff reviewed the licensing drawings and the bill of materials and finds that the applicant has provided sufficient information to permit a review of the shielding design of the CTB. On this basis, the staff finds the information provided in the SAR for the CTB is acceptable. Chapter 5 of this SER documents the structural evaluation of this ISFSI design, and upon which the shielding design relies. The design includes the Version C lid design, HI-TRAC CS, and CTB design.

The applicant also provided a diagram (drawing 10940, proprietary) of the HI-STORE CIS Facility that includes the general layout, the restricted area boundary, the controlled area boundary, and the CTB in SAR section 1.5, “Licensing Drawings.” In accordance with SAR section 7.4.2.1, “Normal Conditions,” the minimum distance from the ISFSI pad to the controlled area boundary is 400 meters (1312 feet). The applicant is required to control public access to the controlled area and to record the annual dose at the controlled area boundary.

The staff reviewed the information provided in the SAR and finds that the controlled area boundary meets the requirement of 10 CFR 72.104(b), which states, “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.” The staff also finds that the facility meets the requirements of 10 CFR 72.106(c) because the applicant will have the means to restrict access to the controlled area. On these bases, the staff determined that the applicant’s design for the controlled area boundary is acceptable.

In SAR section 3.1.4.4, “Surveillance of the HI-STORM UMAX Storage Systems,” the applicant stated the following:

> Radiation doses emitted from the storage casks are measured by thermoluminescent dosimeters (TLDs) located at the protected area (PA) and owner controlled area (OCA) boundaries to ensure doses are within 10 CFR 20.1301 and 10 CFR 72.104 or 40 CFR 191 limits.

The staff reviewed the arrangement of the radiation monitoring measures. The staff finds that the term “OCA” used in the SAR is the same as the controlled area as defined in 10 CFR 72.3. On this basis, the staff determined that there is reasonable assurance that the annual dose received by a real individual at or beyond the controlled area will not exceed the dose limits specified in 10 CFR 72.104(a) under normal and off-normal operations and 10 CFR 72.106(b) under design basis accident conditions with the limit of 2,000 hours per year occupancy time as specified in the SAR. The staff reviewed the information for the facility design and finds that the applicant has provided information that is sufficient for the staff to perform a shielding evaluation of the HISTORE CIS Facility design and that satisfies the requirements of 10 CFR 72.24(b), (c)(3), and (e).

7.3.3 Authorized Contents

In the SAR, the applicant described the SNF and NFHW to be stored in the HI-STORE CIS Facility. SAR chapter 1 describes how the first phase construction of the facility will have a capacity of 8,680 metric tons of uranium in the form of commercial SNF in a total of 500 welded canisters. The PWR fuel must be stored in the MPC-37 canisters and the BWR fuel must be stored in the MPC-89 canisters as defined in the TS for the HI-STORE CIS Facility.

TS 2.1, “Approved Contents, Fuel Specifications and Loading Conditions,” for the HI-STORE CIS Facility provides detailed specifications for the allowable contents, including the maximum burnups of the authorized contents (68.2 gigawatt-days per metric ton of uranium (GWd/MTU) for PWR fuel assemblies and 65 GWd/MTU for BWR assemblies), which the applicant specified through a reference to tables 7.C.8 and 7.C.10 in Revision 3 of the HI-STAR 190 SAR.

In accordance with TS 2.1 and SAR tables 1.0.4, “Canisters Allowed for Storage in HI-STORM UMAX at HI-STORE,” and 1.0.5, “Transport Casks Allowed for Receipt of Canisters at HI-STORE,” SNF received at the facility must be loaded and sealed in the MPC-37 or MPC-89 canisters transported exclusively by using the HI-STAR 190 packaging system. TS 2.1 further limits the allowable contents to that specified in appendix 7.C to the HI-STAR 190 SAR, Revision 3. As stated in TS 2.1, the burnup, enrichment, and cooling time (BECT) combination for each loading zone in each loading pattern is defined in table 7.C.8 and table 7.C.10 of appendix C to the HI-STAR 190 SAR, Revision 3. The BECT explicitly defines the qualification criteria for the SNF to be stored. Because the TS for the HI-STORE CIS Facility incorporates these fuel qualification criteria, the allowable contents specified in the HI-STAR 190 SAR, Revision 3, are identical to the allowable contents of the HI-STORE CIS Facility

The HI-STAR 190 package is more limiting because 10 CFR 71.47(b)(3) requires the maximum dose rate to not exceed 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle. Whereas there is no dose rate limit on the surface of a storage cask. As such, the authorized contents for a transportation package are more limiting in order to meet the prescribed dose rate limits set forth in 10 CFR 71.47(b)(3). In addition, the UMAX system allows for storage of mixed oxide (MOX) fuel whereas the HI-STAR 190 transportation package prohibits transportation of MOX fuel. Because the BECTs specified in revision 3 of the HI-STAR 190 SAR are more limiting, the applicant used them in the source term calculations for the HI-STORE CIS Facility.

The staff reviewed the fuel qualification tables as defined in tables 7.C.8 and 7.C.10 of the HI-STAR 190 SAR, Revision 3), and finds that the allowable SNF contents are clearly defined for calculations of the source terms of these contents.

The staff reviewed the descriptions of the approved contents in the TS for the HI-STORE CIS Facility and the HI-STAR 190 SAR, Revision 3. The staff finds that the requested approved contents in TS 2.1 are consistent with the shielding analyses in the SAR. On this basis, the staff determined that the applicant has accurately described the approved contents with sufficient details for the staff to verify the shielding design of the HI-STORE CIS Facility. On these bases, the staff finds that the description of the authorized contents meets the regulatory requirements of 10 CFR 72.24(a). The staff also finds that the description of the approved contents is consistent with the acceptance criterion of NUREG-1567 for specification of allowable contents. On these bases, the staff determined that the applicant's description of the approved contents is acceptable.

7.3.4 Source Calculations

7.3.4.1 Neutron and Gamma Sources

The radiation source terms that are important to the shielding design of the HI-STORE CIS Facility consist of two parts: gammas and neutrons. The major control parameters for source terms include the fuel assembly BECT combination. Using the BECT, the applicant further calculated the radiation sources for each region of a loading pattern for the MPC-37 for PWR fuel and the MPC-89 for BWR fuel. Specifically, the applicant calculated the source terms for the approved contents of the HI-STORE CIS Facility using the following steps:

(1) Determine the decay heat limit for each basket loading region of the MPC-37 and MPC-89 canisters. Tables 7.C.7 and 7.C.9 of the HI-STAR 190 SAR, Revision 3, give the final maximum decay heat load value(s) for the available loading patterns for PWR and BWR fuel, respectively.

(2) Determine all the limiting BECT combinations based on decay heat and limits and the fuel specifications for PWR and BWR fuel as presented in tables 7.C.8 and 7.C.10, respectively, of the HI-STAR 190 SAR, Revision 3.

(3) Calculate the fuel neutron, photon, and fuel hardware cobalt source terms using the bounding BECT combinations derived in step 2.

(4) For SNF in each basket-loading region, compare the source terms and select the maximum source strengths for each energy group (neutrons and photons) or the maximum activity (cobalt) as the design-basis sources for shielding calculations.

(5) For canisters containing NFHW, use the maximum source strength (e.g., neutron sources) or cobalt-60 activity as the design-basis sources in shielding calculations.

The applicant described in detail the source term calculation in the technical report "HI-STAR 190 Source Terms and Loading Patterns Using SCALE 6.2.1" (proprietary), Revision 2, dated April 30, 2019. In this report, Holtec demonstrated that the BECT provides bounding source terms for all approved contents (i.e., both SNF and NFHW), as specified for the HI-STAR 190 transportation package. The applicant used the bounding source terms in all shielding analyses related to the HI-STORE CIS Facility operations, which include receipt of the MPCs from the HI-STAR 190 package; HI-TRAC CS loading and unloading operations; storage operations under normal, off-normal, and accident conditions consistent with structural evaluations of the systems; and carbon-14 generation.

Axial Burnup

The proprietary HI-STAR 190 source term technical report describes the dependencies of the source terms as a function of the axial burnup profiles in its shielding models since the gamma source terms of the SNF are dependent on the axial fuel burnup. The report used the same burnup profile values as used in the shielding calculations for the approved UMAX Amendment 2 and HI-STAR 190 package.

The staff reviewed the applicant's scaling of the source distributions along the axial direction of the SNF assembly and finds that the profiles the applicant used are consistent with NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in BWR Burnup Credit Analyses," issued March 2003; NUREG/CR-7224, "Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit," issued August 2016; and the guidance in NUREG-1567. Therefore, based on its review, the staff determined that the applicant has used appropriate burnup profiles to account for the peaking of the source terms along the axial directions of the SNF assemblies. On this basis, the staff finds that the burnup profiles used by the applicant are acceptable.

Neutron Source

Fission reactions between the neutrons emitted by the SNF and the fissile materials in the SNF produce secondary neutrons. This neutron source is also known as the "subcritical multiplication source." The applicant included this neutron source in the neutron shielding model by using the default neutron transport calculation of the Monte Carlo N-Particle Transport (MCNP) code (LANL, 2003).

The applicant provided the neutron source terms in SAR table 7.1.5(A), "Calculated PWR Neutron Source Per Assembly for Design Basis Bounding Source Terms," and table 7.1.5(B), "Calculated BWR Neutron Source Per Assembly for Design Basis Bounding Source Terms," for the design-basis PWR and BWR fuel, respectively. As discussed above, the shielding calculation models explicitly include the neutron sources produced by subcriticality.

Gamma Source

The applicant provided the gamma source terms in SAR table 7.1.2(A), “Calculated PWR Fuel Gamma Source Per Assembly for Design Basis Bounding Source Terms,” and table 7.1.2(B), “Calculated BWR Fuel Gamma Source Per Assembly for Design Basis Burnup and Cooling Time,” for the design-basis PWR and BWR fuel, respectively. The applicant also provided the fuel hardware gamma source terms in SAR table 7.1.4(A), " Calculated ^{60}Co Source Per Assembly for Design Basis PWR Fuel at Design Basis Bounding Source Terms,” and table 7.1.4(B)," Calculated ^{60}Co Source Per Assembly for Design Basis BWR Fuel and Bounding Source Terms,” for the design-basis PWR and BWR fuel, respectively.

As stated in the HI-STAR 190 source term technical report, Holtec Report No. HI-2167524, the applicant also included the secondary gammas that are produced by the capture reaction when a neutron reacts with SNF and cask structural materials. When an atom in the fuel or the cask structural materials captures a neutron, it may decay to stable state by emitting gamma radiation, which is not part of the gammas that have been calculated for the SNF or NFHW stored in the cask.

For NFHW, the applicant used the source terms specified in the HI-STAR 190 SAR, Revision 3. The applicant also provided detailed discussions on the various NFHW, including the material compositions, their uses in the reactors, expected exposures, design-basis cooling times, and the calculated source terms.

The staff reviewed the calculation approach and the results of the primary sources provided by the applicant in the SAR. Based on the information in the SAR for the HI-STORE CIS Facility and the HI-STAR 190 SAR, Revision 3, and its own confirmatory calculation, the staff determined that the source terms presented in these SARs are consistent. On these bases, the staff finds that the source terms calculated by the applicant are conservative and acceptable.

7.3.4.2 Carbon-14 Produced at the HI-STORE CIS Facility

One of the phenomena associated with dry cask storage is the production of the radioactive isotope carbon-14 when air is irradiated by neutrons. Neutrons emitted from the SNF stored in the ISFSI react with the air that flows through the gap between the SNF canister and the VVM storage module, as well as the air surrounding the top of the VVM. The applicant considered this additional radiation source and calculated the production of carbon-14 in the air that flows through the gap between the canister and VVM wall and the air surrounding the ISFSI. The applicant included its calculations of carbon-14 and its contribution to the total dose to the real individual at the controlled area boundary and the occupational workers who operate the ISFSI in the calculation package “HI-STORM UMAX C-14 Dose Rate versus Distance,” Revision 0, dated November 18, 2020.

In this calculation package, the applicant described the methods and assumptions it used in the calculation of carbon-14 production and dispersion. The applicant calculated carbon-14 production and dispersion using the guidance from NRC Regulatory Guide 4.20, “Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors,” Revision 1, issued April 2012, and U.S. Environmental Protection Agency (EPA) Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” issued September 1988, and Federal Guidance Report No. 12, “External Exposure to Radionuclides in

Air, Water, and Soil," issued September 1993, for calculation of gaseous radioactive material dispersion in the air.

The staff reviewed the carbon-14 production and the doses to the real individual at the controlled area boundary and to the workers who operate the facility. Based on its review, the staff finds that the calculated dose includes all possible pathways for the committed effective dose equivalent to the occupational workers as well as the real individual at the control area boundary. The applicant used the guidance provided by the NRC staff and EPA to assess carbon-14 dispersion. The staff confirmed that this calculation methodology meets the acceptance criteria in the relevant NRC and EPA guidance. On this basis, the staff determined that the carbon-14 source calculation performed by the applicant is acceptable.

7.3.5 Shielding Calculations

As stated earlier in this chapter, the shielding design of the HI-STORE CIS Facility includes six major SSCs that are important to safety, including the MPCs, VVMs, ISFSI pad, controlled area boundary, HI-TRAC CS transfer cask, and CTB. The HI-STORE CIS Facility also uses the HI-STAR 190 transportation cask.

In SAR section 7.4.2, "Dose and Dose Rate Estimates," the applicant calculated the total doses at the controlled area boundary with the contributions from the various stages of the facility operations. The applicant also calculated the dose rate as a function of distance from the ISFSI pad to the controlled area boundary. Specifically, the applicant provided the dose or dose rate estimates for the following operations:

- dose contribution from the SNF stored in the 500 MPCs (MPC-37 and MPC-89) loaded in the HI-STORE CIS Facility, consistent with the description in the TS for the facility, and conservative assumptions about the content of each canister (applicant's calculation assumed that all 500 canisters have been loaded in the ISFSI VVMs)
- dose contribution from the HI-STAR 190 transportation package at reception and during unloading of the canisters from the HI-STAR 190 transportation package
- dose contribution from the HI-TRAC CS while it is inside the CTB
- dose contribution from HI-TRAC CS during transport and loading of the canisters into the VVMs in the ISFSI pad
- dose contribution from carbon-14 generated when the canister is in the VVMs in the ISFSI pad
- dose contribution from the HI-STAR 190 transportation packages while the packages are at the staging area waiting for reception

For radiation protection requirements of 10 CFR Part 20, "Standards for Protection against Radiation," the applicant also calculated the following:

- dose rate versus distance from the HI-STAR 190 to the controlled area boundary
- dose rate from the canister transfer operation that takes place in the CTB
- dose rate versus distance from the HI-TRAC CS to the controlled area boundary

- dose rates at the surface, 0.5 meter (1.6 foot), 1 meter (3.3 feet), and 2 meters (6.6 feet) from the HI-TRAC CS
- dose rates at the surface and 1 meter (3.3 feet) from a single HI-STORM UMAX

The applicant credited the shielding provided by the concrete walls of the CTB in its calculation of the dose rate versus distance (outside the CTB) for the canister transfer operations that take place inside the CTB. The structural review of the HI-STORE CIS Facility has determined that the CTB meets the structural performance acceptance criteria and is classified as an important-to-safety structure. Chapter 5 of this SER details the staff's structural review.

7.3.5.1 Normal Condition

As described in SAR section 7.4.2.1, "Normal Conditions," the applicant calculated the dose for the real individual at the control area boundary for the HI-STORE CIS Facility under normal conditions of operations. The following are the 10 CFR 72.104 criteria for radioactive materials in effluents and direct radiation during normal operations:

(1) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 millirem (mrem) to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ.

(2) Operational restrictions must be established to meet ALARA objectives for radioactive materials in effluents and direct radiation. Subpart C, "Occupational Dose Limits," and Subpart D, "Radiation Dose Limits for Individual Members of the Public," in 10 CFR Part 20 specify additional requirements for dose limits. SAR chapter 11 specifically addresses these regulations. In accordance with ALARA practices, SAR table 11.3.2 establishes design objective dose rates for the HI-STORM FW system.

The applicant documented the calculated annual dose for the real individual at and beyond the controlled area boundary in SAR table 7.4.3, "Dose Rates as a Function of Distance from 500 Loaded HI-STORM UMAX VVMs for Fuel Assemblies with Bounding Source Terms from Reference [7.1.1]." The results demonstrate that the facility, including the ISFSI pad loaded with 500 canisters, will meet the regulatory dose limits of 10 CFR 72.104 with a large safety margin.

In the calculation of the dose for a real individual at the controlled area boundary, the applicant assumed 2,000 hours per year occupancy time. In SAR section 7.4.2.1, the applicant stated that the land surrounding the facility's controlled area boundary belongs to the U.S. Bureau of Land Management, and therefore no residence will be built on the land. The only occupants are oil field workers or other temporary workers. Because the maximum annual work time is 2,087 hours, the staff determined that this assumption is acceptable.

The staff reviewed the shielding analyses performed by the applicant for the HI-STORE CIS Facility under normal operations. The calculations include reception of the loaded canisters from the HI-STAR 190 package, transfer of the canister from the HI-STAR 190 package to the HI-TRAC CS transfer cask, transport of the canister from the canister transfer building to the ISFSI pad and loading the canister to the VVM, closure of the VVM, and daily maintenance of the storage system. The staff finds that the applicant has considered all operations, and the applicant's shielding calculation models are consistent with the licensing drawings. On these bases, the staff finds that the applicant has identified all plausible operations and applicable shielding requirements for the HI-STORE CIS Facility.

7.3.5.2 Off-Normal and Accident Conditions of Operations

For off-normal conditions at the HI-STORE CIS Facility, in SAR Table 15.0.1, “Material Incorporated by Reference in this Chapter,” the applicant incorporated by reference information about off-normal conditions from section 12.1 of the HI-STORM UMAX FSAR. SAR table 15.0.1 also noted that the HI-STORM UMAX Version C system is essentially the same as the version approved for use in the HI-STORM UMAX docket and the severity of events is no greater than off-normal and accident events evaluated in the HI-STORM UMAX FSAR; it follows that the consequences evaluated in it are bounding.

SAR section 15.2, “Off-Normal Events,” discusses the following four off-normal events as credible:

- off-normal pressure (SAR section 15.2.1)
- off-normal temperature (SAR section 15.2.2)
- partial blockage of air inlet and outlet ducts (SAR section 15.2)
- hypothetical nonquiescent wind (SAR section 15.2)

Section 12.1 of the HI-STORM UMAX FSAR, which the staff accepted in its certification of the HI-STORM UMAX storage system, concluded that none of these off-normal events would impact shielding.

In SAR section 7.4.2.2, “Off-Normal and Accident Conditions,” and section 15.3, “Accidents,” the applicant identified the potential accident conditions during the operation of the HI-STORE CIS Facility. In SAR section 7.4.2.2, the applicant discussed a missile impact on an adjacent loaded canister during the construction of the new VVMs and a potential fire accident involving the HI-TRAC CS transfer cask. For the missile impact event, the applicant stated that this was the bounding accident case from a shielding perspective and referenced an evaluation in sections 5.1 and 5.3 of the HI-STORM UMAX FSAR. The referenced evaluation concluded that regulatory dose limits would be met. For the fire accident event, the applicant referenced Holtec Report No. HI-2177553, “Thermal Analysis of HI-TRAC CS Transfer Cask,” Revision 3, dated August 13, 2021. This analysis also concluded that the regulatory dose limit of 10 CFR 72.106 would be met.

The staff reviewed the applicant’s justifications for considering the accident conditions and concluded that the applicant has properly identified all plausible accidents that may occur during the operations of the facility. On this basis, the staff finds that there is reasonable assurance that these two accidents are the most severe ones that may cause the system to lose its shielding safety function. Section 7.3.7 of this SER documents the staff’s detailed evaluation of the applicant’s shielding analyses for the HI-STORE CIS Facility under accident conditions.

7.3.6 Computer Codes and Flux to Dose Rate Conversions

7.3.6.1 Computer Codes Used in Shielding Calculations

As described in “HI-STORE CIS Facility Site Boundary Dose Rates Calculations for HI-STORM UMAX System,” Revision 4, dated March 22, 2022, the applicant used the MCNP-5.1 computer code and the cross section library, “MCNP—A General Monte Carlo N-Particle Transport Code,” Version 5 (2003), distributed with the code, in all of its shielding analyses for the HI-STORE CIS

Facility and the HI-STAR 190 transportation package. The applicant used the same computer code and cross section library in its carbon-14 production calculation.

The MCNP is a general-purpose, continuous-energy, generalized-geometry, Monte Carlo solution method-based radiation transport code developed by Los Alamos National Laboratory. The code is designed to track many particle types over a broad range of energies. It is widely used in the nuclear industry and accepted for use by the NRC (NUREG-1567) since it is verified and validated and has been thoroughly benchmarked for shielding analyses. Examples of applications include radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear criticality safety, detector design and analysis, and fission and fusion reactor designs. The code treats three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. On these bases, the staff determined that the use of the MCNP code for the HI-STORE CIS Facility shielding calculations is acceptable.

The cross section library distributed with MCNP-5.1 is a combination of cross section data mainly from ENDF/B-VI and other sources. These cross section data are well suited to a wide range of shielding applications, including the NRC-approved UMAX storage system and HI-STAR 190 package, and thus are adequate for the HI-STORE CIS Facility shielding design calculations.

7.3.6.2 Flux-to-Dose-Rate Conversion

The MCNP code calculates the neutron and gamma flux at specified locations, and the calculated fluxes must be converted into dose rates for comparison to regulatory limits. As described in the boundary dose calculation, the applicant used MCNP to calculate neutron and photon flux and the 1977 edition of the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1 standard for the neutron and gamma flux to dose rate conversion factors. The staff reviewed the dose rate calculation in ANSI/ANS 6.1.1 (1977) and finds that this approach is consistent with the guidance in NUREG-1567 and therefore acceptable.

7.3.7 Shielding Evaluations

As discussed in the preceding sections of this chapter, the shielding evaluation for the HI-STORE CIS Facility consists of several calculation subcomponents. For clarity, the staff documented its review of the applicant’s shielding calculations in component-by-component order. The following subsections of this SER document the staff’s evaluation of each component or system that contributes to dose to the public (i.e., a real individual at and beyond the control area boundary) and the occupational workers who perform operations of the facility pursuant to the regulatory requirements of 10 CFR 72.104 and 10 CFR 72.106.

7.3.7.1 Shielding Evaluation for the HI-TRAC CS Transfer Cask

7.3.7.1.1 Model description for the HI-TRAC CS transfer cask

As discussed above, the HI-TRAC CS transfer cask is used to receive, move, and install the MPCs in the VVMs. The applicant included the shielding component arrangement, the dimensions, and bill of materials as shown in licensing drawing No. 10868, Revision PR2, of the SAR. The applicant developed a shielding model for the HI-TRAC CS transfer cask in accordance with the information shown in the licensing drawing. The applicant provided the calculated dose rates around the HI-TRAC CS in Table 7.4.1, “Dose, Revision Rates from the HI-TRAC CS MPC-37 Design Basis Fuel with Bounding Source Terms.” In table 7.4.4, the applicant also provided the dose rate at 100 meters (328 feet) from the HI-TRAC CS and the dose rate versus distance from a single HI-TRAC CS with bounding source terms.

The applicant provided an output file of the MCNP shielding model for the HI-TRAC CS to demonstrate that the shielding calculations for the transfer cask have properly converged for all detector locations.

The staff reviewed the model using the graphical display software Visual Editor to examine the consistency of the model with the licensing drawing of the HI-TRAC CS. The staff also reviewed the values in the corresponding output file and found that the applicant has modeled the HI-TRAC CS transfer cask accurately in the MCNP model. The staff also finds that the shielding calculations have properly converged for all detector locations (i.e., all tallies have passed the 10 statistical tests that are implemented in the code to ensure that the results are reliable). On this basis, the staff finds that the shielding calculations performed by the applicant for the HI-TRAC CS transfer cask are acceptable.

7.3.7.1.2 Material properties for the HI-TRAC CS transfer cask

The applicant provided the material compositions and densities for the HI-TRAC CS transfer cask in SAR table 7.3.1, “Composition of the Materials – HI-STORE CIS Facility.” The staff reviewed the descriptions for the materials provided in the SAR and finds that the descriptions are consistent with the bill of materials as defined in the licensing drawings and used in the shielding model for calculating the dose rate around the transfer cask and the contribution to dose by a real individual at the controlled boundary. Also, based on the conclusion from the staff’s material review documented in chapter 17 of this SER, the staff finds that the applicant has adequately described materials used in the HI-TRAC CS.

7.3.7.1.3 Summary of dose rate calculation for HI-TRAC CS Transfer Cask

The applicant calculated the dose rate in near field around the HI-TRAC CS transfer cask for normal, off-normal, and accident conditions of operations. The applicant provided the calculated results in SAR Table 7.4.1, “Dose Rates from the HI-TRAC CS MPC-37 Design Basis Fuel.”

The staff reviewed the information in the SAR for dose rate from the HI-TRAC CS transfer cask and finds that the applicant has provided a conservative shielding calculation and the information meets the acceptance criteria of NUREG-1567. On this basis, the staff determined that the applicant’s shielding calculations for the HI-TRAC CS are acceptable.

7.3.7.2 Shielding Calculations for the VVM Storage Module

7.3.7.2.1 Shielding model description for the VVM storage module

The applicant developed two shielding models for the HI-STORE CIS Facility, including a detailed VVM storage module containing 500 canisters in a 25 by 20 array.

The shielding model for the VVM includes detailed representation of the Version C closure lid. The applicant provided a graphic view of the MCNP model for the VVM closure lid model in SAR Figure 7.4.2, “Dose Locations for the HI-STORM UMAX Version C” (proprietary). In SAR table 7.4.2, the applicant provided the calculated dose rates at the locations identified in figure 7.4.2.

The staff reviewed the models described in the boundary dose calculation to examine the consistency of the model with the licensing drawing of the VVM and the ISFSI. On this basis, the staff finds that the shielding calculations performed by the applicant for the VVM and the ISFSI are acceptable.

7.3.7.2.2 Material properties for the VVM storage module

The applicant provided the material compositions and densities for the VVM model in SAR table 7.3.1. In SAR section 7.3.1, the applicant further pointed to table 5.3.2 of the FSAR for the UMAX system design for additional information on the properties of materials used in the HI-STORE CIS Facility.

The staff reviewed the descriptions of the materials in the SAR and the referenced UMAX FSAR and finds that they are consistent with the bill of materials as defined in the licensing drawings and used in the shielding model for calculating the dose rate at the key locations of the top of the VVM and the contribution to dose by a real individual at the controlled boundary. The staff also finds that the properties of the materials are, with one exception, consistent with the material specifications in PNNL-15870, “Compendium of Material Composition Data for Radiation Transport Modeling,” Revision 1, dated March 4, 2011, for commonly used shielding materials. The only exception is the controlled low-strength material (CLSM), which the applicant defined as a self-compacted, cementitious material used primarily as a backfill in place of compacted fill. The applicant stated that many terms are currently used to describe this material, such as flowable fill, unshrinkable fill, controlled density fill, flowable mortar, flowable fly ash, fly ash slurry, plastic soil-cement, and soil-cement slurry (as used in American Concrete Institute (ACI) 229R-99, “Controlled Low-Strength Materials”). CLSM and lean concrete are also referred to as “self-hardening engineered subgrade.” The staff reviewed the material composition used in the shielding model for the VVM and found that it is conservative compared to all the potential choices for the backfill materials.

In the shielding model, the applicant used soil instead of the CLSM. The staff reviewed the material property of the CLSM and finds that using soil is more conservative because soil has less shielding capacity for both neutron and gamma radiation.

In addition, the applicant performed analyses to demonstrate that there will be no significant changes in the CLSM material property that may affect its shielding function as a backfill material between the adjacent VVMs over the licensed period of storage operations. The staff’s materials review in section 17.3.8 of this SER, concluded that this material would have no significant changes in its material properties that could result in loss of its shielding function. Based on the conclusion from the materials review documented in chapter 17 of this SER, the staff determined that the applicant has adequately described materials used in the VVM model. On this basis, the staff finds that the materials used in the shielding model for the VVM are conservative and meet the recommendations of NUREG-1567 and therefore are acceptable.

7.3.7.2.3 Summary of dose rate calculation for the VVM storage module

The applicant summarized the dose rate calculation for the VVM storage module in SAR table 7.4.2, “Dose Rates Adjacent to and 1 Meter from the HI-STORM UMAX Module for Normal

Conditions, MPC-37 Design Basis Zircaloy Clad Fuel with Bounding Source Terms.” The applicant provided the dose rates at the important locations around the closure lid of the VVM storage module. These locations include five detectors:

(1) Detector No. 1 is at the side of the closure lid where the closure lid has the thinnest shielding material.

(2) Detector No. 2 is at the opening of the vent port, which is aligned with the opening of the potential streaming paths.

(3) Detector No. 3 is at the top of the closure lid that is directly aligned with the gap between the canister and the inner wall of the storage well.

(4) Detector No. 4 is directly located at the top of the vent cover.

(5) Detector No. 5 is at the side of the closure lid, but it is aligned with the interface between the closure lid and the storage well top flange.

The applicant also provided dose rate information in SAR table 7.4.2.

In addition, the applicant presented dose rate information in SAR Table 7.4.3, “Dose Rates as a Function of Distance from 500 Loaded HI-STORM UMAX VVMs for Fuel Assemblies with Bounding Source Terms from Reference [7.1.1].”

The staff also finds that the burnup profiles have been applied properly for the gamma and neutron sources in their respective transport calculations. [

]. This feature allows for implicit inclusion of the neutrons produced by fission reactions when the neutrons traverse the fuel and react with the fissile materials.

In addition, the applicant developed neutron transport models that include MODE N, P card, and separate tallies for both neutron and gamma radiation. With this approach, the applicant obtained the dose rates from both the neutrons and secondary gammas that are produced by (n, gamma) reactions when the neutrons from the SNF react with the fuel or structural materials.

The staff reviewed the input and output files and finds that the models have properly considered all gamma and neutron sources, which include primary and secondary neutrons and gammas from the fuel. The shielding models also properly treat the gammas produced by the activated fuel hardware. On these bases, the staff determined that the applicant has accurately modeled the VVM storage cask and the ISFSI pad loaded with 500 casks for calculations of the dose at the controlled area boundary and dose rates near the casks. The staff finds that the applicant’s modeling approach and calculations meet the acceptance criteria of NUREG-1567 and are therefore acceptable.

7.3.7.3 Dose Rate from Carbon-14 Produced in the VVM Storage Module

The applicant included the carbon-14 contribution to the dose of the real individual on or beyond the controlled area boundary and the occupational workers at the site. Section 7.3.4.2 of this SER documents details of the staff's evaluation of the carbon-14 dose rate calculation.

7.3.7.4 Dose Rate from HI-STAR 190 Package

Based on the design and the TS, the HI-STORE CIS Facility will accept only the MPC-37 and MPC-89 canisters that are transported to the site using the HI-STAR 190 transportation package. By design, the HI-STAR 190 package may be placed at a staging area inside the controlled area boundary before final acceptance inspection and unloading. The applicant recognized that the radiation from the HI-STAR 190 package that remains in the staging area will also contribute to the dose at the controlled area boundary. To account for this radiation, the applicant performed a shielding calculation for the HI-STAR 190 package to estimate the contribution to dose received by a real individual at the controlled area boundary during receipt, inspection, and unloading operations.

In SAR table 7.0.1, the applicant incorporated by reference the HI-STAR 190 SAR for the shielding model and estimated dose for the authorized contents that produce bounding dose rates. The applicant provided a calculation package, “Dose Versus Distance Calculations for the HI-STAR 190 at the HI-STORE CIS Facility” (proprietary), dated November 26, 2020, that estimates the contribution to the total annual dose to a real individual at the controlled area boundary. The applicant calculated the dose rates using the MCNP computer code and the associated cross section library. In table 8-1 and table 8-2 of the calculation package, the applicant provided the dose rate as a function of distance from the edge of the cask to the controlled area boundary for a single HI-STAR 190 package containing the MPC-37 and MPC-89 canister, respectively. In SAR table 7.4.7, the applicant provided an annual dose versus distance curve for an estimated staging time of 2,000 hours per year.

The staff reviewed the assumptions used by the applicant for this dose rate calculation and determined that they are appropriate and acceptable. The 2,000 hours of operation time for reception and unloading of the HI-STAR 190 package approximately represent the full-time annual working time for a typical worker. Therefore, the staff determined that the assumption is conservative and acceptable.

The source terms, model, and material properties used in the dose rate calculations are identical to those used in Revision 3 of the HI-STAR 190 SAR. Because the safety analyses for the HI-STAR 190 package have been reviewed and accepted by the NRC staff in the review and certification of the package as documented in section 5 of the SER for the HI-STAR 190 package dated August 8, 2017, the staff did not perform any further detailed technical review of the SAR for the HI-STAR 190. Because the requirements for radiation protection for the HI-STAR 190 package are more stringent (i.e., the dose rate at 2 meters (6.6. feet) from the surface of the package is limited to 10 mrem/hour), the staff determined that the contribution from the HI-STAR 190 to the total dose at the controlled area boundary is insignificant because the radiation level decreases by the square of the distance. The staff finds that the estimated dose rate contribution from the HI-STAR 190 package to the HI-STORE CIS Facility is conservative and acceptable.

In the calculation package “Dose Versus Distance Calculations for the HI-STAR 190 at the HI-STORE CIS Facility” (proprietary), the applicant summarized, in table 8-1 and table 8-2, the dose rate as a function of distance from the package, with a maximum of 1,000 meters (3,280 feet). The staff reviewed the calculated dose rates and finds that the results are reasonable and consistent with the results presented in the SAR for the HI-STAR 190 package. On this basis, the staff determined that the calculations for the HI-STAR 190 dose contribution to the HI-STORE CIS Facility doses are acceptable. The staff considered the results in its evaluation of the design of the radiation protection program, documented in SER chapter 11.

7.3.8 Summary of Shielding Calculations

The applicant summarized its shielding calculations in SAR table 7.4.7, which provides a total estimated annual dose to a real individual at the controlled area boundary of the HI-STORE CIS Facility. This dose includes the contributions to the annual dose to a real individual from the fully loaded 500 MPCs, the annual dose from ten HI-STAR 190 packages at the staging area, two MPCs in the HI-TRAC CS transfer casks or HI-STAR 190 packages in the CTB, and the dose from the carbon-14 that is generated in the air passing through the gap between the VVM and the canister.

The staff reviewed the summary of the shielding calculations in the SAR and finds that the applicant has included contributions from all potential radiation sources. Since there are no fuel facilities or other manmade radiation sources, the staff considers the applicant’s summary of shielding calculation results to be complete and acceptable. In addition, it is conservative to assume that the ISFSI is loaded with all 500 canisters at once because the decay times of the sources during the loading process are not credited. Completion of loading the 500 canisters into the ISFSI probably would take several years. On these bases, the staff finds that the applicant’s summary of shielding calculations meets the acceptance criterion and thus is acceptable.

7.3.9 Shielding Analyses for Accident Conditions

The applicant analyzed the potential accident scenarios that will impact the shielding function of the HI-STORE CIS Facility. In SAR section 7.4.2.2, the applicant identified the missile impact during construction next to a loaded canister and fire with the transfer cask as the plausible accidents that would impede the shielding function.

7.3.9.1 Tornado and Tornado-Borne Missile

The applicant identified a tornado and tornado-borne missile impact during construction next to a loaded canister as the only off-normal or accident condition applicable to the HI-STORM UMAX storage system under storage operations. The applicant previously analyzed this accident condition and presented the results in sections 5.1 and 5.3 of the HI-STORM UMAX FSAR, Revision 3. SAR table 15.0.1 incorporates by reference the off-normal and accident information from chapter 12 of that FSAR. The evaluation of this missile impact event, which the applicant identified as the bounding accident case affecting the UMAX VVM from a shielding perspective, shows that the potential dose meets the regulatory dose limits in 10 CFR 72.106(b). Therefore, the staff finds this acceptable.

7.3.9.2 Fire Accident

In SAR sections 7.4.2.2 and 15.3.1, the applicant discussed its evaluation of a fire accident condition for the HI-TRAC CS in which some areas of the cask could heat up significantly, causing these areas to exceed concrete temperature limits and experience degradation in shielding function. The applicant performed shielding calculations with models that assume that the density of the HI-TRAC CS concrete decreases approximately 21 percent as shown in SAR table 7.3.1. The resulting areal density (in grams per square centimeter) of the concrete shielding assumed for accident conditions is conservatively lower than the areal density of the post thermal accident remaining concrete shielding, considering degradation, as described in the calculation package Holtec Report No. HI-2177553, “Thermal Analysis of HI-TRAC CS Transfer Cask, Rev. 1” (proprietary), dated November 20, 2020.

The applicant calculated the dose rate at 100 meters (328 feet) from the HI-TRAC CS with an assumption that the accident would last 30 days. The applicant presented the results of the HI-TRAC CS accident in SAR table 7.4.4.

The staff reviewed the calculations and assumptions used in the dose calculations for the HI-TRAC CS under accident conditions. The staff finds that the chosen dose detector at a location 100 meters (328 feet) from the transfer cask under accident condition is acceptable because the closest distance from the CTB to the controlled area boundary is 99 meters (325 feet), and the CTB wall and structure will provide additional shielding in case an accident with the HI-TRAC CS occurs in the CTB.

7.3.9.3 Tipover

In SAR section 15.3.13, the applicant stated that tipover of the HI-TRAC CS is not a credible event and therefore did not perform a shielding analysis of this event. The NRC accepted this conclusion in its review and certification of the UMAX system design as documented in its SER issued on April 2, 2015. Specifically, section 3.4.3.1 of the SER states the following:

> The MPCs were evaluated by the applicant for a non-mechanistic tipover in CoC No. 1032. The staff finds the applicant’s evaluation that non-mechanistic tipover is not a credible event acceptable in that evaluation. No additional evaluation was necessary for the UMAX because UMAX is an inground storage system rendering non-mechanistic tipover not possible.

Based on this conclusion, the staff did not further review the dose resulting from a tipover event.

7.3.9.4 Drop Impact

In SAR section 15.3.14, the applicant stated that accident drops involving the HI-STAR 190 or the HI-TRAC CS are not credible as heavy load handling requires redundant drop protection as described in SAR sections 4.5.1, 4.5.2, and 4.5.3. Based on the structural evaluation, the staff concludes that the redundant drop protection of the HI-STAR 190 and HI-TRAC CS provides reasonable assurance that a drop event would not occur for the HI-STAR 190 package or the HI-TRAC CS transfer cask. Based on this conclusion, the staff determined that a shielding analysis for the drop event scenario is not necessary. Additionally, any accident conditions during and before unloading the canister from the HI-STAR 190 package are bounded by the regulations in 10 CFR 71.51(a)(2), and the dose rate limit under this regulation is more stringent. For these reasons, the staff did not further evaluate the dose from design accidents for storage operations.

7.3.9.5 Explosion

The applicant performed an accident analysis for a hypothetical explosion of potential hazardous cargo transported along ground transportation routes (roadways and railcars) near the proposed HI-STORE CIS Facility. The applicant performed this analysis based on the method outlined in Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,” Revision 2, issued April 2013, to establish the bounding overpressure from the potential hazardous cargo transported along ground transportation routes. Regulatory Guide 1.91 contains guidance on evaluating the effects of explosions on nuclear power plant structures. The applicant provided the details of its hazard analysis in Holtec Report No. HI-2200797, “Hazards Assessment of Ground Transportation Routes near HI-STORE” (proprietary), Revision 1, dated August 7, 2020. The staff documents its review of this hazard analysis in the accident analysis chapter of this SER, in section 15.3.2.8.

7.3.9.6 Summary of Shielding Analyses for Accident Conditions

The staff reviewed the shielding calculations for the various stages of operations and finds that the applicant has considered all practical normal and off-normal operating conditions. On this basis, the staff finds that the applicant’s consideration of operating conditions is complete and acceptable.

7.3.10 Staff Evaluation

The staff reviewed the shielding analyses provided in the SAR, the applicant’s responses to the staff requests for additional information, and the applicant’s proposed TS. The staff notes that the system contents result in relatively significant direct radiation dose rates, which is a concern primarily for operations involving the transfer cask (i.e., loading, unloading, and transport) for the HI-STORE CIS Facility system. The additional ALARA precautions to be implemented in the HI-STORE CIS Facility operations to minimize doses to personnel demonstrate compliance with public dose limits in 10 CFR Part 20 and 10 CFR Part 72.

The staff reviewed the accident evaluation and finds it to be acceptable for the design changes requested in the application. The staff has reasonable assurance that the direct radiation from the HI-STORE CIS Facility satisfies the regulatory requirements of 10 CFR 72.106(b) at and beyond a controlled boundary of 100 meters (328 feet) from the design-basis accidents.

The staff also reviewed the shielding analyses for the ISFSI under accident conditions. Based on its review, the staff determined with reasonable assurance that the HI-STORE CIS Facility shielding design meets the regulatory requirements of 10 CFR 72.106.

7.3.11 Confirmatory Analyses

The staff, with technical support from scientists at the Oak Ridge National Laboratory, performed confirmatory analyses on the source terms, dose rate from the HI-TRAC CS, dose rate from the VVM storage module, dose rate from carbon-14 generated at the ISFSI, dose rate from the HI-STAR 190 package, and the annual dose of a real individual at and beyond the controlled area boundary. The staff also performed confirmatory calculation for dose of the HI-TRAC CS under design-basis accident conditions. The results of the confirmatory calculations confirmed that the calculated doses meet the regulatory limits as prescribed in 10 CFR 72.104 and 10 CFR 72.106 with significant safety margins for the ISFSI operations. Based in part on the results of the Oak Ridge confirmatory results, the staff determined that there is reasonable assurance that the HI-STORE CIS Facility meet the regulatory requirements of 10 CFR 72.104 and 10 CFR 72.106.

7.4 Evaluation Findings

Based on the NRC staff's review of information in the HI-STORE CIS Facility application, the applicant's responses to staff requests for additional information, the TS, and operating limits, the staff finds with reasonable assurance the following:

- The SAR specifies the radioactive materials to be stored at the proposed facility in sufficient detail that adequately defines the allowed materials and permits evaluation of the facility shielding design for the proposed materials. The SAR includes analyses that are adequately bounding for the radiation source terms associated with the proposed materials' specifications. (10 CFR 72.24(c) and 10 CFR 72.120(b))

- The SAR describes the facility SSCs, including those that are important to safety that are relied on for shielding, in sufficient detail to allow evaluation of their effectiveness for the proposed license term. The descriptions include design criteria and design bases for the design, fabrication, construction, and performance requirements of SSCs important to safety. (10 CFR 72.24(b), 10 CFR 72.24(c), 10 CFR 72.120(a), and 10 CFR 72.120(b))

- The facility design includes SSCs and features to shield personnel from radiation exposure to meet 10 CFR 72.126(a)(6) and for radiation protection under normal and accident conditions to meet 10 CFR 72.128(a)(2). The radiation protection review, documented in SER chapter 11, describes the evaluation of the suitability of the shielding to perform these functions.

- The SAR provides reasonable and appropriate information, including dose rates, to allow evaluation of the facility's compliance with 10 CFR 72.24(e). The radiation protection review, documented in SER chapter 11, describes this evaluation.

- The SAR provides reasonable and appropriate information, including dose rates, to enable performance of the evaluations required in 10 CFR 72.24(m) and to allow evaluation of the facility's ability to meet the radiation protection requirements for members of the public in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20. This information includes impacts to shielding and dose rates to support evaluations of compliance with the requirements in 10 CFR 72.122(b)(2)(i), 10 CFR 72.122(c), and 10 CFR 72.122(e). The radiation protection review documented in SER chapter 11 describes these evaluations.

The applicant's evaluation includes appropriate shielding analyses for the configurations of facility SSCs and features that exist during the different stages of storage operations, including the impacts of normal, off-normal, and accident conditions. The evaluation includes dose rates that support the facility's ability to meet the radiation protection requirements in 10 CFR 72.104, 10 CFR 72.106, and 10 CFR Part 20, including doses to members of the public and occupational doses estimated to result from facility operations. The applicant adequately considered and incorporated ALARA principles into the facility design and operations. For these reasons, the staff finds reasonable assurance that the design features relied on for shielding for the HI-STORE CIS Facility have been adequately identified and evaluated. The staff reached this finding on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, the statements and representations in the SAR, and the staff's confirmatory analyses.

7.5 References

American Concrete Institute (ACI) 229R 99, "Controlled Low-Strength Materials," 1999.

American National Standards Institute (ANSI/)/American Nuclear Society (ANS) 6.1.1, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," 1997.

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection against Radiation."

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8 CRITICALITY SAFETY EVALUATION

Holtec International (the applicant) applied for a license to construct and operate an independent spent fuel storage installation (ISFSI), the HI-STORE Consolidated Interim Storage (CIS) Facility, for the storage of commercial spent nuclear fuel pursuant to the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." The applicant described the criticality safety design features of and provided criticality safety for the proposed HI-STORE CIS Facility in chapter 8, "Criticality Evaluation" of the facility safety analysis report (SAR), Revision 0T, dated January 20, 2023.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information to verify that the HI-STORE CIS Facility will remain subcritical under normal, off-normal, and design-basis accident conditions during receipt, handling, transfer, and storage operations involving spent nuclear fuel. In its review, the staff followed the guidance in NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000. The following sections of this safety evaluation report (SER) document the staff's technical review of and conclusions about the criticality safety design of the HI-STORE CIS Facility.

8.1 Scope of the Review

The objective of the staff's criticality safety review is to verify that the criticality safety design of the HI-STORE CIS Facility meets the regulatory requirements of 10 CFR 72.124, "Criteria for nuclear criticality safety." The staff evaluated the applicant's criticality safety analysis for the facility by reviewing the information provided in the SAR, the documents incorporated by reference pursuant to 10 CFR 72.18, "Elimination of repetition," the applicant's responses to the staff's requests for additional information, and the staff's previous evaluations for the canisters and their contents proposed to be stored at the facility, with consideration of the limiting conditions as given in the proposed technical specifications (TS) for the HI-STORE CIS Facility, dated November 23, 2022.

8.2 Regulatory Requirements

The applicant identified the following regulatory requirements as applicable to the criticality safety of the proposed HI-STORE CIS Facility:

- 10 CFR 72.24, "Contents of application: Technical information"
- 10 CFR 72.40, "Issuance of license"
- 10 CFR 72.44, "License conditions"
- 10 CFR 72.124

8.3 Staff Review and Analysis

The staff reviewed the information provided by the applicant concerning the criticality safety design of the HI-STORE CIS Facility. The information includes the stored spent fuel specifications, criticality safety design features, analytical means, and the criticality analysis documents that the applicant incorporated by reference.

The HI-STORE CIS Facility is an in-ground spent fuel storage system. Its design is based primarily on the in-ground spent fuel storage system design of the HI-STORM UMAX canister storage system, as certified in Certificate of Compliance (CoC) No. 1040, Amendment 2, dated January 6, 2017. SAR chapter 8 discusses the criticality safety features of the facility. In SAR chapter 15, “Accident Analysis,” the applicant discussed the effects of off-normal and accident scenarios at the HI-STORE CIS Facility.

The applicant did not perform separate criticality safety analyses for the HI-STORE CIS Facility. Instead, in SAR table 8.0.1, “Material Incorporated by Reference,” the applicant incorporated by reference the criticality safety analyses for the HI-STAR 190 transportation package SAR, Revision 3, dated November 2, 2018; and the HI-STORM UMAX canister storage system final safety analysis report (FSAR), Revision 3, dated June 29, 2016, as permitted by 10 CFR 72.18. The criticality safety analyses in the HI-STORM UMAX FSAR further incorporate by reference the criticality safety analysis for canisters as described in the HI-STORM flood/wind (FW) multipurpose canister (MPC) storage system FSAR, Revision 4, dated June 24, 2015.

In addition, HI-STORE CIS Facility SAR section 4.0, “Introduction,” states the following:

> Unlike the generic HI-STORM UMAX system, the Short-Term Operations at the HI-STORE facility do not involve any activity related to loading fuel into canisters: the canisters arrive at the HI-STORE CIS facility in the HI-STAR 190 (NRC docket #71-9373). The short term operations begin at the point the transport package is received at the site and end at the point the canister is placed in a HI-STORM VVM for interim storage.

Further, the applicant’s proposed TS 2.1, “Approved Contents, Fuel Specifications and Loading Conditions,” requires that the fuel assemblies and canisters to be stored at the ISFSI meet certain conditions specified in the HI-STAR 190 transportation package SAR, Revision 3.

The staff reviewed the information provided in the SAR, the previously approved HI-STORM UMAX FSAR, the supporting criticality safety analysis for the canisters documented in the FSAR for the HI-STORM FW system, the SAR for the HI-STAR 190 transportation package design, the applicant’s responses to the staff’s requests for additional information, and the limiting conditions as specified in the proposed TS for the HI-STORE CIS Facility. Based on its review, the staff determined that the applicant has provided sufficient information for the staff to perform a detailed review, and the application meets the regulatory requirement of 10 CFR 72.24.

8.3.1 Authorized Contents

Based on the description in SAR table 1.0.1, “Overview of the HI-STORE Facility,” the HI-STORE CIS Facility will be built in 20 stages. Each stage of the facility will have a capacity of 8,680 metric tons of uranium in the form of commercial spent fuel in up to 500 welded canisters. The current application and this review cover only the first stage. Proposed TS 2.1 states that pressurized water reactor (PWR) and boiling-water reactor (BWR) spent fuels must be stored in

MPC-37 and MPC-89 canisters, respectively. Therefore, no mixed load of these fuel types is authorized in these canisters.

In addition, proposed TS 2.1 allows the storage of only intact spent PWR and BWR fuel assemblies loaded in MPC-37 and MPC-89 canisters, respectively. TS 2.1 further states that these fuel assemblies must also meet the restrictions on burnup, enrichment, and cooling time as specified in table 7.C.8 and table 7.C.10 of the HI-STAR 190 SAR, Revision 3. As such, only canisters (MPC-37 and MCP-89) and their contents meeting all these conditions can be stored in the HI-STORE CIS Facility. Proposed TS table 2-1, “Loading Patterns for MPC-37 (PWR Fuel Assembly,” and table 2-2, “Loading Patterns for MPC-89 (BWR Fuel Assembly),” further define the allowable loading patterns. Because the fuel characteristics specified in TS 2.1 are more limiting than the fuel characteristics specified in the HI-STORM UMAX CoC, which are based on the fuel specifications provided in the HI-STORM FW FSAR, the allowable contents for the HI-STORE CIS Facility are only a subset of the spent fuel authorized for storage as specified in the CoC for the HI-STORM UMAX system. SAR section 4.1, “Materials to be Stored,” describes the spent fuel to be stored in the HI-STORE CIS Facility. The staff reviewed and compared the descriptions of the authorized contents in the SAR and the TS. The staff finds that the authorized contents are consistent with the criticality safety analyses provided in the HI-STORM FW FSAR, Revision 4, and the HI-STAR 190 SAR, Revision 3. On these bases, the staff finds that the applicant has provided a detailed description of the authorized contents sufficient for the staff to verify the criticality safety design of the HI-STORE CIS Facility and therefore meets the regulatory requirements of 10 CFR 72.24(b), as well as 10 CFR 72.44(a), which states, “Each license issued under this part shall include license conditions. The license conditions may be derived from the analyses and evaluations included in the Safety Analysis Report and amendments thereto submitted pursuant to § 72.24. License conditions pertain to design, construction and operation.”

8.3.2 Description of the Criticality Safety Design Features

With respect to the criticality safety of a dry storage system, 10 CFR 72.124(b) states the following:

> When practicable, the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

As described above, according to the proposed TS for the HI-STORE CIS Facility, only MPC-37 and MPC-89 canisters containing PWR and BWR spent fuel, respectively, can be stored at the HI-STORE CIS Facility. SAR table 1.0.5, “Transport Casks Allowed for Receipt of Canisters at HI-STORE,” also requires that the MPCs be transported to the ISFSI only using the HI-STAR 190 packaging system, and TS 2.1 requires that spent fuel received in MPC-37 and MPC-89 canisters meets the limits defined in chapter 7 of SAR Revision 3 for the HI-STAR 190 package. In accordance with chapter 6 of the FSAR for the HI-STORM FW storage system and chapter 6 of the SAR for the HI-STAR 190 package, the MPC-37 and MPC-89 canisters ensure criticality safety using the combination of a limited quantity of fissile materials, fixed neutron poison plates in the canisters, spaces between the fuel assemblies in the canisters, and spaces between the canisters in the ventilated vertical module (VVM) storage configurations. Thus, comparing the criticality safety design features applied to spent fuel at the HI-STORE CIS Facility with the regulatory requirements of 10 CFR 72.124(b), the staff finds that criticality safety design of the HI-STORE CIS Facility is consistent with the regulatory requirements for spent fuel dry storage system criticality safety design. On these bases, the staff determined that the criticality design features meet the regulatory requirements and therefore are acceptable.

8.3.3 Criticality Safety Analysis

The applicant discussed the criticality safety design in SAR chapter 8. With respect to compliance with 10 CFR 72.124(a), SAR table 1.0.5 requires that the MPCs must be transported using the HI-STAR 190 transportation packaging system, and proposed TS 2.1 specifies that the spent fuel must meet the restrictions specified in table 7.C.8 and table 7.C.10 of the HI-STAR 190 SAR, Revision 3.

The criticality safety analysis presented in chapter 6 of the HI-STAR 190 SAR demonstrates that an infinite array of packages flooded with fresh water will remain subcritical as required by 10 CFR 71.55(b), 10 CR 71.55(d) (under normal conditions of transport), and 10 CFR 71.55(e) (under hypothetical accident conditions), respectively. Additionally, the staff reviewed the operations described in SAR chapter 3, “Operations at the HI-STORE CIS Facility,” and determined with reasonable assurance that, at the HI-STORE CIS Facility, there would not be any natural phenomena or manmade events that could move multiple canisters together to form a closely contacted hexagonal array similar to the infinite array of fresh-water-flooded canisters described in the HI-STAR 190 SAR. On this basis, the staff determined that the HI-STORE CIS Facility meets the regulatory requirements of 10 CFR 72.124(a), also known as the double contingency requirement for criticality safety (i.e., “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”). Therefore, the HI-STORE CIS Facility storing 500 canisters in a square array of VVMs will remain subcritical under normal, off-normal, and accident conditions of operations. The criticality analyses presented in the HI-STAR 190 SAR present a bounding scenario for the criticality safety of the HI-STORE CIS Facility. Detailed discussions of the staff’s assessments of criticality safety for the HI-STAR 190 transportation package appear in the staff’s related SER, “Safety Evaluation Report, Docket No. 71-9373, Model No. HI-STAR 190 Package, Certificate of Compliance No. 9373, Revision No. 0,” dated August 8, 2017.

SAR section 8.5, “Criticality Monitoring,” states that a criticality monitoring system is not required because the spent fuel remains packaged in its stored configuration and the canisters will not be opened at the facility. This is consistent with the acceptance criteria of NUREG-1567 because (1) the canisters will typically remain dry inside, (2) the applicant has demonstrated that an infinite number of canisters flooded with fresh water will remain subcritical, and (3) the facility has only a limited number of canisters. Therefore, a criticality accident at the HI-STORE CIS

Facility is not credible. For these reasons, the staff determined that a criticality monitoring system is not necessary, and the facility meets the requirements of 10 CFR 72.124(c).

The staff verified that the applicant has clearly identified and adequately described the design features that are important to criticality safety of the ISFSI. The staff finds that the design features are based on a favorable geometry, neutron poison plates installed in the fuel basket, and burnup credit to meet the requirements of 10 CFR 72.124(a) and (b). As discussed in SER section 8.3.3.2.2, the applicant has demonstrated that the neutron multiplication factor, k_{eff}, for the ISFSI is below 0.95 with all biases and bias uncertainties included. This is consistent with the acceptance criterion for criticality safety as provided in NUREG-1567. On these bases, the staff determined that there is reasonable assurance that the criticality safety design of the HI-STORE CIS Facility meets the regulatory requirements of 10 CFR 72.124.

8.3.3.1 Safety Analysis Method

8.3.3.1.1 Model Configuration

This section describes the staff's evaluation of the analytical means used by the applicant to demonstrate that the spent nuclear fuel stored at the HI-STORE CIS Facility will remain subcritical under normal, off-normal, and design-basis accident conditions of operations.

In SAR section 8.3.1, "Model Configuration," the applicant stated that the model configuration, including material properties, for the criticality analysis is incorporated by reference from section 6.3 of the HI-STORM FW system FSAR, Revision 4. In addition, SAR table 1.0.4, "Canisters Allowed for Storage in HI-STORM UMAX at HI-STORE," and table 1.0.5 require that all spent fuel to be stored in sealed MPC-37 or MCP-89 canisters be transported to the facility using the HI-STAR 190 package. Therefore, the spent fuel canisters received at the site are also subject to the CoC for the HI-STAR 190 package, which includes further restrictions to the fuel to be qualified for transportation. Tables 7.C.8 and 7.C.10 in the HI-STAR 190 SAR, Revision 3, which the applicant incorporated into proposed TS 2.1, provide detailed specifications for the additional restrictions on the allowable contents. Because all spent fuel canisters at the HI-STORE CIS Facility are received in the HI-STAR 190 transport casks, the limiting conditions specified in tables 7.C.8 and 7.C.10 of the HI-STAR 190 SAR apply to fuel stored at the HI-STORE CIS Facility.

As described in section 7.0 of the HI-STAR 190 SAR, MPCs loaded into HI-STAR 190 transportation packages must meet the loading requirements for the HI-STORM UMAX and HI-STORM FW storage systems, plus meet additional content requirements, resulting in the most bounding content limit among the three systems. Because the criticality analysis for the HI-STAR 190 package bounds the criticality analyses for the HI-STORM UMAX and HI-STORM FW storage systems for the authorized contents, the staff focused its review on the criticality safety analyses provided in the SAR for the HI-STAR 190 package. Chapter 6 of the HI-STAR 190 SAR provides the criticality safety analyses for this package. As stated in chapter 6, the analysis considered a single package and an infinite array of packages under normal conditions of transport and hypothetical accident conditions as prescribed in 10 CFR 71.71, "Normal conditions of transport," and 10 CFR 71.73, "Hypothetical accident conditions," respectively. The staff reviewed the accident analysis for the HI-STORE CIS Facility described in SAR chapter 15 and determined that there are no site-specific conditions that could result in an infinite number of canisters in a closely packed hexagonal array as analyzed for the HI-STAR 190 package. Also, a canister in the HI-STORE CIS Facility would never experience the impact of a 30-foot drop during operations as addressed by 10 CFR 71.73(c)(1). On these bases, the staff determined that piling up an infinite array of canisters is not a credible event at the HI-STORE CIS Facility. As such, the staff determined that the accident scenarios analyzed in chapter 6 of the HI-STAR 190 SAR for criticality safety bound all credible normal, off-normal, and accident conditions of the HI-STORE CIS Facility operations described in SAR chapters 3 and 15. On these bases, the staff finds that the HI-STORE CIS Facility criticality safety design meets the regulatory requirements of 10 CFR 72.124(a).

8.3.3.1.2 Material Properties

SAR section 8.3.1 states that the model configuration, including material properties for the criticality analysis, is incorporated by reference from section 6.3 of the HI-STORM FW storage system FSAR and section 6.3.2 of the SAR for the HI-STAR 190 package. Details on the staff's evaluation of the material properties appear in section 7.3.2 of the staff's SER for the HI-STORM FW system, dated July 14, 2011, and section 6.3.2 of the SER for the HI-STAR 190 transportation package. On these bases, the staff determined that the applicant has provided detailed descriptions for the materials of construction for the ISFSI components related to criticality safety control and correctly used the material properties in the criticality safety analyses.

8.3.3.2 Criticality Safety Analysis

For criticality safety analyses for the HI-STORE CIS Facility, in SAR chapter 8, "Criticality Evaluation," the applicant incorporated by reference the criticality safety analyses for the HI-STORM FW system. The applicant also stated that a discussion of how these HI-STORM FW results apply to the HI-STORM UMAX system is incorporated by reference from section 6.2 of the HI-STORM UMAX canister storage system FSAR, Revision 3. Additionally, because the applicant incorporated by reference the criticality safety analyses described in chapter 6 of the SAR for the HI-STAR 190 package, the criticality safety analysis for the HI-STAR 190 transportation package, which the applicant uses to transport all spent fuel to the facility, applies to all spent fuel received at the facility. The applicant did not perform new criticality safety analyses for the HI-STORE CIS Facility. The staff reviewed the criticality safety analyses provided in chapter 8 of the HI-STORM FW system FSAR and chapter 6 of the HI-STAR 190 transportation package SAR. Based on its review, the staff concluded that no additional calculations are necessary for the facility because there are no site-specific conditions that would invalidate the conclusion of these criticality safety analyses when applied to the HI-STORE CIS Facility. The criticality safety analyses for the HI-STAR 190 transportation package bound all normal, off-normal, and design-basis accident conditions of operations of the HI-STORE CIS Facility. On these bases, the staff determined that the criticality safety analyses for the HI-STAR 190 package have demonstrated that the HI-STORE CIS Facility will remain subcritical under normal, off-normal, and all accident conditions of operations.

8.3.3.2.1 Computer Program

Because SAR table 1.0.5 requires spent fuel stored at the HI-STORE CIS Facility to be received in HI-STAR 190 transportation packages, the criticality safety analyses, including the computer code and cross section, for the HI-STAR 190 apply to spent fuel at the HI-STORE CIS Facility. The criticality safety analyses for the HI-STAR 190 transportation package used the three-dimensional Monte Carlo code MCNP with ENDF/B-VII. Section 6.3.3 of the HI-STAR 190 package SAR, Revision 3, states that MCNP and CASMO5 Version 2.00.00 are used for the criticality analyses of the HI-STAR. This SAR further states that MCNP design basis calculations used continuous energy cross-section data, based on ENDF/B-VII, as distributed with the code. The CASMO code is used to determine the spent fuel material composition that the MCNP code needs for burnup credit analysis. The staff reviewed section 6.8 of the SER for the HI-STAR 190 transportation package and finds that the SER has explicitly documented the bases for the approval of the burnup credit analyses for the HI-STAR 190 transportation package. Hence, the staff performed no further detailed review of the criticality safety analyses.

8.3.3.2.2 Neutron Multiplication Factor

The staff reviewed chapter 6 of the SAR for the HI-STAR 190 transportation package, which the applicant incorporated by reference, and the corresponding SER that documents the staff's review and conclusions. Based on its review, the staff confirmed that the k_{eff} of the HI-STORE CIS Facility loaded with allowed fuels will not exceed 0.95 under all normal, off-normal, and accident conditions. The criticality safety design of the facility meets the acceptance criteria for criticality safety as described in NUREG-1567, section 8.4.1.1, for assuring criticality safety of a specific ISFSI. On this basis, the staff finds that the HI-STORE CIS Facility design meets the regulatory requirements of 10 CFR 72.124.

8.3.3.2.3 Computer Code Benchmarking Analysis

In SAR table 8.0.1, the applicant incorporated by reference criticality safety analyses from the HI-STAR 190 transportation system SAR and the HI-STORM FW system FSAR and explained how the HI-STORM FW results apply to the HI-STORM UMAX system. Among the criticality safety analyses for these systems, the criticality safety analyses for HI-STAR 190 transportation package provide a bounding scenario for criticality safety. Therefore, the staff focused its review on the SAR and SER for the HI-STAR 190 criticality safety evaluation and computer code benchmarking analyses. Based on its review, the staff finds that the SER provided a detailed discussion of the bases for accepting the code benchmarking analyses, including verification of the benchmarking analyses for the MCNP5 and CASMO codes and cross sections of the HI-STAR 190 package, and there is no need to repeat the review.

8.3.3.2.4 NRC Independent Criticality Analysis

Based on its review, the staff determined that an independent criticality analysis with additional criticality calculations was not necessary for the HI-STORE CIS Facility. This is because the applicant incorporated by reference the calculations previously accepted in the criticality safety analysis of the HI-STAR 190 transportation package, and because the hypothetical accident conditions applied to the HI-STAR 190 package bound all credible site-specific factors that could impact the criticality safety of the HI-STORE CIS Facility. For this reason, the staff did not perform further independent criticality analysis.

8.3.3.3 Burnup Credit

The applicant's proposed TS 2.1 requires that fuel received onsite in MPC-37 and MPC-89 canisters meet the burnup, enrichment, and cooling time restrictions from tables 7.C.8 and 7.C.10 of the HI-STAR 190 SAR, Revision 3, that the staff has previously found acceptable. The licensee loading the HI-STAR 190 package must ensure that the correct fuel has been loaded. This requirement provides reasonable assurance that the correct fuel will be loaded in canisters received at the HI-STORE CIS Facility. On this basis, the staff found that the burnup credit analysis for the HI-STAR 190 remains valid for the HI-STORE CIS Facility.

8.4 Evaluation Findings

Based on its review of the SAR, the staff determined the following:

- The applicant has described the structures, systems, and components important to criticality safety in sufficient detail in the SAR to enable an evaluation of their effectiveness in accordance with 10 CFR 72.24(b) and 10 CFR 72.24(c).

- The design, procedures, and materials to be stored for the proposed HI-STORE CIS Facility provide reasonable assurance that the activities authorized by the license can be conducted without endangering public health and safety, in compliance with 10 CFR 72.40(a)(13) with respect to nuclear criticality safety.

- The proposed license conditions, including the TS, include those items necessary to ensure nuclear criticality safety in the design, fabrication, construction, and operation of the HI-STORE CIS Facility in accordance with the requirements of 10 CFR 72.24(g) and 10 CFR 72.44(c).

- The proposed design and operations of the HI-STORE CIS Facility (including handling, packaging, transfer, and storage of the radioactive materials to be stored) provide reasonable assurance that the materials will remain subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The SAR analyses adequately demonstrate that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with the uncertainties in the data and methods used in calculations. They also demonstrate safety for the handling, packaging, transfer, and storage of the stored materials under normal, off-normal, and accident conditions in compliance with 10 CFR 72.124(a) and (b).

- A criticality monitoring system is not required at the HI-STORE CIS Facility since the criticality safety analysis has demonstrated that the ISFSI will remain subcritical under all storage operations and hence is in compliance with 10 CFR 72.124(c).

8.5 References

Code of Federal Regulations, Title 10, Part 71, “Packaging and Transportation of Radioactive Materials.”

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NRC, “Safety Evaluation Report, Docket No. 71-9373, Model No. HI-STAR 190 Package, Certificate of Compliance No. 9373, Revision No. 0,” August 8, 2017. ML17222A083.

9 CONFINEMENT EVALUATION

In chapter 9, “Confinement Evaluation,” of the safety analysis report (SAR), Revision 0T, dated January 20, 2023, Holtec International (Holtec or the applicant) addressed the confinement criteria adopted for storage at the proposed HI-STORE Consolidated Interim Storage (CIS) Facility. In addition, the applicant provided details on the proposed spent nuclear fuel (SNF) to be stored at the site. The applicant summarized the system design features that ensure radiological releases are within limits and will remain as low as reasonably achievable, and that the SNF cladding and SNF assemblies are protected from degradation during storage.

9.1 Scope of Review

Unless otherwise stated, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed and evaluated the applicant’s confinement evaluation in SAR chapter 9 by examining the documents cited in or attached to the SAR and the applicant’s responses to the staff’s requests for supplemental information and requests for additional information (RAIs). The staff reviewed this information to ensure that radiological releases to the environment from the Holtec CIS Facility storage system will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures. To make its determination, the staff reviewed the application with regard to radionuclide confinement analysis, confinement monitoring, and protection of stored materials from degradation. Specifically, the staff reviewed the information in the application on confinement to evaluate (1) the applicant’s estimate of the amounts of radionuclides that would be released to the environment under normal operations, anticipated occurrences, and design-basis accident conditions, (2) the proposed monitoring systems, and (3) the systems for protecting stored materials from degradation.

9.2 Regulatory Requirements

The regulatory requirements relevant to the confinement evaluation of the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.44, “License conditions”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

9.3 Staff Review and Evaluation

This chapter addresses three review objectives. The first is to evaluate whether the facility design that includes the MPC-37 and MPC-89 canisters in the HI-STORM UMAX (Underground—MAXimum capacity) Canister Storage System, Docket No. 72-1040, meets regulatory performance standards for confinement given the applicant's estimates of the amount of radionuclides that would be released to the environment under (1) normal operations and anticipated occurrences and (2) design-basis accident conditions. The NRC considers estimates of releases, together with local environmental transport mechanisms (i.e., meteorology and hydrology) and distances to the controlled area boundary, to determine whether the design meets these regulatory performance standards. The staff's specific evaluations of the applicant's release estimates against the regulatory requirements in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," appear in chapter 11, "Radiation Protection Evaluation," and chapter 15, "Accident Evaluation," of this safety evaluation report (SER), in accordance with NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000.

The second review objective is the evaluation of the applicant's proposed monitoring systems. This evaluation includes monitoring systems for storage confinement and additional systems for measuring effluents during normal operations and accidents.

The third review objective is to evaluate systems to protect stored materials from degradation.

9.3.1 Review of Design Features

Section 1.0, "Introduction," of the SAR explains that this licensing submittal is limited to only multipurpose canisters (MPCs), specifically the MPC-37 and MPC-89 designs, which were previously certified in the HI-STORM UMAX Canister Storage System. Section 1.0 of the SAR also indicates that the aforementioned canisters are limited to only being received using the HI-STAR 190 package, Docket No. 71-9373, identified in SAR table 1.0.5, "Transport Casks Allowed for Receipt of Canisters at HI-STORE." SAR table 1.0.1, "Overview of the HI-STORE Facility," states that the type of storage system used at the site is the HI-STORM UMAX. The basis for the staff's approval of the HI-STORM UMAX system, including the MPC-37 and MPC-89 as used in the HI-STORM UMAX, appears in the initial issuance of the NRC SER for the HI-STORM UMAX Canister Storage System dated April 2, 2015. The staff reviewed the HI-STORE CIS Facility proposed license and verified that storage would be authorized only in casks designed in accordance with Certificate of Compliance (CoC) No. 1040, Amendment Nos. 0, 1, and 2, for the HI-STORM UMAX Canister Storage System and that appendix A, "Technical Specifications for the HI-STORE Consolidated Interim Storage (CIS) Facility," to the applicant's proposed Materials License No. SNM-1051 describes loading patterns and other technical specifications (TS) for the MPC-37 and MPC-89 designs only.

SAR section 9.0, "Introduction," states that the HI-STORM UMAX final safety analysis report (FSAR), Docket No. 72-1040, references the HI-STORM Flood/Wind (FW) system, Docket No. 72-1032. The staff verified that section 3.2.1.2, "Multi-Purpose Canisters," of the initial issuance of the HI-STORM UMAX SER states, "As described in the FSAR, the HI-STORM UMAX system utilizes two MPCs as confinement vessels: the MPC-37 for pressurized water reactor (PWR) fuel and the MPC-89 for boiling water reactor (BWR) fuel. These MPCs have been previously reviewed and approved for storage (CoC No. 1032) and all relevant evaluations are presented in the HI-STORM FW FSAR." Therefore, the staff notes that the HI-STORM UMAX docket references the HI-STORM FW docket, that the NRC initially evaluated and found acceptable the designs for the MPC-37 and MPC-89 for the HI-STORM FW, and that subsequently it evaluated and found acceptable those designs in a review of the HI-STORM UMAX Canister Storage System. The applicant referenced the HI-STORM FW in the HI-STORE CIS Facility SAR; however, the HI-STORE CIS Facility SAR only proposes the MPC-37 and MPC-89 in the HI-STORM UMAX Canister Storage System for use at the HI-STORE CIS Facility, whose safety evaluations are included in the HI-STORM FW FSAR, Holtec Report No. HI-2114830, Docket No. 72-1032, Revision 4, dated June 24, 2015. Relevant safety evaluations for the MPC-37 and MPC-89 canisters from the HI-STORM FW FSAR are applicable to the HI-STORE CIS Facility and are described below.

In SAR table 9.0.1, "Material Incorporated by Reference in this chapter," the applicant incorporated by reference chapter 7, "Confinement Evaluation," of the HI-STORM UMAX FSAR, Holtec Report No. HI-2115090, Docket No. 72-1040, Revision 3, dated June 29, 2016, which the NRC staff evaluated in the SERs for the HI-STORM UMAX initial issuance and Amendment Nos. 1 and 2. Therefore, based on the staff's evaluation in the HI-STORM UMAX SERs, the staff determined that the referenced information in chapter 7 of the HI-STORM UMAX FSAR, Revision 3, is applicable to the HI-STORE CIS Facility confinement evaluation and that this incorporation by reference is, therefore, acceptable. Also in SAR table 9.0.1, the applicant incorporated by reference chapter 7, "Confinement," of the HI-STORM FW FSAR, Revision 4, which the NRC staff evaluated in the SERs for the HI-STORM FW initial issuance and Amendment Nos. 1 and 2; therefore, based on the staff's evaluation in the HI-STORM FW SERs, the staff determined that the referenced information in chapter 7 of the HI-STORM FW FSAR, Revision 4, is applicable to the HI-STORE CIS Facility confinement evaluation and that this incorporation by reference is, therefore, acceptable. In SAR table 1.0.4, "Canisters Allowed for Storage in the HI-STORM UMAX at HI-STORE," the applicant specifically stated that only the MPC-37 and MPC-89 canisters are allowed for storage at the HI-STORE CIS Facility. Therefore, the staff's confinement evaluation is only for the MPC-37 and MPC-89 canisters in the HI-STORM UMAX at the HI-STORE CIS Facility.

The applicant stated, in SAR section 5.1.2, "Design Criteria," that the MPC enclosure vessel is designed and fabricated as a Class 1 pressure vessel in accordance with Section III, Subsection NB, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), 2007 Edition, with certain necessary alternatives, as discussed in section 2.2, "HI-STORM FW Design Loadings," of the HI-STORM FW FSAR. The HI-STORM FW FSAR, section 2.0.1, "MPC Design Criteria," states that the principal exception to the ASME Code pertains to the MPC lid, vent, and drain port cover plates and closure ring welds to the MPC lid and shell. The HI-STORM FW FSAR, lists ASME Code alternatives in table 2.2.14, "List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)." The staff also verified that section 3.1.2, "Structural Design Criteria," of the initial issuance of the HI-STORM FW SER states that, for the MPC, the principal design criteria are provided from its design as a Class 1 pressure vessel, in accordance with Section III, Subsection NB, of the ASME Code. The staff finds the use of ASME Code, Section III, Subsection NB, to be consistent with the guidance in chapter 9, "Confinement Evaluation," of NUREG-1567 and, the evaluation finding F5.1 in the initial issuance of the HI-STORM FW SER. Therefore, the use of ASME Code, Section III, Subsection NB, with the list of ASME Code alternatives in table 2.2.14 of the HI-STORM FW FSAR, is acceptable.

The applicant stated in SAR section 9.0 that only radioactive material in seal-welded canisters may be accepted and placed into storage at the HI-STORE CIS Facility. The staff found this to be consistent with what the applicant more specifically described in SAR section 9.0: (1) storage at the HI-STORE CIS Facility is limited to the canisters certified for storage in the HI-STORM UMAX, Docket No. 72-1040, (2) the HI-STORM UMAX FSAR references the HI-STORM FW, Docket No. 72-1032, and (3) the applicant's relevant evaluations of the canisters are presented in the HI-STORM FW FSAR. Furthermore, this licensing submittal and the staff's confinement evaluation are limited to only the MPC-37 and MPC-89 canisters in the HI-STORM UMAX.

The applicant described, in SAR section 9.2.1, "Storage Systems," how the confinement boundary of the HI-STORM UMAX includes the MPC shell, MPC baseplate, MPC lid, port cover plates, closure ring, and associated welds for each MPC-37 and MPC-89. Section 7.1.2, "Confinement Penetrations," of the HI-STORM FW FSAR, which the applicant incorporated by reference in SAR table 9.0.1, states that the closure ring provides redundant closure to the lid-to-shell weld and vent and drain port cover plate welds and satisfies 10 CFR 72.236(e). The staff verified that the confinement boundary components are depicted on SAR Drawing Nos. 6505 and 6512 (proprietary, see section 1.5, "Licensing Drawings") and that the confinement boundary components are listed on the drawings as important to safety (ITS).

The staff verified that table 2.2.14 of the HI-STORM FW FSAR describes how the MPC lid and closure ring welds are welded independently to provide a redundant seal, the closure ring also provides independent redundant closure for the vent and drain port cover plates, and the vent and drain port cover plate welds are helium leakage rate tested. The staff review of the storage systems documented in the SERs associated with the HI-STORM FW system and HI-STORM UMAX Canister Storage System respective CoCs and amendments were conducted using the regulatory requirements of 10 CFR 72 Subpart L, "Approval of Spent Fuel Storage Casks." Within 10 CFR 72 Subpart L, 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval," provides the requirements for dry storage systems that may be used by general licensees. The staff notes that it previously found in its review that the HI-STORM UMAX Canister Storage System with MPC-37 and MPC-89 canisters provides redundant sealing of the confinement system, there are no bolted closures or mechanical seals in the MPC confinement boundary, and, therefore, the requirement in 10 CFR 72.236(e) is met.

The applicant stated in SAR table 4.2.1, "ITS Classification of SSCs [Structures, Systems, and Components] that Comprise the HI-STORE CIS Facility," that the canisters (i.e., the MPC-37 and MPC-89 in the HI-STORM UMAX) provide leaktight confinement, and the canisters are ITS Category A. The staff verified that the source for the ITS determination is HI-STORM FW FSAR, table 2.0.2, "MPC 37 Enclosure Vessel (Drawing # 6505)," and table 2.0.6, "MPC 89 Enclosure Vessel (Drawing # 6512)," and that Drawing Nos. 6505 and 6512 of the HI-STORM FW FSAR also show that the confinement boundary components are ITS.

Based on its review described above, the staff finds that SAR chapter 9, together with chapter 7 of the HI-STORM UMAX Canister Storage System FSAR and chapter 7 of the HI-STORM FW FSAR, which are both incorporated by reference in the SAR, describe a complete set of confinement structures, systems, and components ITS in sufficient detail to permit evaluation of their effectiveness at the HI-STORE CIS Facility and, therefore, are acceptable.

9.3.2 Radionuclide Confinement Analysis

9.3.2.1 Identification of Release Events

The applicant summarized, in SAR section 4.3.1.4, “Confinement,” that the MPC provides confinement for all design-basis normal, off-normal, and postulated accident conditions, and the confinement criteria for the MPCs are incorporated by reference from Section 2.0.6, “Confinement,” of the HI-STORM UMAX FSAR. The applicant also stated, in SAR section 9.2.1, that all normal, off-normal, and accident conditions relevant to confinement integrity for which the canister is certified in the HI-STORM UMAX, Docket No. 72-1040, are equal to or less severe at the HI-STORE CIS Facility. Based on the staff’s review, this is consistent with the applicant’s statement in SAR section 4.7, “Summary of Design Criteria,” that the design criteria in SAR chapter 4, “Design Criteria for the HI-STORE CIS Systems, Structures and Components,” ensure that during normal, off-normal, and accident conditions, the confinement boundary is not breached. The staff notes the highest average monthly maximum temperature at the HI-STORE CIS Facility site exceeds the assumed 26.67 degrees Celsius (°C) (80 degrees Fahrenheit (°F)) ambient temperature boundary condition for the HI-STORM UMAX and the assumed 16.67°C (62°F) ambient temperature boundary condition that the applicant applied in the HI-STORE CIS Facility storage system thermal model. However, in Holtec Report No. HI-2177591, “Thermal Evaluations at the HI-STORM UMAX at HISTORE CIS Facility,” Revision 2, dated September 27, 2021, which the staff reviewed and referred to in chapter 6 of this SER, the applicant discussed the steady-state thermal analysis at an ambient temperature of 94°F (34°C) that bounds the highest average monthly maximum temperature at the HI-STORE CIS Facility site, which is 93.62°F (34.23°C). The applicant also concluded in SAR section 9.2.1 that, as a result of the design-basis off-normal and accident conditions that could challenge the integrity of the confinement system, there is no effect on the confinement function of the MPC, and all pressure boundary stresses remained within allowable ASME Code values. The staff reviewed the off-normal and accident conditions in SAR chapter 15, “Accident Analysis,” and finds that the analysis of off-normal and accident conditions consistently indicates that there is no effect on the confinement function of the MPC, and as appropriate for an off-normal pressure condition or site-specific earthquake accident condition, all stresses remain within allowable ASME Code values or design criteria and therefore 10 CFR 72.122(b) is met.

In SAR table 15.0.1, “Material Incorporated by Reference in this chapter,” the applicant incorporated by reference the evaluations of the leakage of one MPC seal weld off-normal condition and the confinement boundary leakage accident condition from section 12.1.3, “Leakage of One MPC Seal Weld,” and section 12.2.7, “Confinement Boundary Leakage,” respectively, of the HI-STORM UMAX FSAR. Section 12.1.3 of the HI-STORM UMAX FSAR incorporates by reference section 12.1.3, “Leakage of One Seal,” of the HI-STORM FW FSAR. Section 12.1.3 of the HI-STORM FW FSAR concludes that the MPC design, welding, testing, and inspection that includes a redundant welded closure meets the requirement of 10 CFR 72.236(e) such that leakage from the confinement boundary is considered not credible, and the staff reviewed and found this acceptable in section 5.3 of the HI-STORM FW initial issuance SER. Section 12.2.7 of the HI-STORM UMAX FSAR concludes that the information contained in chapter 7 of the HI-STORM FW FSAR demonstrates that the MPC is designed, fabricated, tested, and inspected to meet the guidance of Interim Staff Guidance (ISG)-18, “The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation,” dated May 2, 2003, such that unacceptable leakage from the confinement boundary is not credible, which the staff reviewed and found acceptable in section 5.3 of the HI-STORM FW initial issuance SER.

In its August 16, 2021, response to RAI 5-9-S, the applicant concluded that the cask transfer building (CTB) collapse is not credible under design-basis normal and accident condition loads. The applicant’s response to RAI 6-26 also concluded that the CTB collapse is not credible and that the HI-STAR 190 transportation package, which contains either an MPC-37 or MPC-89 canister, remains leaktight while on site before and during receipt inspection, which the staff finds to be acceptable. Section 5.3.4.4.2, “CTB superstructure,” of this SER addresses the accident condition of a large tornado missile with a non-collapse criterion of the CTB. The applicant’s response to RAI 5-37, dated April 15, 2022 (proprietary), also concludes that the CTB doors were analyzed for tornado missiles and there are administrative controls to halt all operations if there is a tornado watch or warning and concluded that the CTB is qualified against natural phenomena. Section 5.3.4.4.2 of this SER also addresses how the HI-STAR 190 transportation package, the transfer cask, and the stack-up condition provide tornado missile protection while they are in the CTB.

9.3.2.2 Evaluation of Release Events

The applicant summarized, in SAR section 9.2.1, how the stresses the canister will experience at the HI-STORE CIS Facility are lower than those allowed for the canisters certified in the HI-STORM UMAX CoC No. 1040, Docket No. 72-1040, because the design-basis heat load and therefore internal gas temperature is lower at the HI-STORE CIS Facility. The applicant clarified in the response to RAI 6-2-S, dated August 16, 2021, that the HI-STORM UMAX FSAR refers to chapter 3, “Structural Evaluation,” of the HI-STORM FW FSAR for the stress qualification of the MPC, including the calculation of the pressure- and temperature-induced stresses in the MPC. The staff reviewed HI-STORE CIS Facility SAR table 4.1.1, “Maximum Decay Heat Load for MPC-37 (PWR Fuel Assembly),” and table 4.1.2, “Maximum Decay Heat Load MPC-89 (BWR Fuel Assembly),” and compared the decay heat values to those in HI-STORM UMAX FSAR table 2.1.8, “HI-STORM UMAX MPC-37 Permissible Heat Loads,” and table 2.1.9, “HI-STORM UMAX MPC-89 Permissible Heat Loads,” and HI-STORM FW FSAR table 1.2.3, “MPC-37 Heat Load Data (See Figure 1.2.1),” and table 1.2.4, “MPC-Heat Load Data (See Figure 1.2.2).” Based on its review, the staff finds that the decay heat is lower for the HI-STORE CIS Facility as compared to that described for the HI-STORM UMAX and HI-STORM FW.

The applicant also explained, in its August 16, 2021, response to RAI 6-2-S (proprietary) and in the associated Holtec Report No. HI-2177591, that it performed a thermal analysis with the HI-STORM UMAX using the highest average monthly maximum temperature at the HI-STORE CIS Facility site, 94°F (34.4°C), and adopted the resulting confinement boundary temperatures in the stress analysis in SAR chapter 5, “Installation and Structural Evaluation.” The staff confirmed that the internal temperatures and pressures for the HI-STORM FW in table 4.4.3, “Maximum Temperatures in HI-STORM FW Under Long-Term Normal Storage,” and table 4.4.5, “Summary of MPC Internal Pressures Under Long-Term Storage,” of the HI-STORM FW FSAR are greater than those for the HI-STORE CIS Facility given in table 6-11, “Results of HI-STORM UMAX at High Ambient Temperature,” of Holtec Report No. HI-2177591 for the confinement boundary MPC lid and shell. The staff reviewed the applicant-provided differential thermal expansion results in Holtec Report No. HI-2177591 and found the results showed no interference between components. The staff finds that the temperature-induced and pressure-induced stresses will be lower in the HI-STORE CIS Facility canisters than allowed for in the HI-STORM UMAX and, therefore, are acceptable.

The applicant also stated in SAR section 9.2.1 that all lifting and handling operations involving canisters at the HI-STORE CIS Facility are performed with single-failure-proof equipment and, therefore, no additional mechanical loading events would affect the canister confinement function. The single-failure-proof equipment includes the CTB crane, described in SAR section 4.5.2, “Cask Transfer Building (CTB) Crane,” and the vertical cask transporter (VCT) described in SAR section 4.5.3, “Vertical Cask Transporter.” The applicant described in SAR section 4.5.3 how the VCT has features that protect against uncontrolled lowering of the carried load, in addition to having certain parts of the VCT meet the guidance in NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980. The applicant also described in SAR section 4.5, “Lifting Equipment (CTB Crane & VCT), Special Lifting Devices and Miscellaneous Ancillaries,” portions of the VCT that are considered to be special lifting devices, along with any associated equipment. The applicant described in SAR section 4.5.1.3, “Single Failure Proof Criteria,” the criteria under which a lifting device, or special lifting device, is considered to be single-failure proof. Therefore, the staff concludes, based on its review of the information described in this paragraph, that the storage conditions evaluated for the HI-STORM UMAX are equal to or greater than the storage conditions at the HI-STORE CIS Facility.

The staff verified that section 10.1.4, “Leakage Testing,” of the HI-STORM UMAX FSAR, describes how the applicant completed leakage rate testing of the entire confinement boundary in accordance with the American National Standards Institute (ANSI) N14.5-1997, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment,” to the leaktight acceptance criterion, 1.0×10^{-7} reference cubic centimeter per second (ref-cm^3/s). Specifically, section 2.0.6 of the HI-STORM UMAX FSAR, which the applicant incorporated by reference as described in SAR section 4.3.1.4, states that the MPC design meets the guidance in ISG-18, so that leakage of radiological material from the confinement boundary is not credible. The staff verified that section 5.1, “Confinement System,” of the initial issuance of the HI-STORM UMAX SER, states, “All the confinement components (including the confinement welds and the base metals) of the HI-STORM UMAX are required for helium leak testing, except the lid-to-shell weld since the weld meets the criteria of ISG-18. The confinement boundary of the HI-STORM UMAX is helium leakage tested to be leak-tight (1.0×10^{-7} ref-cm^3/sec) in accordance with the leakage test methods and procedures of ANSI N14.5-1997.” The staff also verified that section 8.9, “Supplemental Information for the Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage,” of NUREG-1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” Revision 1, issued July 2010, summarizes that there is reasonable assurance that no credible leakage of radioactive material would occur through the structural lid-to-canister shell weld of an austenitic stainless steel canister, and that helium leakage testing of this specific weld is unnecessary, provided the weld is executed and examined in accordance with the methods described in section 8.9 of NUREG-1536. The staff notes that the description of the leak testing in the HI-STORM UMAX SAR for the MPC-37 and MPC-89 canisters is consistent with ANSI N14.5 and ISG-18 as determined in the initial issuance of the HI-STORM UMAX SER and therefore based on following this guidance can be used at the HI-STORE CIS Facility.

Section 2.0.6 of the HI-STORM UMAX FSAR states that no confinement dose analysis is performed. The staff verified that section 5.4.4, “Confinement Analyses,” of NUREG-1536 explains that, for casks that are demonstrated to be leaktight, the review procedures discussed in section 5.5.3, “Nuclides with Potential for Release,” and section 5.5.4 of NUREG-1536 are not applicable. The staff therefore finds that because the HI-STORE CIS Facility would only use the HI-STORM UMAX Canister Storage System with the MPC-37 and MPC-89 canisters, which meet the ANSI N14.5 leaktight acceptance criterion, the applicant need not perform the confinement dose analysis and demonstration of leakage rate testing described above in this SER.

The applicant described, in section 2.0.6 of the HI-STORM UMAX FSAR, how the confinement function of the MPC-37 and MPC-89 canisters is verified through pressure testing, helium leak testing, and weld examination, in accordance with chapter 10, “Acceptance Criteria and Maintenance Program,” of the HI-STORM FW FSAR. The staff verified that pressure testing, helium leak testing, and weld examination are described in section 10.1.2.2.2, “MPC Confinement Boundary”; section 10.1.4, “Leakage Testing”; and section 10.1.1, “Fabrication and Nondestructive Examination (NDE),” respectively, of the HI-STORM FW FSAR. In SAR table 17.0.2, “Material Incorporated by Reference in this chapter,” the applicant incorporated by reference section 8.13, “Examination and Testing,” of the HI-STORM UMAX FSAR, which specifies that examination and testing requirements for the MPC and HI-TRAC are described in section 8.14, “Examination and Testing,” of the HI-STORM FW FSAR. Section 8.14 of the HI-STORM FW FSAR further describes helium leak testing of the canister and welds and also specifies that a comprehensive discussion on the examinations and testing that are conducted during the manufacturing process is provided in section 10.1 of the HI-STORM FW FSAR, which includes the aforementioned sections of this paragraph in chapter 10 of the HI-STORM FW FSAR. The staff finds that the NRC had previously approved the pressure testing, helium leak testing, and weld examination described in the HI-STORM FW FSAR in the initial issuance of the HI-STORM FW SER, and the staff determined that the referenced information in section 8.1.3 of the HI-STORM UMAX FSAR, Revision 3, is applicable to the HI-STORE CIS Facility confinement evaluation and that this incorporation by reference is, therefore, acceptable.

The applicant explained in SAR section 9.2.1 that to demonstrate the continued condition of no credible leakage at the HI-STORE CIS Facility, its personnel must review the 10 CFR 72.48, “Changes, tests, and experiments,” screenings or evaluations for the canister original licensing basis and evaluate them against the HI-STORE CIS Facility site-specific license to determine whether a change requires NRC approval. The staff verified that section 5.5.5.c of the TS describes the review of the 10 CFR 72.48 screenings or evaluations. In SAR section 9.2.1, the applicant also stated that the canister must not be subject to any incident beyond normal conditions of transport by which the package has been qualified in accordance with 10 CFR 71.71, “Normal conditions of transport.” The staff verified that section 5.5.5, “Canister Acceptance Program,” of the TS also states that “Canisters that have undergone an accident while in Part 71 transportation shall not be accepted at the HI-STORE facility," and therefore is acceptable.

The applicant further states (also in SAR section 9.2.1) that, during transportation to the HI-STORE CIS Facility, canister transportation operations are bounded by the HI-STAR 190 transportation package SAR, Revision 3, dated November 2, 2018, chapter 4, sections 4.5 through 4.7. Section 4.5, "Description of the Inner Containment System," describes the HI-STAR 190 transportation package's inner containment system, which includes the MPC enclosure vessel consisting of the MPC base plate, canister shell, lid, port covers, and closure ring. Section 4.6, "Inner Containment System Under Normal Conditions of Transport," of the HI-STAR 190 transportation package SAR describes how section 2.6, "Normal Conditions of Transport," of that same SAR demonstrates that the inner containment system components are maintained within their code-allowable stress limits during normal conditions of transport. Section 3.1, "Description of Thermal Design," of the HI-STAR 190 transportation package SAR shows that all containment system components are maintained within their peak temperature and pressure limits for normal conditions of transport. Section 4.7, "Inner Containment Integrity Under Hypothetical Accident Conditions of Transport," of the HI-STAR 190 transportation package SAR states that section 2.7, "Hypothetical Accident Conditions," of that same SAR shows that the inner containment system components are maintained within their code-allowable stress limits during all hypothetical accident conditions of transport. Chapter 3, "Thermal Evaluation," of the HI-STAR 190 transportation package SAR shows that all containment system components are maintained within their peak temperature and pressure limits for hypothetical accident conditions. Sections 4.6.2 and 4.7.2 of the HI-STAR 190 transportation package SAR, both entitled, "Containment Criteria," also describe how the ANSI N14.5-1997 leaktight criterion is applicable to the inner containment system leakage tests. Based on its review of the descriptions in sections 4.5 through 4.7 of the HI-STAR 190 transportation package SAR, because the MPC-37 and MPC-89 canisters for the HI-STORM UMAX Canister Storage System are limited to only being received using the HI-STAR 190 transportation package, the staff's SER for the HI-STAR 190 transportation package, dated August 8, 2018, and the leakage rate testing upon arrival at the HI-STORE CIS Facility, which is described in SAR section 9.2.1 and SAR chapter 10, "Conduct of Operations Evaluation," the staff finds that there is reasonable assurance that the MPC's confinement boundary integrity is maintained during transport to the HI-STORE CIS Facility.

In addition, the applicant described in SAR section 9.2.1 how the canister must pass the leak test and other receipt inspections described in SAR chapter 10 while in the HI-STORE CIS Facility receiving area. The staff verified that section 5.5.5.b.2 of the TS describes the receipt inspection leakage rate testing of each canister to be stored at the HI-STORE CIS Facility, in accordance with ANSI N14.5-2014, and to the leak testing methods and acceptance criteria in TS table 5-1, "Canister Leakage Test Performance Specifications."

The confinement boundary includes the lid-to-shell weld that is covered with a closure ring. In SAR section 9.2.1, the applicant stated that leakage rate testing is performed on the redundant closure ring and confinement boundary lid-to-shell weld together. The staff finds that this is consistent with the leakage rate testing described in SAR section 10.3.3.1, "Receipt and Inspection of Transportation Cask and Canister," and table 10.3.2, "Canister Leakage Test Performance Specifications," and the staff finds it acceptable, given that the canister is loaded before arrival at the HI-STORE CIS Facility and that the redundant closure ring and confinement boundary lid-to-shell weld are leak tested together.

The applicant stated in SAR section 9.2.2, “Operational Activities,” that when the MPC is on site at the HI-STORE CIS Facility and inside the HI-STAR 190 transportation package intact containment boundary for transportation, the MPC is the confinement boundary for the material. The applicant also stated in SAR section 9.2.2 that receipt inspections need to be performed and passed on each MPC to demonstrate the confinement boundary has not degraded during transport. The applicant stated in SAR section 9.2.2 that specific receipt inspections include gas sampling for fission products, explicitly Krypton-85, taken from the HI-STAR 190 transportation package’s closure lid access port, which allows access to the volume between the MPC and the interior of the HI-STAR 190 transportation package. In SAR section 10.2.2.3, “Other Testing,” the applicant described the use of NRC Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Revision 2, issued June 2009, for gas sampling equipment validation. The staff finds Regulatory Guide 1.21 to be acceptable because it was primarily developed based on 10 CFR Part 20, “Standards for Protection against Radiation,” and also cites 10 CFR 72.44(d), which establishes environmental monitoring requirements for each facility holding a specific license under 10 CFR Part 72 authorizing receipt, handling, and storage of spent fuel, high-level radioactive waste, and reactor-related Greater than Class C waste, and because Regulatory Guide 1.21 describes a method for reporting these results.

In SAR section 9.2.2, the applicant stated that, if no fission products are detected, the volume between the canister and the transportation package should be evacuated, flushed with nitrogen, and helium leak tested. The applicant described this sequence, for the purpose of removing residual helium, in SAR section 10.3.3.1, with the nitrogen backfill pressure limit specified in table 10.3.4, “Transport Cask Flushing/Backfill Requirements.” The applicant further indicated in SAR section 10.3.3.1 that the evacuation is followed by a leak test of the MPC. The applicant specified, in SAR table 10.3.2, that an evacuated envelope gas detector leakage rate test A.5.4 in ANSI N14.5-2014 is to be performed using a helium mass spectrometer to the ANSI N14.5-2014 leaktight acceptance criterion of 1.85×10^{-7} ref-cm^3/s of helium using helium as the tracer gas, and at a sensitivity of 9.2×10^{-8} ref-cm^3/s of helium. The staff finds the leakage rate acceptance criterion is consistent with the definition of leaktight in ANSI N14.5-2014 and the sensitivity is less than half of the acceptance criterion, which is also consistent with ANSI N14.5-2014; therefore, the leakage rate acceptance criterion and sensitivity are acceptable.

The applicant further explained, in SAR section 10.2.2.3, that the leak testing equipment will be calibrated in accordance with ANSI N14.5 both before and after leak testing; therefore, the staff finds this to be consistent with the calibration standard described in ANSI N14.5. The applicant indicated the following related to leakage rate testing in SAR section 10.3.3.1:

- Leakage rate testing procedures are approved by an American Society for Nondestructive Testing Level III examiner in leak testing, in accordance with ANSI N14.5-2014.

- Leakage rate testing procedures shall clearly define the test equipment arrangement.

- Leakage rate testing is performed by qualified personnel in accordance with the Holtec quality assurance program.

The staff finds that leakage rate testing procedures that are approved by an American Society for Nondestructive Testing Level III examiner in leak testing in accordance with ANSI N14.5-2014 and performed by qualified personnel in accordance with Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification of Nondestructive Testing,” issued 2006, as specified in SAR section 10.3.3.1, are both important aspects to provide reasonable assurance of confinement integrity. The staff further verified that section 5.5.5.b.2 of the TS describes leakage rate testing of each canister to the leaktight acceptance criteria in accordance with ANSI N14.5-2014 that is referenced in table 5-1 of the TS. Therefore, the staff finds that the HI-STORM UMAX Canister Storage System with MPC-37 and MPC-89 canisters is leaktight at the HI-STORE CIS Facility and, therefore, that the quantity of radioactive nuclides released to the environment from this storage system satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

In SAR section 9.2.2, the applicant stated that, if the leaktight criteria cannot be determined, or if fission products are detected, the port will be resealed and the canister will not be accepted at the HI-STORE CIS Facility. The staff verified that sections 5.5.5.b.1 and 5.5.5.b.2 of the TS describe the acceptance testing of loaded transport casks and canisters received at the HI-STORE CIS Facility and, for canisters that do not meet acceptance criteria, shipment to the nuclear plant of origin or other facility licensed to perform fuel loading procedures. The staff finds this to be consistent with the SAR description and therefore acceptable.

9.3.2.3 Radionuclide Confinement Analysis Conclusion

The staff concludes that the confinement boundary integrity for the HI-STORM UMAX Canister Storage System to be placed at the proposed HI-STORE CIS Facility will be maintained under normal, off-normal, and accident conditions, and that the regulatory requirements on 10 CFR 72.104(a) and 72.106(b) are met based on the following:

- the incorporation by reference, in SAR table 9.0.1, of chapter 7 from both the HI-STORM UMAX FSAR, Docket No. 72-1040, and the HI-STORM FW FSAR, Docket No. 72-1032
- the applicant’s identification of normal, off-normal, and accident release events
- the not credible CTB collapse
- the structural and thermal analysis results summarized in SAR chapter 9
- the use of single-failure-proof lifting equipment
- the pressure testing, helium leak testing to the leaktight acceptance criteria in accordance with ANSI N14.5, and weld examination performed on the MPC-37 and MPC-89 canisters while stored in the HI-STORM UMAX or FW before arrival at the HI-STORE CIS Facility

- the review of 10 CFR 72.48, “Changes, tests, and experiments,” screenings or evaluation for the MPC original licensing basis that are evaluated against the HI-STORE CIS Facility site-specific license that is described in the TS

- the MPC not being subject to any incident beyond normal conditions of transport that is described in the TS

- the use of the HI-STAR 190 transportation package

- the proposed receipt inspection testing, including sampling for krypton-85, and helium leakage rate testing of the MPC that is described in the TS

- compliance with the aging management program (AMP) described in SAR chapter 18, “Aging Management Program,” and further in SER section 9.3.4, “Protection of Stored Material from Degradation,” with the staff’s evaluation in SER section 17.3.16, “Maintenance and Aging Management.”

9.3.3 Confinement Monitoring

9.3.3.1 Confinement Casks or Systems

In SAR section 9.4.1.1, “Closure Seal Monitoring System,” the applicant stated that the canisters are seal welded and, consistent with the approved initial certificate of the HI-STORM UMAX and FW storage systems, monitoring of the closure is not required. The staff finds that this is consistent with section 9.5.3, “Confinement Monitoring,” of NUREG-1567 that states that casks closed entirely by welding do not require seal monitoring.

Further, in SAR section 9.4.1.2, “Continuous Monitoring System,” the applicant indicated that no monitoring of airborne radiation is needed in and around the storage area. In SAR section 9.4.2, “Effluents,” the applicant maintained that there are no effluents generated at the HI-STORE CIS Facility and that there is no potential for transport of radioactive materials to the environment through any aquifer; therefore, effluent monitoring is not necessary. The applicant further stated in SAR section 16.6, “Regulatory Compliance,” that, since the MPC meets the ANSI N14.5 leaktight criteria, which are described in SAR section 10.3.3, “Conduct of Operations,” release of effluents from MPCs are of the order of magnitude to be considered negligible and with no impact on public health and safety. The staff verified that section 5.5.1, “Radioactive Effluent Control Program,” of the TS states the following:

> Canisters containing radioactive materials are designed to meet the leaktight criterion per ANSI N14.5 under normal, off-normal and hypothetical accident conditions, and are not opened or breached during operations and storage at the HI-STORE CIS Facility. Upon arrival of canisters in transport casks, acceptance tests are performed on the loaded transport casks in accordance with Section 5.5.5.b to verify the confinement of the canisters post-transportation. Therefore, specific operating procedures for the control of radioactive effluents and monitoring program for effluents at the HI-STORE CIS Facility are not required.

Therefore, the staff finds, based on the review of the information described above, that the design and proposed operations of the HI-STORE CIS Facility include acceptable measures to minimize the potential for transport of radioactive materials to the environment through the aquifer and, therefore, will meet the requirements in 10 CFR 72.44(d)(2) and 72.122(b).

9.3.3.2 Pool and Waste Management Facilities

In SAR section 9.3.1, “Pool Facilities,” the applicant stated that the HI-STORE CIS Facility has no pool facilities or any other water-based storage or handling facility. In SAR section 9.3.2, “Waste Management Facilities,” the applicant further stated that the HI-STORE CIS Facility has no facilities for management of radioactive waste. The applicant justified this facility provision by stating that the facility actually generates only insignificant amounts of radioactive waste. Specifically, the applicant explained that the canisters received at the HI-STORE CIS Facility are seal welded with no credible leakage of radioactive contents, the transportation packages that contain the canisters are checked for contamination upon receipt and during processing and extraction of the canisters, and small gas samples are taken during receipt inspection of the canisters. In addition, the applicant explained that any waste collected during receipt inspections or operations would be transported off site. Therefore, based on its review of the information described in SER section 9.3.3.2, the staff concludes that the small amount of waste generated at the HI-STORE CIS Facility will be managed effectively and meets the regulatory requirements in 10 CFR 72.24(l)(2).

9.3.4 Protection of Stored Materials from Degradation

In SAR section 9.2.1, “Storage Systems,” the applicant discussed the need to comply with the aging management requirements outlined in SAR chapter 18, if canisters are stored beyond the initial license period of 20 years. This provides reasonable assurance that conditions that could be detrimental to the confinement function of the canister are identified and, if necessary, mitigated. SER section 17.3.16 includes the staff’s evaluation of the HI-STORE CIS Facility’s AMP.

In SAR section 9.5.1, “Confinement Casks or Systems,” the applicant stated that any potential degradation beyond the previously approved canister license is addressed in the AMP described in SAR section 18.5, “Canister Aging Management Program”; section 18.11, “Tilt Frame Aging Management Program”; section 18.13, “Learning Based AMP”; and section 18.15, “Ameliorating the Risk of Canister Degradation Over a Long-term Storage Duration.” The staff’s evaluation of this information appears in SER section 17.3.16. The staff finds that the design and proposed operations of the HI-STORE CIS Facility, which includes the receipt inspection helium leak testing and the AMP, provide adequate measures for protecting the spent fuel cladding against degradation that might otherwise lead to gross ruptures of the material to be stored, and, therefore, meet the requirements of 10 CFR 72.122(h)(1).

9.4 Evaluation Findings

Based on its review of the SAR, the staff concludes that the relevant requirements described in section 9.2, “Regulatory Requirements,” of this SER have been met, and the staff determined the following:

- Chapter 9 of the HI-STORE CIS Facility SAR, and chapter 7 of the HI-STORM UMAX Canister Storage System FSAR and chapter 7 of the HI-STORM FW FSAR, which are both incorporated by reference into the HI-STORE CIS Facility SAR, describe confinement structures, systems, and components that are ITS in sufficient detail so as to permit the NRC staff's evaluation of their effectiveness at the HI-STORE CIS Facility.

- The HI-STORM UMAX Canister Storage System, with the MPC-37 and MPC-89 canisters, at the HI-STORE CIS Facility provides redundant sealing of the confinement system. There are no bolted closures or mechanical seals in the MPC confinement boundary.

- The design and proposed operations of the HI-STORE CIS Facility include acceptable measures that minimize the potential for transport of radioactive materials to the environment through the aquifer, in compliance with 10 CFR 72.122(b).

- The HI-STORM UMAX Canister Storage System at the HI-STORE CIS Facility, with the MPC-37 and MPC-89 canisters, is leaktight and its quantity of radionuclides released to the environment satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

- The design of the HI-STORE CIS Facility and proposed operations, as described in the SAR, which include receipt inspection helium leak testing and an AMP, provide adequate measures for protecting the spent fuel cladding against degradation that might otherwise lead to gross ruptures of the material to be stored, in compliance with 10 CFR 72.122(h)(1).

- The HI-STORM UMAX Canister Storage System at the HI-STORE CIS Facility, with the MPC-37 and MPC-89 canisters that are the specified confinement system for PWR and BWR fuel, respectively, have been evaluated by appropriate tests or by other means acceptable to the Commission. This includes the HI-STORM UMAX Canister Storage System at the HI-STORE CIS Facility with MPC-37 and MPC-89 canisters that are helium leak tested to leaktight, as incorporated by reference as described in F9.1 and F9.4; receipt inspection helium leak testing; and the HI-STAR 190 transportation package analysis to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

- The staff concludes that the design of the confinement system of the HI-STORM UMAX Canister Storage System at the HI-STORE CIS Facility with MPC-37 and MPC-89 canisters is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the HI-STORM UMAX Canister Storage System at the HI-STORE CIS Facility with MPC-37 and MPC-89 canisters will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis, and accepted engineering practices.

9.5 References

American National Standards Institute (ANSI), "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-1997, February 1998.

ANSI, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," ANSI N14.5-2014, June 2014.

American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, 2006.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 2007.

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection against Radiation."

Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Material."

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

Holtec International, "Final Safety Analysis Report on the HI-STORM FW MPC Storage System," Holtec Report No. HI-2114830, Docket No. 72-1032, Revision 4, June 24, 2015. Agencywide Documents Access and Management System Accession No. ML15177A338.

Holtec International, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Holtec Report No. HI-2115090, Docket No. 72-1040, Revision 3, June 29, 2016. ML16193A339.

Holtec International, "Safety Analysis Report on the HI-STAR 190 Package," Holtec Report No. HI-2146214, Docket No. 71-9373, Revision 3, November 2, 2018. ML18306A911.

Holtec International, "HI-STORE Consolidated Interim Storage Facility, Requests for Additional Information Round 2, Attachment 2 Holtec Letter 5025068," August 16, 2021. ML21228A204 (proprietary version ML21228A203).

Holtec International, "Thermal Evaluations of the HI-STORM UMAX at HISTORE CIS Facility," Holtec Report No. HI-2177591, Revision 2, NRC Docket 72-1051, September 27, 2021. ML21331A009.

Holtec International, "Attachment 2 to Holtec Letter 5025071, Responses to HI-STORE CISF Round 3 RAIs," April 15, 2022. ML22108A117.

Holtec International, "Attachment 2 - Proposed HI-STORE License/Technical Specifications," November 23, 2022. ML22331A005.

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Holtec Report No. HI-2167374, Revision 0T, Docket No. 72-1051; transmittal dated January 20, 2023. ML23025A112.

U.S. Nuclear Regulatory Commission (NRC), NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” July 1980. ML070250180.

NRC, NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” March 2000. ML003686776.

NRC, “The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation,” Spent Fuel Project Office Interim Staff Guidance-18, May 2, 2003. ML031250620.

NRC, Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Revision 2, June 2009. ML091170109.

NRC, NUREG-1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” Revision 1, July 2010. ML101040620.

NRC, “Safety Evaluation Report, Docket No. 72-1032, Holtec International HI-STORM Flood/Wind System, Certificate of Compliance No. 1032,” July 14, 2011. ML111950325.

NRC, “Safety Evaluation Report, Docket No. 72-1032, HI-STORM FW MPC Storage System, Holtec International, Inc., Certificate of Compliance No. 1032, Amendment No. 1,” December 17, 2014. ML14351A475.

NRC, “Safety Evaluation Report, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040,” April 2, 2015. ML15093A510.

NRC, Certificate of Compliance for Spent Fuel Storage Casks No. 1040, Amendment No. 0, Docket No. 72-1040, issued April 6, 2015, for the Holtec International HI-STORM UMAX Cask Storage System. ML15093A509. Package ML15093A498.

NRC, Certificate of Compliance for Spent Fuel Storage Casks No. 1040, Amendment No. 1, Docket No. 72-1040, issued September 8, 2015, for the Holtec International HI-STORM UMAX Cask Storage System. ML15252A418. Package ML15252A426.

NRC, “Safety Evaluation Report, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Amendment No. 1, Holtec International, Inc., Certificate of Compliance No. 1040,” September 8, 2015. ML15252A423.

NRC, “Safety Evaluation Report, Docket No. 72-1032, HI-STORM Flood/Wind Multipurpose Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1032, Amendment No. 2,” October 25, 2016. ML16280A032.

NRC, “Safety Evaluation Report, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040, Amendment No. 2,” January 6, 2017. ML16341B129.

NRC, Certificate of Compliance for Spent Fuel Storage Casks No. 1040, Amendment No. 2, Docket No. 72-1040, issued January 6, 2017, for the Holtec International HI-STORM UMAX Cask Storage System. ML16341B080. Package ML16341B061.

NRC, “Safety Evaluation Report, Docket No. 71-9373, Model No. HI-STAR 190 Package, Certificate of Compliance No. 9373, Revision No. 0,” August 8, 2018. ML17222A083.

10 CONDUCT OF OPERATIONS EVALUATION

In chapter 10, "Conduct of Operations," of its safety analysis report (SAR), Revision 0T, dated January 20, 2023, Holtec International (the applicant) described the organization for the design, fabrication, construction, testing, operation, modification, and decommissioning of the proposed HI-STORE Consolidated Interim Storage (CIS) Facility. The applicant included descriptions of the organizational structure, personnel responsibilities and qualifications, and interface with contractors and other outside organizations, as well as a quality assurance plan and an emergency response plan. Separate from its application, the applicant provided a physical security plan (PSP, nonpublic), a safeguards contingency plan (SCP, nonpublic), and a security training and qualification plan (TQP, nonpublic). The applicant stated that the radiation safety plan described in SAR chapter 11, "Radiation Protection," will be implemented at the facility to comply with Subpart H, "Physical Protection," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-related Greater than Class C Waste."

10.1 Scope of Review

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the applicant's information related to the conduct of operations for the HI-STORE CIS Facility by reviewing the documents cited in or attached to the SAR, the applicant's responses to the staff's requests for supplemental and additional information, and other relevant literature. The staff reviewed this information to determine whether the applicant had described an appropriate infrastructure to manage, test, and operate the facility, including provisions for effective training, emergency planning, and physical security programs.

10.2 Regulatory Requirements

The NRC requirements relevant to evaluation of the conduct of operations for the proposed HI-STORE CIS Facility appear in the following NRC regulations:

- 10 CFR 72.24, "Contents of application: Technical information"
- 10 CFR 72.28, "Contents of application: Applicant's technical qualifications"
- 10 CFR 72.40, "Issuance of license"
- 10 CFR 72.180, "Physical protection plan"
- 10 CFR 72.182, "Design for physical protection"
- 10 CFR 72.184, "Safeguards contingency plan"
- 10 CFR 72.190, "Operator requirements"
- 10 CFR 72.192, "Operator training and certification program"
- 10 CFR 72.194, "Physical requirements"
- 10 CFR 73.51, "Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste"
- 10 CFR 73.71, "Reporting of safeguards events"

Additional requirements related to security appear in "Additional Security Measures for Access Authorization and Fingerprinting at Independent Spent Fuel Storage Installations," dated February 4, 2016, and "Additional Security Measures for the Physical Protection of Dry Independent Spent Fuel Storage Installations," dated September 28, 2007 (nonpublic).

10.3 Staff Review and Evaluation

Unless otherwise stated, the staff reviewed and evaluated the applicant's description of the conduct of operations for the proposed site discussed in chapter 10 of SAR, Revision 0T; documents cited in or attached to the SAR; the applicant's responses to the staff's requests for additional information; and the security-related plans submitted separately from the application.

The staff reviewed and evaluated the application's discussion of the proposed organizational structure; preoperational testing and startup operations; normal operations; personnel selection, training, and certification; emergency planning; PSP; security TQP; and SCP.

10.3.1 Organizational Structure

In SAR section 10.1, "Organizational Structure," the applicant stated that the organization responsible for storage of spent fuel at the HI-STORE CIS Facility is described in this section. The organization must be documented and updated, as appropriate, in organization charts. In addition, lines of authority, responsibility, and communication shall be defined and documented in organizational charts, functional descriptions of departments, and job descriptions for key personnel positions.

10.3.1.1 Corporate and Onsite Organization

The staff reviewed the applicant's corporate and onsite organizations detailed in SAR section 10.1.1, "Corporate and On-Site Organization," and the proposed requirements in Technical Specification 5.2, "Onsite and Offsite Organizations," of Appendix A, "Proposed Technical Specifications." The applicant identified the Holtec Corporate Executive as the executive responsible for the HI-STORE CIS Facility, with overall responsibility for safe operation of the site. The HI-STORE CIS Facility Site Manager reports to the Corporate Executive and is responsible for safe operation of the site; maintaining personnel trained and qualified in accordance with the "Holtec International & Eddy Lea Energy Alliance Underground Consolidated Interim Storage Facility - Training and Qualification Program," Revision 3, Holtec Report No. HI-2177562, dated October 9, 2020 (proprietary; hereafter referred to as the training and qualification program); day-to-day implementation of the Holtec International Quality Assurance Program, latest approved revision on docket 71-0784, (proprietary); and operation of all HI-STORE CIS Facility structures, systems, and components (SSCs) that are important to safety. The Site Manager is the highest level of management on site at the HI-STORE CIS Facility. The applicant illustrated the organizational structure in SAR figures 10.4.1, "Holtec Corporate Organization" and 10.4.2, "HI-STORE Site Organization." The staff reviewed Holtec International Nuclear Quality Assurance Manual, Revision 15 (proprietary), dated June 10, 2022, as the latest approved revision of the Holtec Quality Assurance Program.

In Technical Specification (TS) 5.1, "Responsibility," of Appendix A, "Proposed Technical Specifications," the applicant also committed to establishing a normal order of succession and delegation of authority in which the Site Manager will designate, in writing, personnel who are qualified to act in his or her absence.

In SAR section 10.1.2, “Support Staff (ISFSI Specialists),” the applicant described support staff who can be available to provide support and expertise to the Site Manager, and come from the corporate staff, onsite staff, or contract personnel:

- Quality assurance personnel are responsible for implementing the requirements of the Holtec Quality Assurance Program, including the maintenance of appropriate records. Quality assurance will ensure that the appropriate steps are included in site procedures for operation and maintenance to ensure that all activities are performed in accordance with the HI-STORE CIS Facility license.

- The site nuclear compliance engineer is responsible for oversight of facility modifications. Engineering support staff, either on or off site, will support the site nuclear engineer.

- The Radiation Protection Manager is responsible for radiation safety at the HI-STORE CIS Facility, the planning and direction of the facility radiation protection and as low as is reasonably achievable (ALARA) programs and procedures, and the operation of the health physics laboratory.

- Operating personnel are responsible for the receipt, inspection, and transfer of canisters arriving on site in accordance with site procedures.

- Maintenance personnel are responsible for mechanical, electrical, and instrument maintenance for buildings, fencing, mechanical equipment, and all other site equipment. These personnel also provide operations coverage for those periods when loaded canisters are handled, and routine site maintenance and surveillance when canisters are not being handled. They are responsible for ensuring that appropriate records are maintained in accordance with SAR section 10.3.2, “Records,” and the site license requirements.

- Security personnel are responsible for maintaining the security of special nuclear materials that are within the physical confines of the HI-STORE CIS Facility, including providing initial responses to security intrusions as described in the site security plan.

- Site administrative personnel are responsible for administrative functions, including the maintenance of records in accordance with section 10.3.2 of the SAR and the site license requirements. These personnel are also responsible for ensuring appropriate hiring standards are followed in the selection of staff members.

In SAR table 10.1.1, “Staffing Qualifications and Operating Organization,” the applicant described the HI-STORE CIS Facility independent safety reviewers (ISRs) as individuals who do not have direct involvement in the performance of the activities under review but who may be from the same functionally cognizant organization as the individuals performing the original work. The ISR must have 5 years of professional-level experience and either a bachelor’s degree in engineering or the physical sciences or the equivalent in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) -3.1-1981. The Holtec Corporate Executive must designate the qualified ISRs in writing.

The NRC staff noted that the organizational descriptions include managerial and administrative controls, such as site procedure and records controls. The descriptions also include a reference to implementation of the Holtec Quality Assurance Program that includes controls for procedures and records; surveillance, testing, and inspection; training and certification of personnel; and audit controls.

Based on its review described above, the staff finds the applicant's description of its corporate and onsite organizations adequate because it covers the onsite personnel functions, responsibilities, and authorities related to each aspect of the installation, including interfaces with outside organizations and contractors, as applicable. Therefore, the description meets the requirements of 10 CFR 72.24, 10 CFR 72.28, and 10 CFR 72.40.

10.3.2 Preoperational Testing and Startup Operations

The staff reviewed the information in SAR section 10.2, "Pre-operational Testing and Startup Operations," to determine whether the applicant adequately described its plans for preoperational and startup testing and other tests and inspections of equipment and systems to be used for safe handling, transfer, and storage of spent nuclear fuel (SNF) arriving at the HI-STORE CIS Facility. The applicant stated that all testing and inspection is performed before the initial loading of canisters into the HI-STORE CIS Facility, with the exception that dose rate measurements of the HI-STORE vertical ventilated module (VVM) are made after the initial loading to verify that the storage system satisfies the design criteria described in the SAR. These results will be maintained as records for future retrieval. The staff's review considered how the information in the SAR addresses the requirements of 10 CFR 72.24(p), which requires a description of the program covering preoperational testing and initial operations.

10.3.2.1 Administrative Procedures for Conducting the Test Program

The staff reviewed the applicant's description of the administrative procedures for the conduct of testing, including the system used for controlling the development and execution of the test procedures. In SAR section 10.2.1, "Administrative Procedures for Conducting the Test Program," the applicant described how preoperational and startup test procedures will be developed and implemented to support the preoperational testing and startup programs that meet the requirements of the Holtec Quality Assurance Program. Test results will be documented, evaluated, and approved by the applicant and must be shown to be within the acceptance criteria specified in the test procedures. The applicant noted that those modifications necessary because of testing results will be indicated through a specific process documented by procedure and will also require an evaluation if it is necessary to retest after a modification has been implemented. Based on the staff's review of SAR section 10.2.1, the staff finds the administrative procedures for conducting the test program to be acceptable.

10.3.2.2 Preoperational Testing Plan

The staff reviewed the preoperational testing plan described in SAR section 10.2.2, "Pre-operational Testing Plan," and Technical Specification 5.5.3, "Pre-Operational Testing and Training Exercise of HI-STORE CIS Facility Systems and Equipment," which the applicant described as being divided into two parts: preoperational testing and startup testing. Also included is a description of other tests and inspections important to proper operation and integrity of the storage system and handling equipment. The objective of the preoperational testing program is to verify that the storage system components can operate safely and effectively upon startup of operation. The scope of these tests will include operational testing of equipment, functional testing, fabrication acceptance tests and inspections, and equipment validation and calibration. Table 10.2.1, “Pre-Operational, Startup, and Other Tests,” of the application summarizes all the preoperational, startup, and other tests.

The staff noted that, in addition to the tests and inspections described in the SAR, the applicant will inspect all safety-significant equipment before use to ensure fabrication was performed in accordance with the design drawings. Further, the applicant will test radiation shielding materials for shielding effectiveness and will review test reports to that verify required steel and concrete properties to meet structural and shielding requirements will be determined by testing during construction.

For the HI-STORE VVM storage system, the applicant described the need to only test the optional air outlet temperature monitoring system before initial loading of spent fuel in storage because the system uses passive cooling.

The applicant described how preoperational testing of equipment will be performed associated with the transfer of a full-size and -weight dummy multipurpose canister (MPC) from receipt to insertion in the VVM. During this transfer, the function of the evolution component will be evaluated and the adequacy of procedures, communication, and personnel safety verified. The staff noted that the operation includes, but is not limited to, receiving the HI-STAR 190 transport cask, removing the transport cask from the shipping railcar, integrity testing of the canister that includes leakage testing, measuring cavity gas krypton-85 levels, upending and transferring the transport cask to the canister transfer facility, removing the transport cask lid, installing rigging and lifting apparatus on the MPC, loading the dummy MPC into the HI-TRAC CS [concrete shielded] transfer cask, transferring the MPC into the VVM, and installing the VVM closure lid. Upon completion of the plan, the applicant will evaluate and document the effectiveness of procedures, actions, and equipment to improve operations for subsequent spent fuel shipments.

Load tests are also performed under the preoperational test plan. As part of these tests, the following components are evaluated to ensure fabrication acceptance criteria are met: the cask transfer building crane, vertical cask transporter lift brackets and structure, HI-STAR 190 transport cask lifting trunnions and associated lift yoke, tilt frame, transport cask horizontal lift beam, HI-TRAC CS transfer cask lifting trunnions, HI-TRAC CS lower shield gates, HI-TRAC CS lift links, HI-TRAC CS lift yoke, and MPC lift attachment and lifting device extension. The tests also include (1) functional testing of the HI-TRAC transfer cask, (2) validation of the HI-STAR 190 transport cask gas sampling equipment using a National Institute of Standards and Technology-traceable validation source in accordance with NRC Regulatory Guide (RG) 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Revision 2, issued June 2009, (3) calibration of leak test equipment in accordance with the requirements of ANSI N14.5-2014, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment,” before and after leak test measurements, and (4) testing of the resistance temperature detector monitoring system that includes acceptance testing of the system before preoperational tests and also before installation of an MPC into each VVM. Based on the staff’s review of SAR section 10.2.2, the staff finds the preoperational testing plan to be acceptable.

10.3.2.3 Startup Testing

The applicant stated that, upon startup, testing or measuring the external radiation dose rates for each VVM will be performed after the loading with SNF is completed. The dose rates will be confirmed to be less than the maximum expected dose rates from chapter 11 of the SAR, “Radiation Protection Evaluation.” These measurements will confirm that personnel exposure is bounded by the safety analysis in the SAR. The staff finds the testing plan to be acceptable. The startup test plan is also referenced in proposed license condition 17 and shall be submitted to the NRC at least 90 days prior to receipt and storage of spent fuel at the facility.

10.3.2.4 Conclusion

Based on its review, the staff finds the applicant’s description of its preoperational testing and initial operations acceptable because the preoperational testing plan and startup test plan, as described in SAR section 10.2, includes the necessary tests and provides for proper evaluation, approval, and use of the test results. Therefore, the description meets the requirements of 10 CFR 72.24(p).

10.3.3 Normal Operations

The staff reviewed the information in SAR section 10.3, “Normal Operation,” to determine whether the applicant adequately described the administrative controls, procedures, and recordkeeping in support of normal operations. In addition, the staff reviewed the application to confirm the operating, maintenance, testing, and surveillance functions important to safety are described in the SAR and technical specifications, as applicable. The review considered how the information in the SAR addresses the following regulatory requirements:

- In 10 CFR 72.24(h), the NRC requires a plan for the conduct of operations, including the planned managerial and administrative control systems.
- In 10 CFR 72.40(a)(13)(i), the NRC requires reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public.

10.3.3.1 Procedures

The staff reviewed the applicant’s description of the administrative controls for the use of procedures for performing quality activities at the HI-STORE CIS Facility. In SAR section 10.3.1, “Procedures,” and Technical Specification 5.4, “Procedures,” the applicant described the HI-STORE CIS Facility program for conducting normal facility operations important to safety using written procedures as defined in the Holtec Quality Assurance Program, including the use of procedures to implement the “HI-STORE CIS Facility Fire Protection Plan,” Revision 0 (proprietary), training and certification of personnel, and procedure change controls.

The staff also reviewed the applicant’s description of the conduct of operations at the HI-STORE CIS Facility that will be used to develop detailed operating procedures. In SAR section 10.3.3, “Conduct of Operations,” the applicant described the generic criteria and the detailed operational steps required from receipt of the transportation cask on site to installation of the spent fuel canister in the cavity enclosure container, including the removal of canisters from the cavity enclosure container. The generic criteria require that the procedures are in conformance with the HI-STORM UMAX final safety analysis report (FSAR), Revision 3, and the HI-STORM FW System FSAR, Revision 4, where applicable, as all SNF canisters being stored at the HI-STORE CIS Facility will be from those system designs. The detailed steps contain references to the acceptance criteria for completion of required tests, inspections, and surveillances, as well as operational limits. The staff verified that the general criteria and detailed operational steps could be reasonably performed, did not pose a conflict with the previously approved HI-STORM UMAX or HI-STORM FW operational requirements, and were consistent with the acceptance criteria and operational limits developed from the technical evaluations performed in this application.

SAR table 10.3.1 states that inspections of VVM plenums would occur annually or following a severe weather event that may introduce foreign materials, in order to visually verify inlet and outlet plenums are free of significant material and air passages are not degraded. Likewise, SAR section 10.3.4, “Maintenance Program for the HI-STORM UMAX VVM Systems,” notes that air vent screens and VVM plenum air passages undergo periodic surveillance to ensure there are no blockages. Air vent extensions can be temporarily removed from surrounding VVMs to keep them out of the way during HI-TRAC transportation, allowing the HI-TRAC to travel past the VVMs, and then reinstalled afterwards. SAR section 10.3.3.5 states that the openings for air vent extensions are covered by a low-profile temporary cover screen assembly that prevents debris from entering the vent opening without blocking air flow. Finally, SAR section 10.2.2 and table 10.2.1 state that the optional air temperature monitoring system is calibrated and tested before loading a canister into a VVM. The staff finds that these operations (i.e., keeping ventilated air passages free of blockage) support the ventilated passive cooling basis of the HI-STORE CIS UMAX thermal design described in the SAR chapter 6, “Thermal Evaluation.”

SAR sections 10.3.4 through 10.3.8 describe the maintenance activities for SSCs at the HI-STORE CIS Facility. The staff notes that maintenance is one part of the applicant’s program to manage potential degradation of SSCs throughout the term of the license. As described in SAR section 18.14, “Timing of Aging Management Implementation,” SSCs will be maintained by a combination of two site programs:

(1) Maintenance (documented in SAR chapter 10): Maintenance inspections and tests will be performed on all equipment constructed exclusively for the HI-STORE CIS Facility. These activities will begin at the start of operation and continue throughout the term of the license.

(2) Aging management (documented in SAR chapter 18, “Aging Management Program”): After any SSC reaches 20 years of service, including the time the MPCs have already been in storage at their original independent spent fuel storage installation (ISFSI) sites, additional aging management activities will be performed to address aging-related degradation issues unique to longer term service.

The staff documents its evaluation of the applicant’s activities to ensure that the condition of the HI-STORE CIS Facility SSCs will be adequately maintained in SER section 17.3.16.

Based on its review, the staff finds that the applicant adequately described its plan for the conduct of normal operations important to safety at the proposed site. The HI-STORE CIS Facility programs include detailed written and approved procedures for all important-to-safety site activities consistent with the certificates of compliance and license technical specifications requirements. Therefore, the description meets the requirements of 10 CFR 72.24(h) and 72.40(a)(13).

10.3.3.2 Records

The staff reviewed the applicant's description of the records that will be maintained and the administrative controls for recordkeeping that will be in place. In SAR section 10.3.2, the applicant described the administrative controls for maintaining records to ensure they are identifiable and retrievable. Storage will be in accordance with the requirements of the Holtec Quality Assurance Program.

The applicant also described the types of records that will be maintained under the Holtec Quality Assurance Program. These include, but are not limited to, all records relating to quality assurance, operating records that include maintenance and modifications, records of off-normal occurrences and events associated with radioactive releases, environmental surveys, personnel training and qualification records, material control and accounting records, and radiation survey and occupational dose monitoring on site.

Based on its review, the staff finds the applicant's description of the identified records for retention and the controls for those records to be acceptable because it includes all those records required by regulations and the records are controlled by Holtec's Quality Assurance Program. Therefore, the description meets the requirements of 10 CFR 72.24(h).

10.3.3.3 Normal Operations: Summary

Based on its review, the staff finds the applicant's description of its normal operations acceptable because the procedures and records, as described in SAR section 10.3, include the necessary procedures for important to safety operations and records as required by regulations. Therefore, the description meets the requirements of 10 CFR 72.24(h) and 10 CFR 72.40(a)(13).

10.3.4 Personnel Selection, Training, and Certification

10.3.4.1 Personnel Organization

The staff reviewed SAR section 10.4, "Personnel Selection, Training, and Certification," and Holtec Report No. HI-2177562, "Holtec International & Eddy Lea Energy Alliance Underground Consolidated Interim Storage Facility—Training and Qualification Program," Revision 3, dated October 9, 2020 (proprietary; hereafter referred to as the training and qualification program), referenced in the SAR, and Technical Specification 5.3, "HI-STORE CIS Facility Staff Qualification," of Appendix A, "Proposed Technical Specifications," to determine whether the applicant identified the organization responsible for personnel selection, training, and certification and described a program to ensure that personnel with responsibilities for functions important to safety will be appropriately qualified and trained. This program establishes the guidelines for training and certification and for developing and maintaining a training program using a systematic approach to training similar to the approach defined in 10 CFR 55.4, “Definitions,” to be consistent with industry standards. The review considered how the information in the SAR addresses the following regulatory requirements:

- In 10 CFR 72.24(h), 10 CFR 72.28(b), and 10 CFR 72.40(a)(9), the NRC requires a program for training of personnel under Subpart I, “Training and Certification of Personnel,” of 10 CFR Part 72.
- In 10 CFR 72.24(j) and 10 CFR 72.28(a), the NRC requires the technical qualifications of the applicant to engage in the proposed activities.
- In 10 CFR 72.28(d), the NRC requires a commitment by the applicant to have and maintain an adequate complement of trained and certified personnel before receipt of radioactive material.
- In 10 CFR 72.190, the NRC requires operation of equipment and controls identified as important to safety in the SAR and license to be limited to trained and certified personnel or to be under the direct visual supervision of an individual with training and certification in the operation.
- In 10 CFR 72.192, the NRC requires the applicant to establish a program for training, proficiency, testing, and certification of ISFSI personnel and to submit it to the NRC for approval with the license application.
- In 10 CFR 72.194, the NRC requires the physical condition and general health of personnel certified for the operation of equipment and controls 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.

The staff noted that the training and qualification program describes how the Training Department is responsible for management of the training program and identifies the key responsible personnel; for example, the Training Coordinator and Training Administrative Assistant.

10.3.4.2 Selection and Training of Operating Personnel

SAR section 10.4.2, “Selection and Training of Operating Personnel,” describes how all individuals with unescorted access to the site will receive training on radiation protection, security, emergency plan, quality assurance, fire protection, and industrial safety related to safe operation of the ISFSI. Annual refresher training will be completed by all who need to maintain unescorted access. Operation of equipment and controls identified as important to safety are limited to personnel who are trained and certified in accordance with the training and qualification program or who are under the direct visual supervision of a person who is trained and certified in accordance with the training and qualification program. The applicant also committed to having an adequate complement of trained and qualified personnel before the receipt of SNF or reactor-related Greater than Class C waste as appropriate for storage.

The training and qualification program includes basic radiation worker, emergency response, and position-specific training, with both classroom and on-the-job training. To demonstrate competency, trainees must achieve a grade of 80 percent or above on all examinations. Refresher training is required every 2 years for basic radiation workers, health physicists, surveillance technicians, rail yard operators, crane operators, Holtec equipment operators, and riggers. Emergency response team personnel require annual refresher training. The applicant committed to ensuring that the physical condition and general health of qualified personnel 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. The applicant also described how the minimum qualifications for operating personnel in SAR table 10.1.1, where they are referred to as “specialists,” shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-2014, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” for a comparable position. Qualifications include having a high school diploma or completion of the General Education Development (GED) test, in addition to the applicable HI-STORE CIS Facility training program. In addition, operating personnel shall have a minimum of 2 years of experience in a nuclear facility.

The applicant stated that records will be maintained on the status of trained personnel, new employee training, refresher training of current personnel, lesson plans, course attendance, written examinations and results, completed qualification cards, and offsite training documentation. Based on the staff’s review of SAR section 10.4.2, the staff finds the selection and training of operating personnel to be acceptable.

10.3.4.3 Selection and Training of Security Guards

In SAR section 10.4.3., “Selection and Training of Security Guards,” Holtec committed to providing security training and qualification according to Holtec Report No. HI-2177559, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Physical Security Plan,” dated March 2, 2020. The staff discusses the qualifications for employment in security and security force training in SER sections 10.3.6.8 and 10.3.6.9, respectively.

10.3.4.4 Selection and Training of Radiation Protection Technicians

In SAR section 10.4.4, “Selection and Training of Radiation Protection Technicians,” the applicant described how it will provide sufficient training and certification for radiation protection technicians in accordance with the HI-STORE CIS Facility radiation protection technician training program, which is based on the requirements of ANSI/ANS 3.1-2014. The main purpose of the program is to provide personnel with training in radiation protection, surveys, radiation work permits, calibration of radiation survey equipment, ALARA principles, and packaging of radioactive materials in order to adequately implement the radiation protection program procedures. Required training will also ensure familiarity with the proper response in the event of an emergency in accordance with the radiological emergency plan. The technicians will also receive training in quality assurance, fire protection, security, industrial and chemical safety, and responsibility for safe operation of the HI-STORE CIS Facility. Refresher training will occur annually. For operating personnel, the applicant committed to maintain records on the status of trained personnel, new employee training, and refresher training. Based on the staff’s review of

SAR section 10.4.4, the staff finds the selection and training of radiation protection technicians acceptable.

10.3.4.5 Summary: Personnel Selection, Training, and Certification

Based on its review, the staff finds that the applicant has identified an organization responsible for personnel selection, certification, and training and has developed a personnel training program that includes an acceptable description consistent with the requirements in 10 CFR 72.24, 10 CFR 72.28, and Subpart I of 10 CFR Part 72. In addition, the staff finds that the applicant has provided acceptable technical qualifications, including training and experience, for personnel who will be engaged in the proposed activities in compliance with 10 CFR 72.28(a). The staff shall verify implementation of the training program before initiation of operations with SNF or reactor-related Greater than Class C waste, including adequate completion of staff training and certification before receipt of the radioactive material to be stored.

10.3.5 Emergency Planning

This section describes the staff's review and evaluation of the applicant's emergency response plan (ERP), Holtec Report No. HI-2177535, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Emergency Response Plan," Revision 5, dated November 17, 2022, to determine whether the applicant has established adequate emergency management facilities and procedures to protect workers, the public, and the environment.

In 10 CFR 72.40(a)(11), the NRC requires that for the issuance of a license, the applicant's emergency plan must comply with the regulatory requirements for ISFSI emergency plans in 10 CFR 72.32, "Emergency plan."

The criteria for the NRC's acceptance review of the Holtec International ERP are outlined in section 10.5.5, "Emergency Planning," of NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," issued March 2000, as revised by Spent Fuel Storage and Transportation Interim Staff Guidance (SFSTISG) 16, "Emergency Planning," issued June 2000, and Revision 1 to RG 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," issued April 2011.

10.3.5.1 Facility Description

The staff reviewed section 1.0, "Introduction and Facility Description," of the Holtec ERP, which described the licensed activity, the facility and site, and the area near the site. The Holtec ERP also included the following figures:

- Figure 1.1A, "Map of New Mexico"
- Figure 1.1B, "Proposed Location of HI-STORE CIS Facility"
- Figure 1.1C, "CIS Facility Site Boundaries"
- Figure 1.1D, "Existing Local Facilities Relative to CIS Facility"
- Figure 1.1E, "CIS Facility Layout"

The staff reviewed the following information:

- a detailed description of the site location and layout
- a description of the major structures to be located at the site, including the cask storage area and important supporting structures
- locations of population concentrations
- locations of industrial facilities
- identification of routes for access of emergency equipment or for evacuation
- the types of terrain and the land use patterns around the site

Based on its review, the staff finds that the Holtec ERP provides a brief description of the facility and area near the site and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(1).

10.3.5.2 Types of Accidents

The staff reviewed the information in section 2.0, “Types of Accidents,” of the Holtec ERP, which described the types of accidents that could result in the release of radioactive material, physical locations where they could occur, and how they could occur. The types of accidents considered for the facility, which deploys the HI-STORM UMAX storage system, are identified in SAR section 15.3 and are based on the accidents previously evaluated and approved by the staff in sections 12.2 and 12.3 of the HI-STORM UMAX storage system FSAR, Revision 3, dated June 29, 2016, which the applicant incorporated by reference in SAR table 15.0.1, “Material Incorporated by Reference in this Chapter.” The staff reviewed the following accident scenarios of interest to the facility, including the onsite and offsite consequences of potential accidents:

- cask-handling accidents
- 100 percent fuel rod rupture
- confinement boundary leakage
- fire
- explosion
- blockage of vent holes
- natural events (tornados, lightning, earthquakes, flooding, etc.)

Based on its review, the staff finds that the Holtec ERP identifies each type of radioactive materials accident and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(2).

10.3.5.3 Classification of Accidents

The staff reviewed appendix C, “Facility Emergency Action Levels,” to the Holtec ERP, which provides an emergency classification system, based on emergency action levels, that classified the accidents identified by the applicant in chapter 15 of the SAR. The Holtec ERP includes an emergency classification system for classifying accidents at the Alert classification level consistent with the NRC guidance in RG 3.67. The ERP also contains emergency action levels that specifically characterize the occurrence of accidents that warrant the declaration of the Unusual Event and Alert classifications.

Based on its review, the staff finds that the applicant's ERP contains an emergency classification system consistent with the guidance in RG 3.67 and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(3).

10.3.5.4 Detection of Accidents

The staff reviewed section 2.2, "Detection of Accidents," of the Holtec ERP, which describes the emergency monitoring equipment available for personnel and area monitoring. The emergency monitoring equipment is dependent on personnel observation, fire detection systems, and radiation monitoring instrumentation.

Based on its review, the staff finds that the Holtec ERP adequately describes the means for detecting accident conditions and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(4).

10.3.5.5 Mitigation of Accident Consequences

The staff reviewed the following information in the Holtec ERP related to the mitigation of accident consequences:

- Section 5.3, "Mitigating Actions," describes the equipment, training, and methods to mitigate emergency events. This includes the fire protection system and equipment to mitigate fires. It also describes mitigation of spills, natural events, and injuries, including actions to be taken by trained site personnel to mitigate emergency events.

- Section 5.4, "Protective Actions," describes radiological protective actions to protect onsite personnel.

- Section 4.3.11, "Local Off-site Assistance," describes arrangements made for medical and hospital services, fire, and local law enforcement.

- Section 6.0, "Emergency Response Equipment and Facilities," describes response equipment, facilities, and communications equipment that will be available to support mitigation efforts. The types and locations of the equipment are also covered.

- Section 7.5, "Maintenance and Inventory of Emergency Equipment, Instrumentation and Supplies," describes provisions for inventory and test equipment.

Based on its review, the staff finds that the Holtec ERP adequately describes the means and equipment to mitigate the consequences of the accidents in terms of protection of workers, and the program to maintain equipment consistent with NRC guidance in RG 3.67. Therefore, it contains sufficient information to meet the requirements of 10 CFR 72.32(a)(5).

10.3.5.6 Assessment of Releases

The staff reviewed section 5.2, “Accident Assessment,” of the Holtec ERP and section 11.4, “Radiation Protection Program,” of the SAR, which describe, in general, radiological sampling and monitoring methods that will be used to assess the extent of radioactive releases. The applicant described the types and methods of onsite and offsite sampling and monitoring in section 11.4 of the SAR. The applicant stated that portable survey and personnel monitoring instrumentation, if deemed necessary during normal, off-normal, or accident conditions, will include, but not be limited to, the following:

- low-level contamination meters
- beta/gamma portable survey meters
- alarming beta/gamma personnel friskers
- portable air samplers

Based on the description in SAR section 11.4.2, area radiation monitors will be used in the canister transfer building, since the operations in this building (i.e., transport cask receipt, inspection, and canister transfer operations) pose the greatest risk to the operating staff for radiation exposure. These monitors have audible alarms to warn operating personnel of abnormal radiation levels. Area radiation monitors will not be used outside the canister transfer building, since these areas have relatively low area radiation levels and there are no operations in these areas that could result in rapid change in radiation level and pose a risk for overexposure of personnel.

As described in the Holtec ERP, the attendant Radiation Protection Technician would collect real-time data at or near the incident site and relay those data to the Radiation Safety Officer, the Emergency Coordinator, or both. In the Holtec ERP, the applicant designated the Radiation Protection Manager as the communicator of offsite dose projections or potential impact assessment information to State and local agencies, as appropriate. The Radiation Safety Officer will ensure offsite response organizations have or are provided (if required) adequate dosimetry and briefings.

Based on its review, the staff finds that the Holtec ERP briefly describes the methods and equipment that will be used to assess releases of radioactive material and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(6).

10.3.5.7 Responsibilities

The staff reviewed the responsibilities of facility personnel during normal operations and during emergency situations described in section 4.0, “Responsibilities,” of the Holtec ERP. As described in the Holtec ERP, in an emergency, the Site Emergency Director, or designee, is the individual responsible for managing the activities outlined under the Holtec ERP and for maintaining and updating the Holtec ERP as information and conditions change. The Site Emergency Director delegates these duties to the on-shift Emergency Coordinator until the Site Emergency Director arrives on site to assume command and control of the event. The Site Emergency Director can be reached via telephone to assist and advise the on-shift Emergency Coordinator of his or her recommendations. These duties include the following:

- decision to declare an Unusual Event
- decision to escalate to an Alert
- activation of the onsite emergency response organization
- prompt notification of offsite response authorities to inform them that an Unusual Event or Alert has been declared (normally within 15 minutes of declaring an emergency)
- notification to the NRC Operations Center immediately after notification of offsite authorities, and in any case within 1 hour of the declaration of an Unusual Event or Alert
- decision to initiate any onsite protective actions
- decision to request support from offsite organizations
- decision to terminate the emergency or enter recovery mode

In section 4.0 of the Holtec ERP, the applicant also summarized the responsibilities of the remaining onsite staff, including the Health and Safety Manager, Radiation Safety Officer, security supervision and officers, and managers, supervisors, and staff. The applicant identified other emergency response organization (ERO) members as follows:

- first responders
- operations manager
- logistics member/scribe
- radiation protection technicians
- maintenance personnel

The staff reviewed section 7.0, “Maintaining Emergency Preparedness Capability,” of the Holtec ERP, which describes the normal site organization and identifies the personnel responsible for maintaining and updating the Holtec ERP, implementing procedures, and related records.

Based on its review, the staff finds that the Holtec ERP briefly describes the responsibilities of personnel should an accident occur, including the prompt notification of offsite response personnel and the NRC, as well as responsibilities for developing, maintaining, and updating the plan. Therefore, the staff determined that the Holtec ERP contains sufficient information to meet the requirements of 10 CFR 72.32(a)(7).

10.3.5.8 Notification and Coordination

The staff reviewed section 4.3.12, “Activation of the ERP,” of the Holtec ERP, which contains a commitment to promptly notify offsite response organizations. The applicant stated that it will notify the NRC immediately after notifying offsite response organizations, as required by 10 CFR 72.32(a)(8), no later than 1 hour after declaration of an emergency.

The staff also reviewed section 4.3.11 of the Holtec ERP, which describes the means to notify offsite response organizations; the means to request offsite assistance, including medical assistance; and the identification of the personnel responsible for making the notifications. The

Emergency Coordinator or designee is responsible to activate the ERO and make notifications in a timely manner under accident conditions, during normal and off-normal hours. The applicant described diverse methods of notification in section 6.2, “Communications Equipment,” of the Holtec ERP.

The staff noted that the Holtec ERP includes a primary emergency operations center for management and support personnel to carry out coordinated emergency response activities. The emergency operations center has accident assessment and communication capabilities, which allow notification and activation to be performed even if some personnel, equipment, or parts of the facility are unavailable.

Based on its review, the staff finds that the Holtec ERP (1) contains a commitment to and a brief description of the means to promptly notify offsite response organizations, including the NRC Operations Center, and request offsite assistance and (2) provides a process to establish a control point and a plan for notification and coordination. Therefore, the staff determined that the Holtec ERP contains sufficient information to meet the requirements of 10 CFR 72.32(a)(8).

10.3.5.9 Information To Be Communicated

The staff reviewed section 4.3.12 of the Holtec ERP, which states that whenever an emergency notification is made, the following information will be provided if requested:

- Emergency Coordinator’s name and telephone number
- facility name, location, and status
- time and type of incident (e.g., radioactive release, fire)
- type and quantity of materials involved, to the extent known
- extent of injuries, if any
- extent of radioactive release, to the extent known
- possible hazards to human health and the environment outside the facility
- recommended protective actions, if necessary

Based on its review, the staff finds that the Holtec ERP briefly describes the types of information about facility status, radioactive releases, and recommended protective actions, if necessary, to be given to offsite response organizations and the NRC consistent with NRC guidance in RG 3.67. Therefore, the staff determined that the Holtec ERP contains sufficient information to meet the requirements of 10 CFR 72.32(a)(9).

10.3.5.10 Training

The staff reviewed section 7.2, “Training,” of the Holtec ERP, which describes the general employee training provided to all facility employees who may have to take protective actions (e.g., assembly, evacuation) in the event of an operational emergency. This section states that specialized training is provided to personnel directly involved in emergency response actions. In section 7.2.2, “Emergency Response Personnel Training,” of the Holtec ERP, the applicant stated that ERO training is conducted in accordance with the requirements of the HI-STORE CIS Facility training and qualification program. ERO training will include, but is not limited to, Radiation Worker II, 24-hour hazardous waste operations and emergency response (HAZWOPER) training, how to use personal protection equipment (respirators, eye and ear protection, breathing apparatus, protective clothing), and how to perform basic first aid. Emergency response personnel are not certified fire fighters but do understand the correct methods and techniques for eliminating and responding to fire emergencies.

The applicant stated that drills and exercises are part of the training curriculum. With respect to fires, Holtec will offer radiological response training to the fire departments of the communities of Maljamar and Monument and the city of Eunice in New Mexico.

With respect to medical training, the applicant stated in section 5.3.5 of the Holtec ERP, "Mitigation of Injuries," that the staff at Lea Regional Medical Center in Hobbs, New Mexico, are trained to handle radioactive material incidents and have their own decontamination procedures. The staff reviewed section 7.6, "Letters of Agreement," of the Holtec ERP, which states that upon request, the applicant will provide training for physicians that will include types of radiation, radiation detection and risks, signs and symptoms of radiation exposure, contamination control, and methods of decontamination.

Based on its review, the staff finds that the Holtec ERP briefly describes the training that will be provided to workers on how to respond to an emergency and any special instructions and orientation tours that will be offered to fire, police, medical, and other emergency personnel. Therefore, staff determined that the Holtec ERP contains sufficient information to meet the requirements of 10 CFR 72.32(a)(10).

10.3.5.11 Safe Condition

The staff reviewed section 9.0, "Recovery and Facility Restoration," of the Holtec ERP, which describes the means for restoring the Holtec facility to safe operation after an accident.

The applicant stated that the CIS Facility Site Manager shall ensure the following items are addressed before initiating the recovery plan:

- recovery strategy
- recovery tasks and assignments
- regulatory notifications and follow-ups
- insurance and risk management notification
- logistical support needs
- offsite logistical support needs
- appointment of a Recovery Manager

The ERP states that the Emergency Coordinator is responsible for determining when an emergency is sufficiently stable to enter the recovery phase. The Emergency Coordinator, through the Public Information Officer, disseminates information about the relaxation of public protective actions.

The ERP states that the recovery organization develops and coordinates plans and schedules for recovery operations. The ERP describes recovery to include those actions necessary to return an incident and the surrounding environment to pre-emergency conditions. The applicable Director of Operations, or designee, will serve as the Recovery Manager. The Recovery Manager will establish recovery teams to restore all vital systems back to normal operation. Exposure levels are established for estimating dosage and for protecting workers and the general public from hazardous exposure during recovery activities.

Based on its review, the staff finds that the Holtec ERP briefly describes the means for restoring the facility to safe operation after an accident consistent with NRC guidance in RG 3.67 and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(11).

10.3.5.12 Exercises

The staff reviewed section 7.3, “Drills and Exercises,” of the Holtec ERP, which describes the drill and exercise program, including biennial exercises, annual radiological drills, annual medical drills, and annual fire drills. With respect to fires, the applicant also committed to meet applicable National Fire Protection Association standards. The ERP states that offsite response organizations will be invited to participate in the drills and exercises, which is consistent with NRC guidance that recommends but does not require participation.

Consistent with the requirements in 10 CFR 72.32(a)(12), the Holtec ERP describes documented quarterly communications checks with offsite response organizations to include checking and updating all necessary telephone numbers.

The Holtec ERP describes the evaluation of drills, correction of identified deficiencies, and the confidentiality of exercise scenarios. Drills and exercises will be evaluated in accordance with the implementing procedures for emergency response training and drills. The applicant described that it will critique each drill and exercise using individuals not having direct implementation responsibility for the plan. Critiques of exercises will evaluate the appropriateness of the ERO, emergency procedures, facilities, equipment, training of personnel, and overall effectiveness of an incident response. Deficiencies found by the drill and exercise evaluations will be corrected using the Holtec CIS Facility corrective action process.

Based on its review, the staff finds that the Holtec ERP describes the program for conduct of exercises, drills, and communications tests and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(12).

10.3.5.13 Hazardous Chemicals

The staff reviewed section 10.0, “Compliance with Community Right-to-Know Act,” of the Holtec ERP, which states that Holtec complies with the Community Right-to-Know Act.

Based on its review, the staff finds that the Holtec ERP certifies that the licensee has met its responsibilities under the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Public Law 99-499, and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(13).

10.3.5.14 Comments on the Emergency Response Plan

The applicant stated that the Holtec ERP was given to offsite response organizations for review. The applicant also stated that Holtec currently has fully executed memorandums of understanding with the same offsite response organizations in support of the existing operating waste facility.

Based on its review, the staff finds that offsite response organizations expected to respond in case of an accident have been provided the opportunity to comment on the initial submission of the Holtec ERP before submitting it to the NRC, and comments received were submitted to the NRC with the ERP. Therefore, the staff determined that the Holtec ERP contains sufficient information to meet the requirements of 10 CFR 72.32(a)(14).

10.3.5.15 Offsite Assistance

The staff reviewed section 4.3.11 of the Holtec ERP, which describes the means for requesting assistance from offsite response organizations when necessary. The Emergency Coordinator or designee is responsible for alerting local authorities to emergencies that may affect the environment or public safety outside of the facility.

The Holtec ERP describes local offsite organizations that may assist the facility during an emergency. In section 7.6 of the Holtec ERP, the applicant stated that letters of agreements will be reviewed annually and renewed at least every 4 years, and that training will be offered to offsite response organizations.

Based on its review, the staff finds that the Holtec ERP briefly describes the arrangements for requesting and effectively using offsite assistance and, therefore, contains sufficient information to meet the requirements of 10 CFR 72.32(a)(15).

10.3.5.16 Public Information

The staff reviewed section 5.1.4, “Public Information Program,” of the Holtec ERP, which describes the arrangements for providing information to the public. The Holtec ERP states that depending on the severity of the event, the potential public impact, and the level of public interest, either the CIS Facility Site General Manager (who serves as the Public Information Officer) or the Holtec Corporate Office are the only employees who have authority to disseminate emergency-related public information. The Emergency Coordinator will establish a liaison who will communicate real-time emergency event information directly to the CIS Facility Site General Manager. Based on its review, the staff finds that the information in the Holtec ERP meets the requirements of 10 CFR 72.32(a)(16) because it briefly describes the arrangements for providing information to the public.

10.3.5.17 License Condition

In accordance with the regulatory requirement in 10 CFR 72.44(f), the staff is imposing the following license condition (License Condition 13) about the Holtec ERP:

> The licensee shall follow the “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility –- Emergency Response Plan,” HI-2177535, Revision 5, dated November 17, 2022, and as further supplemented and revised in accordance with 10 CFR 72.44(f).

The staff reviewed and evaluated the Holtec ERP, Revision 5, dated November 17, 2022, and finds that the Holtec ERP and the staff-imposed license condition meet the requirements of 10 CFR 72.32(a) and 10 CFR 72.44(f) and will provide reasonable assurance to adequately protect public health and safety and the environment in the event of an emergency.

10.3.6 Physical Security and Safeguards Contingency Plans

The following sections describe the staff's review and evaluation of the applicant's PSP (nonpublic) in support of the application to license the Holtec CIS Facility. The staff based its review on Revision 3 of the PSP and SCP (nonpublic), Revision 2 of the TQP (nonpublic), and the applicant's responses to NRC requests for additional information, to determine conformance with the applicable regulations and additional security measures specified in section 10.2 of this chapter.

10.3.6.1 Facility Description

SAR section 2.2, "Nearby Industrial, Transportation, Military, and Nuclear Facilities," addresses the hazards for the proposed facility and that an evaluation has been completed. SER chapter 2 discusses hazards for the proposed site.

Based on its review of the Holtec HI-STORE CIS Facility PSP, the staff finds that the applicant provided an adequate assessment of hazards that demonstrate that there is no adverse impact on security forces to include offsite forces. Additionally, the applicant provided a description of the facility and site that includes site maps showing the cask storage area, important supporting structures, and the boundaries of the protected area, as well as descriptions of the area adjacent to the site.

10.3.6.2 General Performance Objectives

In its review of the PSP, the staff reviewed the applicant's general objectives for the physical protection system to determine whether it provides high assurance that activities involving SNF do not constitute an unreasonable risk to public health and safety.

To achieve this objective, the physical protection system must provide for the following performance capabilities in accordance with 10 CFR 73.51(b):

- Store SNF and high-level radioactive waste only within a protected area.
- Grant access to the protected area only to individuals who are authorized to enter the protected area.
- Detect and assess unauthorized penetration of, or activities within, the protected area.
- Provide timely communication to a designated response force whenever necessary.
- Manage the physical protection organization in a manner that maintains its effectiveness.

In addition, the regulations at 10 CFR 73.51(b)(3) require that the physical protection system be designed to protect against loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose specified in 10 CFR 72.106(b) from any design-basis accident.

Based on its review, the staff finds that the Holtec CIS Facility PSP meets the general design objective because (1) it includes commitments that meet the requirements of 10 CFR 72.180 and 10 CFR 73.51, and (2) the NRC will impose the following license conditions related to the NRC's "Additional Security Measures for the Physical Protection of Dry Independent Spent Fuel Storage Installations," and the "Additional Security Measures for Access Authorization and Fingerprinting at Independent Spent Fuel Storage Installations":

LC14-1 The licensee shall follow the Physical Protection Plan entitled, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility - Physical Security Plan," HI-2177559, Revision 3, dated March 2, 2020, as well as changes made in accordance with 10 CFR 72.44(e) and 72.186(b).

LC14-2 The licensee shall follow the Safeguards Contingency Plan entitled, "Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility - Safeguards Contingency Plan," HI-2177560, Revision 3, dated March 2, 2020, as well as changes made in accordance with 10 CFR 72.44(e) and 10 CFR 72.186(b).

LC14-3 The licensee shall follow the Guard Training and Qualification Plan entitled "Holtec International & Eddy Lea Energy Alliance (ELEA) CISF Security Training and Qualification Plan," HI-2177561, Revision 2, dated March 30, 2019, as well as changes made in accordance with 10 CFR 72.44(e) and 10 CFR 72.186(b).

LC14-4 The licensee shall follow the "Additional Security Measures for the Physical Protection of Dry Independent Spent Fuel Storage Installations," dated September 28, 2007.

LC14-5 The licensee shall follow the "Additional Security Measures for Access Authorization and Fingerprinting at Independent Spent Fuel Storage Installations," dated February 4, 2016.

In addition, the staff finds that the applicant meets the plan retention requirements of 10 CFR 73.51(c) because the Holtec CIS Facility PSP includes a commitment to retain the following plans for 3 years after they are updated or superseded: the Holtec CIS Facility TQP and the Holtec CIS Facility SCP.

10.3.6.3 Physical Barrier Systems

The regulations at 10 CFR 73.51(d)(1) state that SNF must be stored only within a protected area so that access to this material requires passage through or penetration of two physical barriers, one barrier at the perimeter of the protected area and one barrier offering substantial penetration resistance.

In the PSP, the applicant provided for SNF to be stored within a protected area such that access to stored SNF requires passage through or penetration of at least two security barriers. The applicant described the first barrier as the protected area barrier composed of double fences that meet the definition of physical barrier in 10 CFR 73.2, "Definitions," and 20-foot isolation zones between the outer and inner fences as well as inside the protected area barrier fence. The applicant further described how the inner isolation zone is provided with an intrusion detection system that can detect penetrations through the protected area isolation zone. The applicant identified the second barrier as the VVMs, which are in-ground and capped with massive steel and concrete lids.

In addition to these physical barriers, the PSP describes a vehicle barrier system (VBS). As discussed in SAR section 1.1, "General Description of Installation," the applicant provided design specifications for the VBS. In SER section 15.3.2.4, "Flood," the staff confirms that these VBS design specifications provide some protection from design basis flood events.

Based on its review of the provisions described above, the staff finds that the commitments in the Holtec CIS Facility PSP for physical barrier systems meet the requirements of 10 CFR 73.51(d)(1) and facility licensing because, once the barriers are installed, an NRC preoperational inspection of the facility will confirm their installation.

10.3.6.4 Illumination

The regulations at 10 CFR 73.51(d)(2) require that illumination must be sufficient to permit adequate assessment of unauthorized penetrations of or activities within the protected area.

Based on its review, the staff finds that the commitments in the PSP to provide adequate illumination to allow surveillance and assessment within the protected area meet the illumination requirements of 10 CFR 73.51(d)(2) and, once the illumination is installed, an NRC preoperational performance inspection will confirm sufficient illumination levels.

10.3.6.5 Surveillance

The regulations at 10 CFR 73.51(d)(3) require that the perimeter of the protected area must be subject to continual surveillance and be protected by an active intrusion alarm system, capable of detecting penetrations through the isolation zone, that is monitored in a continually staffed primary alarm station and in one additional continually staffed location. In addition, the regulations require the following:

- The central alarm station must be located within the protected area and have bullet-resisting walls, doors, ceiling, and floor.
- The interior of the station must not be visible from outside the protected area.
- A timely means for assessment of alarms must be provided.

For alarm monitoring, the redundant location need only provide a summary indication that an alarm has been generated.

In the PSP, the applicant committed to have the capability to detect unauthorized penetrations through the isolation zones at the perimeter of the protected area consistent with the guidance in RG 5.44, Revision 3, “Perimeter Intrusion Alarm Systems,” issued October 1997. As described in the PSP, the intrusion detection system is contiguously located around the protected area and includes features comparable to those systems described in RG 5.44.

In the PSP, the applicant described a central alarm station located in the Holtec CIS Facility security building. The applicant described the central alarm station as a hardened facility within the protected area that is protected by the protected area intrusion detection system, access control, and barriers. The security building monitors all access control and intrusion alarms. A summary indication of alarms also annunciates in an offsite remote monitoring station.

Based on its review, the staff finds that the commitments in the PSP for alarm surveillance and annunciation meet the requirements of 10 CFR 73.51(d)(3), and once the surveillance systems are installed, an NRC preoperational inspection will confirm installation of these surveillance systems.

10.3.6.6 Security Patrols

The regulations at 10 CFR 73.51(d)(4) require that the protected area must be monitored by daily random patrols.

The applicant provided for security force personnel on duty at all times. Normal duties include the operation of the central alarm station and control of personnel entry, including searches of persons who enter the protected area. Security force personnel conduct daily random patrols to monitor the protected area boundaries for the presence of unauthorized persons or activities and for barrier degradation.

Based on its review, the staff finds the commitments in the PSP for patrols meet the requirements of 10 CFR 73.51(d)(4), and an NRC preoperational inspection will confirm implementation of these security patrols.

10.3.6.7 Security Organization

The regulations at 10 CFR 73.51(d)(5) require that the applicant establish a security organization with written procedures. The security organization must include sufficient personnel per shift to provide for monitoring of detection systems and the conduct of surveillance, assessment, access control, and communications to assure adequate response. The applicant must train, equip, qualify, and requalify members of the security organization to perform assigned job duties in accordance with Appendix B, “General Criteria for Security Personnel,” to 10 CFR Part 73, “Physical Protection of Plants and Materials,” Sections I.A.1.a, I.A.1.b, I.B.1.a, and the applicable portions of Section II.

The applicant committed to establishing a security organization that includes trained individuals, oversight, and written procedures to carry out security duties. The applicant stated in the PSP that a shift has sufficient armed individuals to meet regulatory requirements and the potential to increase shift manning levels depending on planned activities. In addition, the applicant will provide guard training and qualification sufficient to meet the commitments described in the Holtec TQP for the use of weapons.

Based on its review, the staff finds the commitments in the PSP for the establishment of a security organization with written procedures meet the requirements of 10 CFR 73.51(d)(5), and an NRC preoperational inspection will confirm the implementation of the security organization and written procedures.

10.3.6.8 Qualifications for Employment in Security

The applicant committed to screen individuals, including security personnel, granted unescorted access to the protected area where SNF is stored before granting such access. The regulations require that security force personnel shall meet the requirements of 10 CFR Part 73, Appendix B, Sections I.A.1.a, I.A.1.b, I.B.1.a, and the applicable portions of Section II. The Holtec CIS Facility access authorization plan includes screening with a 3-year local criminal history check of the individual within the 3-year period before assignment, a psychological evaluation, a Federal Bureau of Investigation criminal history records check, and a behavioral observation.

Based on its review, the staff finds the commitments in the PSP and TQP for the screening and qualification for employment in security meet the requirements of 10 CFR 73.51(d)(5), and an NRC preoperational inspection will confirm implementation of personnel screening and qualification.

10.3.6.9 Security Force Training

The applicant provided the Holtec CIS Facility TQP. The plan documents how the applicant meets the applicable criteria of Appendix B to 10 CFR Part 73.

In the TQP, the applicant stated that security personnel assigned to implement the physical protection program will meet minimum training and qualification requirements, and each individual will have the knowledge, skills, and abilities to carry out assigned duties and responsibilities. The applicant further stated that the Holtec CIS Facility security personnel will receive training consistent with the TQP before assuming any security-related job tasks. To ensure individuals maintain knowledge, skills, and abilities after the initial qualification, the Holtec security training and qualification program provides periodic requalification that is consistent with the time period specified in Part 73 Appendix B. The Holtec CIS Facility security personnel training includes firearms training consistent with the TQP.

Based on the preceding discussion, the staff finds the commitments in the TQP for the security organization meet the requirements of 10 CFR 73.51(d)(5), and once the TQP is implemented, an NRC preoperational inspection will confirm the Holtec CIS Facility security organization and training.

10.3.6.10 Response Liaison

The regulations at 10 CFR 73.51(d)(6) require the applicant to establish a documented liaison with a designated offsite response force or local law enforcement agency (LLEA) to permit timely response to unauthorized penetration or activities. The applicant included a site SCP (nonpublic) that documents a liaison with the Lea County Sheriff's Office in Lovington, New Mexico. The SCP also identifies the Hobbs and Carlsbad Police Departments, the Eddy County Sheriff, and the New Mexico State Police as potential responders to the Holtec CIS Facility. The response of the LLEAs combined with the integrated response and recovery plan of the PSP provide for a timely response.

Based on its review, the staff finds the commitments in the SCP for offsite response meet the requirements of 10 CFR 73.51(d)(6), and once the offsite response commitments are implemented, an NRC preoperational inspection will confirm their implementation.

10.3.6.11 Identification and Controlled Lock Systems

The regulations at 10 CFR 73.51(d)(7) require an applicant to establish and maintain a personnel identification system and a controlled lock system to limit access to authorized individuals.

The applicant included in its PSP an identification system to provide unique identification of individuals granted unescorted access to the protected area at the facility. In addition, the identification system will identify individuals requiring escort while within the protected area.

The applicant committed to establishing a key-and-lock control system to limit access to and within the protected area to authorized individuals.

Based on its review, the staff finds the commitments in the PSP for identification and controlled lock systems meet the requirements of 10 CFR 73.51(d)(7), and once the commitments are implemented, an NRC preoperational inspection will confirm their implementation.

10.3.6.12 Communications Capability

The regulations at 10 CFR 73.51(d)(8) require the applicant to provide redundant communications capability between onsite security force members and the designated response force or LLEA.

In its PSP, the applicant committed to equip each security individual with a two-way radio that has an additional channel to maintain continuous communications with the security posts. The PSP also identifies a primary alarm station with a base radio system and a satellite telephone to maintain contact with the LLEA. An uninterruptible power supply and a generator provide backup power to onsite communication and, therefore, redundant communications capability between the onsite security force and the offsite response force.

Based on its review, the staff finds the commitments in the PSP for communications capability meet the requirements in 10 CFR 73.51(d)(8), and once implemented, an NRC preoperational inspection will confirm communication capability.

10.3.6.13 Access Controls at the Protected Area

The regulations at 10 CFR 73.51(d)(9) require that all individuals, vehicles, and hand-carried packages entering the protected area must be checked for proper authorization and visually searched for explosives before entry.

10.3.6.13.1 Access to Protected Areas

In the PSP, the applicant committed to develop procedures for granting access of vehicles and packages into the protected area. All individuals, packages, vehicles, and materials are searched visually and physically (or by detection equipment if available) for firearms, incendiary devices, explosives, and other items that could be used to take control of the facility or commit radiological sabotage.

10.3.6.13.2 Access Controls at the Protected Area

In the PSP, the applicant committed to develop procedures for granting access of vehicles and packages into the protected area. All individuals, packages, vehicles, and materials are visually and physically (or by detection equipment if available) searched for firearms, incendiary devices, explosives, and other items that could be used to take control of the facility or commit radiological sabotage.

10.3.6.13.3 Escorts and Escorted Individuals

In the PSP, the applicant identified the individuals designated and granted unescorted access into the protected area and described the requirements and procedures for escorting individuals needing access.

10.3.6.13.4 Access Controls at the Protected Area—Staff Evaluation Finding

Based on the preceding discussions, the staff finds that the commitments in the PSP for access control commitments meet the requirements of 10 CFR 73.51(d)(9), and once the access control measures are implemented, an NRC preoperational inspection will confirm their implementation.

10.3.6.14 Procedures

The regulations at 10 CFR 73.51(d)(10) require that written response procedures be established and maintained for addressing unauthorized activities within the protected area including category 5, “Procedures,” of Appendix C, “Licensee Safeguards Contingency Plans,” to 10 CFR Part 73. In addition, the regulations require that the applicant retain a copy of response procedures as a record for 3 years or until termination of the license for which the procedures were developed and retain copies of superseded material for 3 years after each change or until termination of the license.

In the SCP, the applicant described its response procedures for dealing with detection of unauthorized presence or activities within the protected area and detailed the actions to be taken and decisions to be made by each member or unit of the response organization.

Based on its review, the staff finds the commitments in the SCP to develop security procedures meet the requirements of 10 CFR 73.51(d)(10), and once the security procedures are established, an NRC preoperational inspection will review them.

10.3.6.15 Equipment Operability

The regulations at 10 CFR 73.51(d)(11) require that all detection systems and supporting subsystems must be tamper indicating with line supervision and must be maintained in an operable condition, including the surveillance/assessment and illumination systems. The regulations also require the licensee to take timely compensatory measures after discovery of an inoperable condition to ensure no reduction in the effectiveness of the security system.

In the PSP, the applicant committed to test physical protection systems and equipment before facility operation to applicable manufacturers' specifications and the requirements of the PSP. The applicant also committed to a repair and preventive maintenance program and interim compensatory measures until restoration of the system to normal capability. Finally, the applicant committed to implementing the guidance in RG 5.44 for detection systems that are tested for operability at least once every 7 days and performance tested at least semiannually.

Based on its review, the staff finds the commitments in the PSP for equipment operability meet the requirements of 10 CFR 73.51(d)(11), and once the measures to assure equipment operability are implemented, an NRC preoperational inspection will confirm their implementation.

10.3.6.16 Audits

The regulations at 10 CFR 73.51(d)(12) require a review of the physical protection program once every 24 months by individuals independent of both physical protection program management and personnel who have direct responsibility for implementation of the physical protection program. The physical protection program review must include an evaluation of the effectiveness of the physical protection system and a verification of the liaison established with the designated response force or LLEA.

In the PSP, the applicant committed to conduct security audits at least every 24 months by individuals knowledgeable of security operations and systems who are independent of both security program management and of personnel directly responsible for implementation of the security program. The PSP states that the audits will include an evaluation of the effectiveness of the physical protection system and verification of the liaison established with the LLEA, and that the applicant will maintain reports in a form sufficient for auditing and available for inspection for a period of 3 years.

Based on its review, the staff finds that the commitments in the PSP for the audit program meet the requirements of 10 CFR 73.51(d)(12), and once the audit program commitments are implemented, an NRC preoperational inspection will confirm their implementation.

10.3.6.17 Documentation

The regulations at 10 CFR 73.51(d)(13) require retention of documentation as a record for 3 years after the record is made or until termination of the license. Duplicate records to those required under 10 CFR 72.180 and 10 CFR 73.71 need not be retained under the requirements of this section.

In its physical protection program, the applicant described maintaining the following records for a period of 3 years:

- access authorization records
- training and qualification records required by Section II.B of Appendix B to 10 CFR Part 73
- log of individuals granted access to the protected area
- safeguards event report records
- log of all patrols
- physical protection program audit reports
- a record and assessment of each alarm

Based on its review, the staff finds the commitments in the PSP for recordkeeping meet the requirements of 10 CFR 73.51(d)(13), and once the documentation commitments are implemented, an NRC preoperational inspection will confirm their implementation.

10.4 Evaluation Findings

Based on its review, the staff finds the following:

- The SAR includes an acceptable description of the program covering the quality assurance program, preoperational testing, and initial operations in compliance with 10 CFR 72.24(h), (n), and (p).
- The SAR includes an acceptable description of the detailed security measures for physical protection, including design features and the plans required by 10 CFR Part 72, Subpart H.
- The applicant has provided acceptable technical qualifications, including training and experience, for personnel who will be engaged in the proposed activities, in compliance with 10 CFR 72.28(a).
- The application includes an acceptable description of a personnel training program to comply with 10 CFR Part 72, Subpart I.
- The application includes an adequate, acceptable description of the applicant's operating organization, delegations of responsibility and authority, and the minimum

skills and experience qualifications relevant to the various levels of responsibility and authority in compliance with 10 CFR 72.28(c).

- The application includes an acceptable commitment by the applicant to have and maintain an adequate complement of trained and certified installation personnel before receipt of SNF or high-level radioactive waste for storage in compliance with 10 CFR 72.28(d).

- The application includes an emergency plan in compliance with 10 CFR 72.32.

- The application provides acceptable assurance that the applicant is qualified by reason of training and experience to conduct the operations covered by the regulations in 10 CFR Part 72, in compliance with 10 CFR 72.40(a)(4).

- The application provides acceptable assurance about the management, organization, and planning for preoperational testing and initial operations that the activities authorized by the license can be conducted without endangering public health and safety in compliance with 10 CFR 72.40(a)(13).

- The SAR includes an acceptable plan for the conduct of operations in compliance with 10 CFR 72.24(h).

In addition, the staff finds that Revision 3 of the Holtec CIS Facility PSP and SCP, to include Revision 2 of the TQP, is adequate and meets the requirements of 10 CFR 72.180, 10 CFR 73.51, the “Additional Security Measures for the Physical Protection of Dry Independent Spent Fuel Storage Installations,” and the “Additional Security Measures for Access Authorization and Fingerprinting at Independent Spent Fuel Storage Installations.” Further, once the applicant’s physical protection program is fully implemented and confirmed through an NRC preoperational inspection, the staff finds the applicant's program will satisfy the provisions of 10 CFR 72.40(a)(14) by providing for the common defense and security and the protection of public health and safety.

10.5 References

American National Standards Institute, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment,” ANSI N14.5-2014, June 2014.

American National Standards Institute/American Nuclear Society, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” ANSI/ANS 3.1-2014, reaffirmed February 2020.

Code of Federal Regulations, Title 10, Part 55, “Operators’ Licenses.”

Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.”

Code of Federal Regulations, Title 10, Part 73, “Physical Protection of Plants and Materials.”

Holtec International, “Final Safety Analysis Report on the HI-STORM FW MPC Storage System,” Holtec Report No. HI-2114830, Revision 4, Docket No. 72-1032, June 24, 2015. ML15177A338.

Holtec International, “Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System,” Holtec Report No. HI-2115090, Revision 3, Docket No. 72-1040, June 29, 2016. ML16193A339.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) CISF—Security Training and Qualification Plan,” Revision 2 (nonpublic), Report No. HI-2177561, H-037-17, transmittal dated March 30, 2019. ML19094A271.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Physical Security Plan,” Revision 3 (nonpublic), Report No. HI-2177559, H-011-20, transmittal dated March 2, 2020. ML20065H155.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Safeguards Contingency Plan,” Revision 3 (nonpublic), Report No. HI-2177560, H-012-20, transmittal dated March 2, 2020. ML20065H155.

Holtec International, “Holtec International & Eddy Lea Energy Alliance Underground Consolidated Interim Storage Facility—Training and Qualification Program” (proprietary), Holtec Report No. HI-2177562, Revision 3, Docket No. 72-1051; October 9, 2020. ML20283A797.

Holtec International, “Nuclear Quality Assurance Manual,” Revision 15 (proprietary), Docket No. 71-0784, submitted June 10, 2022. ML22161A072.

Holtec International, “HI-STORE CIS Facility Fire Protection Plan” (proprietary), Holtec Report No. HI-2177938, Revision 1, Docket No. 72-1051, November 17, 2022. ML22331A016.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Emergency Response Plan,” Revision 5, Report No. HI-2177535, Docket No. 72-1051, November 17, 2022. ML22331A015.

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Holtec Report No. HI-2167374, Revision 0T, Docket No. 72-1051, January 20, 2023. ML23025A112.

NRC, “Perimeter Intrusion Alarm Systems,” Regulatory Guide 5.44, Revision 3, October 1997. ML003739217.

NRC, “Standard Review Plan for Physical Protection Plans for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,” NUREG-1619, July 1998. ML020720453.

NRC, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” NUREG-1567, March 2000. ML003686776.

NRC, “Emergency Planning,” SFST-ISG-16, June 2000.

NRC, “Additional Security Measures for the Physical Protection of Dry Independent Spent Fuel Storage Installations,” September 28, 2007 (nonpublic). Safeguards LAN/Electronic Safe (SLES) Reference No. NS106675.

NRC, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Regulatory Guide 1.21, Revision 2, June 2009. ML091170109.

NRC, “Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,” Regulatory Guide 3.67, Revision 1, April 2011. ML103360487.

NRC, “Additional Security Measures for Access Authorization and Fingerprinting at Independent Spent Fuel Storage Installations,” February 4, 2016. ML17079A277.

11 RADIATION PROTECTION EVALUATION

In chapter 11, “Radiation Protection Evaluation,” of its safety analysis report (SAR), Revision 0T, dated January 20, 2023, Holtec International (the applicant) described the radiation protection features of the proposed HI-STORE Consolidated Interim Storage (CIS) Facility to ensure that radiation exposures to workers and members of the public meet the regulatory requirements.

11.1 Scope of Review

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated radiation protection for the HI-STORE CIS Facility by reviewing the SAR, documents cited in or attached to the SAR, the applicant’s responses to the staff’s requests for supplemental and additional information, and other relevant literature. The staff reviewed the information to determine whether the applicant’s radiation protection program provides sufficient assurance of the following:

- Radiation exposures and radionuclide releases will be maintained at levels that are as low as is reasonably achievable (ALARA).
- Occupational radiation doses will not exceed the limits specified in the radiation protection standards in Title 10 of the *Code of Federal Regulations* (10 CFR).
- Radiation doses to the public during normal conditions and anticipated occurrences will meet regulatory standards.

11.2 Regulatory Requirements

The NRC requirements relevant to the staff’s evaluation of establishing a radiation protection program at the proposed HI-STORE CIS Facility appear in the following 10 CFR sections:

- 10 CFR 20.1101, “Radiation protection programs”
- 10 CFR 20.1201, “Occupational dose limits for adults”
- 10 CFR 20.1301, “Dose limits for individual members of the public”
- 10 CFR 20.1302, “Compliance with dose limits for individual members of the public”
- 10 CFR 20.1406, “Minimization of contamination”
- 10 CFR 20.1501, “General”
- 10 CFR 20.1701, “Use of process or other engineering controls”
- 10 CFR 20.1702, “Use of other controls”
- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.40, “Issuance of license”
- 10 CFR 72.44, “License conditions”

- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

11.3 Staff Review and Analysis

The applicant proposed the HI-STORM UMAX canister storage system for spent fuel storage at the HI-STORE CIS Facility. This underground system is designed to store spent fuel in a dry configuration in an underground independent spent fuel storage installation (ISFSI). The NRC has certified this HI-STORM UMAX system, as described in the HI-STORM UMAX canister storage system final safety analysis report (FSAR), Revision 3, dated June 29, 2016, under NRC Docket 72-1040.

Unless otherwise stated, the staff reviewed and evaluated the radiation protection evaluation of the proposed site discussed in chapter 11 and other relevant sections of SAR Revision 0T, documents cited in or attached to the SAR, the applicant’s responses to the staff’s requests for additional information, and other relevant literature, including information in the HI-STORM UMAX canister storage system FSAR relevant to radiation protection for the HI-STORE CIS Facility. The review of the radiation protection program and health physics of the proposed facility included SAR chapter 3, “Operations at the HI-STORE Facility,” chapter 4, “Design Criteria for the HI-STORE CIS SSCs,” chapter 7, “Shielding Evaluation,” chapter 10, “Conduct of Operations,” and chapter 11, and considered information in the references cited in safety evaluation report (SER) section 11.5.

The staff followed the guidance provided in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000. The following sections of this SER document the staff’s review of and conclusions about the radiation protection plan provided in the SAR.

11.3.1 As Low As Is Reasonably Achievable Considerations

The primary objective of the applicant’s health physics program is to keep radiation exposure to workers, visitors, and members of the public below regulatory limits and ALARA. The applicant described the health physics program that will be implemented for the HI-STORE CIS Facility in SAR section 3.1.4.6, “Health Physics Operations,” along with the policy and program for maintaining doses ALARA in SAR section 11.1, “As Low As Reasonably Achievable Considerations.” For the HI-STORE CIS Facility, the applicant incorporated by reference the HI-STORM UMAX health physics program from section 11.1 of the HI-STORM UMAX FSAR for maintaining doses ALARA. As described in SAR section 11.1, the program and procedures will be revised and supplemented, as appropriate, to address facility-specific activities and to comply with the description of the ISFSI health physics program in SAR section 3.1, “Description of Operations.” SAR section 11.1.1 also states that the program for maintaining doses ALARA follows the guidance in Regulatory Guide (RG) 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable,” Revision 3, issued June 1978, and RG 8.10, “Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable,” Revision 2, issued August 2016.

11.3.1.1 Design Considerations

SAR section 11.1 describes the design considerations to maintain doses ALARA and delineates the following specific features of the HI-STORE CIS Facility:

- The shortest distance from the edge of the ISFSI pad at the HI-STORE CIS Facility to the control area boundary is 400 meters (1,300 feet). This provides adequate distance to minimize dose rate to offsite personnel, including members of the public. There is spacing between vertically ventilated modules (VVMs) for radiation workers to perform maintenance, clear the blockages of inlet ducts, and conduct surveillances.

- The fuel is stored dry inside the MPC-89 and MPC-37 metal canisters that are sealed by welding and backfilled with inert gas for fuel confinement. Therefore, there is no liquid in the cask, and there is no credible leakage of any radioactive material. Welded seals will prevent airborne radioactive materials from leaking from these multipurpose canisters (MPCs), and once sealed, fuel is not removed from the MPCs at any location outside of the refueling building. The storage system is passive and requires little maintenance.

- Placement of the storage casks underground provides significant shielding.

- Placement of the storage pads is at a sufficient distance 152 meters (500 feet) from administrative buildings so that doses to workers are maintained ALARA.

- The applicant will use a restricted area fence and a security perimeter fence with a locked gate to protect individuals against undue risks from radiation exposure and to prevent unauthorized access to the ISFSI.

- Thick biological shielding on the overpacks will provide gamma and neutron shielding.

The ISFSI pad, canister transfer building, and other facilities and operations at the site will be located within the owner-controlled boundary of the HI-STORE CIS Facility such that the transfer of MPCs containing spent nuclear fuel (SNF) from the cask transfer building to the VVMs will not take place on any public roads. No fuel transfer or SNF loading/unloading occurs at the facility. The ISFSI storage area at the HI-STORE CIS Facility incorporates an engineered soil providing significant shielding, as does each shielded MPC-89 and MPC-37 canister and HI-STORM UMAX overpack. The ISFSI pad is located at a sufficient distance from the controlled area boundary such that offsite exposures will be minimized further.

The staff finds that the design of the HI-STORE CIS Facility provides reasonable assurance that doses to personnel and members of the public will meet the requirements of 10 CFR 72.126(a) and be maintained ALARA. The staff also finds that the requirements for minimizing contamination and the amount of generated radioactive waste outlined in 10 CFR 20.1406 are satisfied. The staff finds that the design of the seal-welded MPCs, which are not opened at the facility, means that the MPCs will generate no effluents and meets the requirements of 10 CFR 72.126(d).

11.3.1.2 Operational Considerations

SAR sections 11.1.2, “Design Considerations,” and 11.1.3, “Operational Considerations,” describe the operational considerations to maintain doses ALARA. SAR chapter 3, summarizes the operating procedures for HI-STORE CIS Facility activities such as cask loading, unloading, and transfer to the ISFSI storage overpack, and the staff discusses them in SER chapters 4 and 10. Specifically, the program to maintain doses ALARA includes the following operational elements:

- Dry-run training will teach personnel about canister transfer procedures, verification of equipment operability and procedure efficiency, and minimization of radiation exposure.
- Controls at the nuclear power plant where fuel is loaded mean that no significant surface contamination will accumulate, resulting in little to no contamination to workers during canister transfer operations.
- Fuel canister transfer between the transport cask and HI-STORM UMAX VVM takes place within a shielded transfer cask.
- The HI-TRAC CS transfer cask will be moved to a HI-STORM UMAX VVM using the vertical cask transporter.
- Use of power-operated tools, when possible, in bolting operations will minimize personnel exposure time.

The staff finds that the applicant’s description of the operational considerations for maintaining doses ALARA satisfies the requirements of 10 CFR 72.24(e). The staff also finds that the described use of RG 8.8 and RG 8.10 in SAR section 11.1.3 for planning operations is appropriate and provides reasonable assurance along with the discussion in SER section 7.3.5 that doses to personnel and members of the public meet the regulatory requirements of 10 CFR 20.1501(a)(1), 10 CFR 72.104, and 10 CFR 72.106. Additionally, the staff finds that the operational considerations listed above meet the requirements of 10 CFR 20.1701 because the applicant will use engineering and process controls to prevent airborne radioactivity.

11.3.2 Radiation Protection Design Features

SAR chapter 11 contains information relevant to the facility’s proposed radiation design features.

11.3.2.1 Installation Design Features

SAR section 11.2, “Radiation Protection Design Features,” describes the facility radiation protection design features, which the applicant stated are in accordance with the facility and equipment design features in Position 2 of RG 8.8. The ISFSI will be located within the HI-STORE CIS Facility owner-controlled area and will house, in underground storage, 500 MPC-37 and MPC-89 canisters for pressurized-water reactor and boiling-water reactor SNF, respectively. All storage cells will contain casks holding SNF. The storage cells will be positioned as shown in SAR figure 1.2.1, which depicts an array of HI-STORM UMAX systems. Periodic inspections, placement of loaded storage casks, and routine security checks are the planned operations that will be conducted at the facility.

The staff finds that the use of regulatory position 2 of NRC RG 8.8, which provides guidance regarding facility and equipment features, as discussed in SAR section 11.2, “Radiation Protection Design Features,” is appropriate. The staff also finds that given the proposed design features described in the SAR, the applicant has satisfied the requirements of 10 CFR 72.126(a). For example, the facility is located away from populated areas, uses sealed canisters that are not opened, and uses shielded casks. SAR sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 4.1, 4.2, 4.3, 7.1.2, 7.4, and 11.2 and Technical Specification 3.2.1, “Canister Surface Contamination,” discuss design features that address radiation monitoring, control of airborne contaminants, instrumentation and controls, and other considerations related to maintaining doses ALARA.

The staff finds that the information in the SAR sections 11.1 and 11.2, including a description of a radiation protection program and measures to maintain containment of SNF in sealed canisters, minimize contamination, monitor radiation, and conduct dry runs of canister transfer operations, provides reasonable assurance that the use of the HI-STORM UMAX system for the HI-STORE CIS Facility will meet the regulatory requirements of 10 CFR 20.1101(b), 20.1101(d), 20.1406, 20.1701, and 20.1702(a). Chapters 3, 4, 7, 10 and 11 of this SER discuss the staff’s evaluations of the radiation shielding features and the confinement features during normal, off-normal, and accident conditions. Based on these evaluations, the staff finds that the radiation protection features for the proposed HI-STORE CIS Facility are acceptable.

11.3.2.2 Access Control

SAR sections 1.1, “General Description of Installation,” and 11.2.2, “Access Control,” describe access control to the facility. SAR figure 2.1.6(a) show the applicant’s property line and owner-controlled area. The applicant’s property line aligns with the controlled area boundary and extends well outside the restricted area boundary around the ISFSI, as shown in SAR figure 2.1.6, “Site Layout.” The applicant exercises authority to control all activities within the owner-controlled area boundaries. Access control to the restricted area around the ISFSI provides for both personnel radiation protection and facility physical protection.

Fences with lockable gates will be installed as indicated in SAR figure 2.1.6(c). As shown in figure 11.2.1, “HI STORE CIS Facility Radiation Protection Zones,” gates appear on the east side of the restricted area where the access road and rail line cross the restricted area boundary. The dose rate outside the controlled area, including direct radiation from ISFSI operations and any other radiation from uranium fuel cycle operations within the region, will not exceed 0.25 millisieverts per year (25 millirem (mrem) per year), as specified in 10 CFR 72.104(a).

Access to the restricted area is established through a single access point. Personal dosimetry is required for each individual entering the restricted area as stated in SAR section 11.2.2. Personal protective equipment such as anticontamination clothing and respirators is removed within the restricted area. The decontamination of personnel, if deemed necessary, is also performed in the security building. The cask storage area and canister transfer building are located in the restricted area.

The staff finds that the proposed access control at the HI-STORE CIS Facility as described in the SAR is acceptable because it provides security fencing and limits access by way of a single point of entry into radiologically controlled areas. Therefore, the staff finds that the requirements to control access are met in accordance with 10 CFR 72.126(a)(3).

The physical security plan describes the measures to prevent the entry of unauthorized personnel into radiologically controlled areas. The staff's review of the security program for the facility appears in section 10.3.6 of this SER.

11.3.2.3 Radiation Shielding

The NRC staff's shielding evaluation appears in chapter 7 of this SER. The staff reviewed the SAR shielding calculations and found them to be acceptable because the dose rates at the onsite and offsite locations were below the limits specified in Part 20 and 10 CFR 72.104(a) and 72.106(b). Based on its review of the information in SAR chapter 7 and the sample calculation files provided by the applicant in "HI-STORE CIS Facility Site Boundary Dose Rates Calculations for HI-STORM UMAX System," Revision 4, dated March 22, 2022, the staff finds that there is reasonable assurance that the ISFSI shielding was adequately analyzed and meets the requirements of 10 CFR 72.128(a)(2).

11.3.2.3.1 Shielding Configurations

Chapter 5 of the HI-STORM UMAX FSAR, Revision 3, includes shielding information relied on and incorporated by reference in chapter 7, and Table 7.0.1 of the HI-STORE CIS Facility SAR. Chapter 3 of the HI-STORM UMAX FSAR discusses shielding configuration, shielding materials, and geometries.

As noted immediately above, the staff evaluates the proposed facility's radiation shielding in SER chapter 7.

11.3.2.3.2 Confinement and Ventilation

The staff evaluates the HI-STORE UMAX confinement system in SER chapter 9 and site-generated waste confinement and management in SER chapter 14. Based on these evaluations and because there are no liquid or gaseous effluent releases, the staff finds that the requirements specified in 10 CFR 72.126(c)(1) are met given the HI-STORE CIS Facility design. Similarly, no ventilation is installed in the facility since there is no credible scenario that requires offgas or particulate filtration, as there is no fuel handling or opening of any canister that requires offgas or particulate filtration.

11.3.2.3.2.1 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The applicant discussed area radiation and airborne radioactivity monitoring instrumentation in SAR section 11.2.5, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation.” Area radiation and airborne radioactivity monitoring instrumentation is required by 10 CFR 72.122(h). All SNF at the ISFSI will be stored in sealed and welded MPCs in HI-STORM UMAX systems. According to the applicant, no credible events could result in the release of radioactive materials from within the MPCs. In addition, no credible events could increase dose rates from direct radiation from the casks stored at the ISFSI pads. For this reason, the applicant stated in SAR section 11.2.5 that the HI-STORE CIS Facility ISFSI pads do not need area radiation and airborne radioactivity monitors. Nonetheless, continuous monitoring and audible radiation level alarms will be used in the canister transfer building area. Additionally, thermoluminescent dosimeters (TLDs) with a lower sensitivity of 0.02 mrem (0.0002 mSv) will be used to monitor and record area doses at appropriate intervals along and within the restricted area fence. Hand-held radiation protection instruments and dosimeters will also be provided during canister transfer operations and routine maintenance at the ISFSI pads.

As discussed in SER Chapters 7, 9, and 15, the staff confirmed that there are no credible events that could result in the release of radioactive materials from within the MPCs or loss of shielding. Therefore, no radiological alarm systems are needed in storage pad areas, and the staff finds that the proposed monitoring in the CTB and the storage area meet the requirements of 10 CFR 72.126(b). The staff also finds that the radiation monitoring instrumentation described in the SAR meets the requirements of 10 CFR 72.126(c)(2) because continuous radiation monitoring will take place in the cask transfer building and TLDs will monitor radiation throughout the restricted area and at the restricted area fence. The staff further finds that actual dose rates around the facility will be monitored adequately to verify compliance with the radiological limits specified in 10 CFR 20.1301(a) and 10 CFR 72.104(a) for members of the public and that any unexpected increases in dose rates will be properly detected in a timely manner.

11.3.3 Dose Assessment

11.3.3.1 Onsite Doses

SAR chapter 7 presents dose assessments. SAR table 7.4.1, “Dose Rates from the HI-TRAC CS MPC-37 Design Basis Fuel with Bounding Source Terms from Reference,” provides dose rates at the surface, 0.5 meter (1.6 foot), 1 meter (3.3 feet), and 2 meters (6.6 feet) (from a HI-TRAC CS transfer cask. SAR table 7.4.2 includes dose rates from the surface and 1 meter (3.3 feet) from the HI-STORM UMAX system. SAR table 11.3.1 gives the estimated occupational exposures to personnel during the different phases of ISFSI operation, including but not limited to (1) receipt of the transport cask, (2) transport of the canister using the HI-TRAC CS transfer cask, (3) transfer of the cask from the HI-TRAC CS to the HI-STORM UMAX ISFSI, and (4) closing of the storage cell lid. SAR figure 3.1.1 lists the operational steps involved in unloading and loading an overpack. According to the applicant, personnel exposures will be below the annual occupational dose limit of 0.05 sievert (5 rem) specified in 10 CFR 20.1201(a) for all doses presented.

In addition, SAR table 11.3.1 includes the estimated number of personnel, the estimated dose rates, and the estimated time for each operational task. The applicant estimated the dose from loading, transferring, and emplacing a single HI-STORM UMAX canister in a storage overpack in the ISFSI VVM to be 1,218.7 mrem (12.187 mSv) for all crew during operations.

The estimated dose rates include both gamma and neutron components and are based on design-basis pressurized-water reactor fuel, as provided in SAR table 7.1.1. SER chapter 7 summarizes the staff's evaluation of the source terms and dose rates.

Section 11.3.4 of the HI-STORM UMAX FSAR, which the applicant incorporated by reference in SAR table 11.0.1, discusses the dose estimates for routine maintenance operations, surveillance, inspections of the facility.

11.3.3.2 Offsite Doses

SAR section 7.4, "Shielding Analyses Methods and Results," addresses offsite collective dose for normal conditions. SAR section 7.4.2.2, "Off-Normal and Accident Conditions," addresses off-normal and accident offsite doses, and sections 5.1 and 5.2 of the HI-STORE UMAX FSAR, Revision 3, analyze these conditions. SAR table 7.0.1 incorporates by reference the dose assessment methodologies from sections 5.1 and 5.2 of the HI-STORM UMAX FSAR.

The applicant summarized its shielding calculations in SAR chapter 7. SAR table 7.4.7, "Annual Dose to a Real Individual at Nearest Controlled Area Boundary," provides a total estimated annual dose to a real individual at the controlled area boundary of the HI-STORE CIS Facility assuming 2,000 hours per year of occupancy time. This dose includes the contributions to the annual dose to a real individual from the 500 fully loaded MPCs, the 10 HI-STAR 190 packages at the staging area, two MPCs in the HI-TRAC CS transfer casks or HI-STAR 190 packages in the canister transfer building, and the carbon-14 that is generated in the air passing through the gap between the VVM and the canister. The results of the calculation as presented in SAR table 7.4.7 yielded an expected dose below the regulatory limit in 10 CFR 72.104(a).

11.3.3.3 Staff Evaluation of Shielding Calculation

The staff reviewed the summary of shielding calculations presented in SAR section 7.4.2, "Dose and Dose Rate Estimates," and finds that the applicant has included contributions from all potential radiation sources. Since there are no fuel facilities or other human-made radiation sources, the staff considered the applicant's summary of shielding calculation results to be complete and acceptable. On these bases, the staff finds that the applicant's summary of shielding calculations meets the acceptance criterion and hence is acceptable.

11.3.3.4 Confirmatory Analysis

The staff, with technical support from Oak Ridge National Laboratory, performed confirmatory analyses on the source terms and dose rates from HI-TRAC CS casks, VVM storage modules, carbon-14 generated by the ISFSI, and HI-STAR 190 packages for the annual dose to a real individual at and beyond the controlled area boundary. The results of the confirmatory calculations show that there is a significant safety margin between the dose limits set forth in 10 CFR 72.104 and 10 CFR 72.106 and the expected annual dose to a real individual from the ISFSI operations. SER chapter 7 details the confirmatory calculations. Based on the confirmatory analysis, the NRC staff determined that there is reasonable assurance that the HI-STORE CIS Facility will meet the regulatory requirements of 10 CFR 72.104 and 10 CFR 72.106.

11.3.4 Health Physics Program

SAR section 3.1.4.6, "Health Physics Operations," describes the radiation protection program.

11.3.4.1 Organization

SAR section 3.1.4.6, describes health physics program operations and references the organization that will implement the health physics program during facility operations. SAR section 10.1, "Organizational Structure," and SAR figure 10.4.2 describe the program organization. The staff evaluates the facility organization in detail in SER section 10.3.1. The Radiation Protection Manager is responsible for health physics activities related to facility operations. The Radiation Protection Manager is independent of the Operations Manager. The Radiation Protection Manager and the Operations Manager both report directly to the Site Manager. The staff finds that the organization of the proposed radiation protection program satisfies 10 CFR 20.1101(a).

11.3.4.2 Equipment, Instrumentation, and Facilities

SAR section 11.4.2, "Equipment, Instrumentation, and Facilities," describes the equipment, instrumentation, and facilities pertinent to the facility radiation protection program.

The facility is located within the owner-controlled area, and the applicant has full authority to control all activities within the ISFSI and owner-controlled area boundaries.

The applicant described the portable and fixed instrumentation and facilities that will be used for facility operations and radiation surveys in accordance with the policies and procedures discussed in SAR section 11.4.3, "Policies and Procedures." The portable survey and personnel monitoring instrumentation during normal, off-normal, or accident conditions includes, but is not limited to, low-level contamination meters, beta/gamma survey meters, beta/gamma personnel friskers, and portable air samplers.

TLDs will be used to determine dose rates at the restricted area and owner-controlled area boundaries. No radioactive effluents are expected during ISFSI operations. In accordance with technical specification 5.5.1 and 10 CFR 72.44(d)(3), an annual report documenting the environmental monitoring program results will be submitted. The staff finds that compliance with the dose limits specified in 10 CFR 72.104(a) will be demonstrated using direct radiation measurements, thereby meeting the requirements of 10 CFR 20.1501(a)(1) and 10 CFR 20.1302(a).

11.3.4.3 Policies and Procedures

SAR section 11.4.3 describes the policies and procedures pertinent to the facility radiation protection program. The operation and use of radiation monitoring equipment will be described in written procedures consistent with the requirements of 10 CFR 20.1101(a), (b), and (c).

Additionally, the applicant proposed administrative control technical specification 5.5.2, “Radiation Protection Program.” This technical specification requires surface dose rate surveys to be performed and requires that the measured dose rates not exceed specified values before undergoing certain operations. The staff reviewed the proposed technical specification and found it acceptable because the HI-STORE CIS Facility will use the HI-STORM UMAX storage system, and the proposed technical specification provides the same controls as the corresponding technical specification for the radiation protection program for the HI-STORM UMAX system, Amendment No. 2, that the staff previously approved in the certificate of compliance technical specifications for the amended HI-STORM UMAX storage system dated January 6, 2017.

The staff finds that the radiation program policies and procedures described by the applicant in SAR section 11.4.3 and in technical specification 5.5.2 are acceptable and provide reasonable assurance that the requirements of 10 CFR 20.1101(a–c) and 10 CFR 72.44(c) are met.

11.4 Evaluation Findings

Based on the review of information in the SAR and its supporting documentation, the staff makes the following findings regarding the radiation protection program for the proposed ISFSI:

- The design and operating procedures of the HI-STORE CIS Facility provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 for meeting the objective of maintaining exposures ALARA, in compliance with 10 CFR 72.24(e).

- The SAR and other documentation submitted in support of the application are acceptable and provide reasonable assurance that the activities authorized by the license can be conducted without endangering public health and safety, in compliance with 10 CFR 72.40(a)(13).

- The proposed license technical specifications include those items necessary to ensure adequate radiation protection in the design, fabrication, construction, and operation of the DSF SSCs in accordance with the requirements in 10 CFR 72.44(c) and to meet the requirements in 10 CFR 72.44(d).

- The SAR provides analyses showing that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limits given in 10 CFR 72.104(a–c) and that releases from any design basis accident will not exceed the limits in 10 CFR 72.106(b).

- The design of the HI-STORE CIS Facility provides suitable shielding and confinement for radiation protection under normal and accident conditions, in compliance with 72.128(a)(2).

- The facility design and operations include adequate means for controlling personnel exposures and for controlling and monitoring effluents and direct radiation, in compliance with 10 CFR 72.126.

- The staff finds that the radiation protection program, as described by the applicant, satisfies the requirements of 10 CFR 20.1101(a) and (c), 20.1302(a), 20.1406, and 20.1501(a)(1).

11.5 References

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

Holtec International, "Licensing Report on the HI-STORE CIS Facility," Revision 0T, Docket No. 72-1051, January 20, 2023. Agencywide Documents Access and Management System Accession No. ML23025A112.

Holtec International, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Holtec Report No. HI-2115090, Revision 3, Docket 72-1040, June 29, 2016. ML16193A339.

Holtec International, "HI-STORE CIS Facility Site Boundary Dose Rates Calculations for HI-STORM UMAX System," Revision 4, March 22, 2022 (proprietary). ML22108A128.

NRC, RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3, June 1978. ML003739549.

NRC NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," March 2000. ML003686776.

NRC, RG 8.10, "Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable," Revision 2, August 2016. ML16105A136.

NRC, "Certificate of Compliance No. 1040, Appendix A, Technical Specifications for the HI-STORM UMAX Canister Storage System," Amendment No. 2, January 6, 2017. ML16341B100.

NRC, "Safety Evaluation Report, Docket No. 71-9373, Model No. HI-STAR 190 Package, Certificate of Compliance No. 9373, Revision No. 1," November 27, 2018. ML18332A029.

12 QUALITY ASSURANCE EVALUATION

In chapter 12, “Quality Assurance Program,” of the safety analysis report (SAR), Revision 0T, dated January 20, 2023, Holtec International (the applicant) described its program for activities at the proposed HI-STORE Consolidated Interim Storage (CIS) Facility affecting quality associated with the design, fabrication, construction, testing, operation, modification, and decommissioning of structures, systems, and components (SSCs) that are classified as important to safety (ITS).

12.1 Scope of Review

The staff reviewed the applicant’s quality assurance program (QAP) description to determine whether the applicant has a QAP that complies with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related Greater than Class C waste,” Subpart G, “Quality Assurance.” The staff also reviewed the “Holtec International Quality Assurance Manual,” Revision 15, dated June 10, 2022 (proprietary), which the applicant referenced in SAR chapter 12.

12.2 Regulatory Requirements

U.S. Nuclear Regulatory Commission (NRC) requirements relevant to the staff’s evaluation of a description of a QAP submitted as a part of the application for the proposed HI-STORE CIS Facility appear in 10 CFR 72.24(n) and Subpart G, “Quality Assurance,” of 10 CFR Part 72.

12.3 Staff Review and Analysis

In chapter 10, “Conduct of Operations Evaluation,” of the safety analysis report (SAR), Revision 0T, dated January 20, 2023, the applicant described its organizational structure, in which the Holtec Corporate Executive is responsible for the HI-STORE CIS Facility and has overall responsibility for safe operation of the site. The Site Manager is also responsible for safe operation of the site, including day-to-day implementation of the Holtec quality assurance manual and operation of all HI-STORE CIS Facility SSCs that are ITS. The quality assurance staff at the HI-STORE CIS Facility is also responsible for the implementation of the requirements of the Holtec quality assurance manual and report off-site to the Holtec Corporate Vice President of Quality.

In chapter 12 of the SAR, the applicant stated that the Holtec International QAP applies to the HI-STORE CIS Facility to control ITS activities related to construction and deployment of the HI-STORM UMAX system and other equipment at the HI-STORE CIS Facility. The Holtec International QAP includes the Holtec quality assurance manual that is approved by the NRC under Docket 71-0784 and associated implementing procedures and miscellaneous quality documents that are designed and administered to meet the applicable requirements of 10 CFR Part 71, Subpart H, “Quality Assurance,” and 10 CFR Part 72, Subpart G. As allowed by 10 CFR 72.140(d), the applicant stated that this approved 10 CFR Part 71 QAP will be applied to spent fuel storage cask activities at the HI-STORE CIS Facility under 10 CFR Part 72.

The staff confirmed that the Holtec International QAP includes the Holtec quality assurance manual and associated implementing procedures and miscellaneous quality documents that are designed and administered to meet the applicable requirements of 10 CFR Part 71, Subpart H, 10 CFR Part 72, Subpart G, and 10 CFR 72.24(n), including the requirement that the QAP applies to the managerial and administrative controls used to ensure safe operation of the ISFSI.

The staff noted that the Holtec quality assurance manual that applies to the HI-STORE CIS Facility was developed as a means to describe the quality assurance requirements that apply to activities affecting quality associated with the design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and use of the HI-STORM UMAX system that will be used at the HI-STORE CIS Facility, and that it applies to other HI-STORE CIS Facility equipment SSCs that are classified as ITS and subject to the requirements of 10 CFR Part 71 and 10 CFR Part 72. The staff also noted that the HI-STORE CIS Facility ITS SSCs are identified in chapter 4 of the SAR.

The staff confirmed that the complete description of the Holtec International QAP is contained in the Holtec quality assurance manual previously approved and inspected by the NRC during activities related to 10 CFR Part 71, Subpart H, and 10 CFR Part 72, Subpart G, under Docket No. 71-0784.

Based on its review, the staff finds that the applicant's QAP is acceptable because the applicant has adopted a currently approved QAP to meet 10 CFR 72.24(n) and 10 CFR 72.140, "Quality assurance requirements," and ensured that its organization with responsibilities related to quality assurance specific to the HI-STORE CIS Facility has been appropriately described in chapter 10 of the SAR.

12.4 Evaluation Findings

Based upon its review and evaluation of the QAP description contained in the application and the HI-STORE CIS Facility organization described in chapter 10 of the application, the staff concludes that the QAP description meets the requirements of 10 CFR 72.24(n) and 10 CFR Part 72, Subpart G. Specifically, the staff concludes the following:

- The QAP describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G.
- The QAP provides control over activities affecting quality to an extent commensurate with the importance to safety as defined in the SAR.
- The SAR in conjunction with the QAP establishes and delineates the authority and duties of persons and organizations performing activities affecting quality.
- The licensee's description of the QAP complies with applicable NRC regulations and industry standards, and the QAP can be implemented for the design, construction, operation, and decommissioning phases of the installation's life cycle.

12.5 References

Code of Federal Regulations, Title 10, *Energy*, Part 71, Packaging and Transportation of Radioactive Material."

Code of Federal Regulations, Title 10, *Energy*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

Holtec International, "Nuclear Quality Assurance Manual," Revision 15 (proprietary), Docket No. 71-0784, submitted June 10, 2022, Agencywide Documents Access and Management System Accession No. ML22161A072.

Holtec International, "Licensing Report on the HI-STORE CIS Facility," Revision 0T, Docket No. 72-1051, January 20, 2023. ML23025A112.

13 DECOMMISSIONING EVALUATION

Holtec International (Holtec or the applicant) described its plan for the decontamination and decommissioning of the proposed HI-STORE CIS Facility in southeastern New Mexico in chapter 13, “Decommissioning Evaluation,” of its safety analysis report (SAR), Revision 0T, dated January 20, 2023 and in attachment 6 of its application, Holtec Report No. HI-2177558, Revision 0, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Decommissioning Plan,” submitted on February 23, 2018.

13.1 Scope of Review

The staff reviewed the information in the application, including the decommissioning plan; SAR chapter 13; as well as SAR chapter 10, “Conduct of Operations Evaluation”; and SAR chapter 14, “Waste Confinement and Management Evaluation,” to determine whether the applicant described how it will address decontamination and decommissioning of the facility in appropriate detail to meet applicable regulatory requirements.

13.2 Regulatory Requirements

The following sections of Title 10 of the *Code of Federal Regulations* (10 CFR) contain the U.S. Nuclear Regulatory Commission (NRC) requirements relevant to the staff’s evaluation of the decommissioning plans of the applicant for the proposed HI-STORE CIS Facility:

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.30, “Financial assurance and recordkeeping for decommissioning”
- 10 CFR 72.130, “Criteria for decommissioning”

13.3 Staff Review and Evaluation

13.3.1 Cask System Design Features

SAR table 1.0.1, “Overview of the HI-STORE Facility,” states that the HI-STORE CIS Facility will use the HI-STORM UMAX Canister Storage System, Docket No. 72-1040. Decommissioning considerations for this cask system appear in section 2.11, “Decommissioning Considerations,” of Holtec Report No. HI-2115090, Revision 3, “Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System,” dated June 30, 2016 (HI-STORM UMAX FSAR), which the applicant incorporated by reference in SAR table 13.0.1, “Material Incorporated by Reference.” During certification of the HI-STORM UMAX System, the NRC staff found the canister system design and its decommissioning considerations to be acceptable.

The HI-STORM UMAX System is a dry, in-ground spent fuel storage system using vertical ventilated modules (VVMs). Each VVM will contain one canister. The VVM comprises a cavity enclosure container (CEC) and divider shell. Canisters can readily be removed from the VVM because the divider shell is not attached to the CEC. This allows for convenient removal or decommissioning, as removing the divider shell does not require welding or unfastening bolts.

As described in section 6, “Decommissioning Plan,” of the HI-STORE CIS Facility decommissioning plan, the canisters are loaded, sealed, and decontaminated at the generator’s facility before being stored at the HI-STORE CIS Facility. The sealed canisters are shipped to the HI-STORE CIS Facility, stored, and removed from the facility without the need to open the canister or handle fuel assemblies, allowing the facility to be operated as a clean facility.

Section 6.2, “Decommissioning Activities,” of the decommissioning plan states that the HI-STORE CIS Facility, including canister transfer equipment, storage cask, and storage pad, is not expected to have residual radioactive contamination because of the following:

- The canisters are sealed by welding.
- When the fuel is loaded at the originating reactors, measures are taken to prevent contamination of the canister’s outer surface.
- The canisters are not permitted to be transported to the facility unless surveys determine that surface contamination levels are below specified limits.
- The canisters will not be opened during transport, handling, or storage at the facility.
- Neutron activation of the storage cask and pad materials will be insignificant because the neutron flux from the spent fuel will be sufficiently low.

The applicant described in SAR section 1.2.1, “HI-STORM UMAX System Overview,” how, as an in-ground system, the HI-STORM UMAX System is protected from projectiles and extreme environmental occurrences such as hurricanes, fires, and explosions. The spent fuel is stored in a dry, inert environment inside a sealed metal canister.

Based on the information described above, the staff finds that, in accordance with 10 CFR 72.130, the cask system design features satisfactorily facilitate decontamination, minimize the quantity of radioactive waste and contaminated equipment, and facilitate removal of radioactive wastes and contaminated material when the facility is decommissioned.

13.3.2 Facility Design and Operational Features

The initial decommissioning plan, SAR section 13.1, “Design Features,” and SAR section 13.2, “Operational Features,” identify design and operational features that facilitate potential decontamination and eventual decommissioning. SAR sections 13.1 and 13.2 also reference the HI-STORM UMAX FSAR for discussion of design and operational features for that storage system that would be applicable to HI-STORE CIS Facility decommissioning. Features discussed in the initial decommissioning plan include minimizing contamination, maintaining accurate records of spills or other unusual occurrences involving the spread of contamination, and maintaining accurate as-built drawings. Section 13.3.3 of this safety evaluation report (SER) discusses these features.

Based on the information described above and the information in 13.3.3, the staff finds that the facility design and operational features satisfactorily facilitate decontamination, minimize the quantity of radioactive waste and contaminated equipment, and facilitate removal of radioactive wastes and contaminated material at the time the facility is decommissioned consistent with the requirements in 10 CFR 72.130.

13.3.3 Decommissioning Plan

Review of the initial decommissioning plan included consideration of the overall adequacy and completeness of the plan, including proposed decontamination and decommissioning activities.

The requirements of 10 CFR 72.30 are as follows:

- 10 CFR 72.30(a) requires that each application under Part 72 include a proposed decommissioning plan that contains sufficient information on the proposed practices and procedures for the decontamination of the site and for disposal of residual radioactive materials after all spent fuel has been removed. This plan must also identify and discuss those design features of the independent spent fuel storage installation (ISFSI) that facilitate decontamination and decommissioning at the end of its useful life.

- 10 CFR 72.30(b) requires that the proposed decommissioning plan also include a decommissioning funding plan.

- 10 CFR 72.30(c) requires that the financial assurance for decommissioning be updated and resubmitted at license renewal and intervals not to exceed 3 years.

- 10 CFR 72.30(d) states that, if residual radioactivity is detected in soils or ground water that require remediation, a new or revised decommissioning funding plan must be submitted.

- 10 CFR 72.30(e) specifies requirements for the decommissioning financial assurance instrument.

- 10 CFR 72.30(f) requires maintenance of records of information important to decommissioning the facility.

  10 CFR 72.30(g) requires each licensee to use financial assurance funds only for decommissioning activities and report adjustments to the balance.

The scope of review for this chapter only addresses 10 CFR 72.30(a) and (f). SER chapter 18, which includes the staff review of the applicant’s decommissioning funding plan, addresses 10 CFR 72.30(b), (c), (e), and (g).

13.3.3.1 Proposed Decommissioning Plan, 10 CFR 72.30(a)

The objective of the review is to determine whether the applicant’s provisions for decommissioning the HI-STORE CIS Facility offer reasonable assurance that decontamination and decommissioning of the facility at the end of its useful life will adequately protect public health and safety.

The applicant submitted an initial decommissioning plan that describes the conceptual program for decontaminating and decommissioning the facility. The applicant stated that, “A final

Decommissioning Plan detailing the activities and procedures for decommissioning will be submitted to NRC after the spent nuclear fuel (i.e., canisters) is removed from the facility."

Section 6.0 of the initial decommissioning plan states the following:

> the objective of decommissioning activities at the HI-STORE CIS Facility is to verify that any potential radioactive contamination is below established release limits, and in the unlikely event of contamination, to identify and remove radioactive contamination that is above the NRC release limits, so that the site may be released for unrestricted use and the NRC license terminated.

To meet this objective, the applicant stated, in section 6.0 of its initial decommissioning plan, that it intends to operate the HI-STORE CIS Facility as a "clean" facility. All components of the facility, including transport casks and storage canisters, are designed to minimize the potential for any contamination. Continual radiological surveys will be performed throughout the life of the facility to identify any possible contamination and to verify that the facility remains clean. The actual decommissioning activities presented in the final decommissioning plan will depend on the operating history of the facility and the results of the initial characterization survey performed at the beginning of the decommissioning period. The applicant stated that residual radioactive contamination is not anticipated at the HI-STORE CIS Facility, as the canisters will be welded shut and sealed at the generator's facility to prevent leaks. Before leaving the generator's facility and shipment to the HI-STORE CIS Facility, the canisters will be surveyed to ensure that the outer surfaces are clean. The canisters will not be opened during transport to the HI-STORE CIS Facility, nor will they be opened during handling or storage at the HI-STORE CIS Facility. Radiological activation of the VVMs and concrete pad material is expected to be insignificant, with radiation levels below the applicable NRC criteria for unrestricted release.

The applicant stated that, after all the canisters stored at the facility have been shipped off site and the decommissioning period begins, the applicant will conduct a historical site assessment. This assessment will consist of a record review and personnel interviews to identify any incidents that may have caused contamination to any area of the site. Areas where contamination may have occurred will be assessed in more detail to determine the extent of any residual contamination. Detailed characterization surveys will be performed to verify that the VVMs and concrete pads have not been contaminated. Confirmation surveys will be conducted on all other areas of the site that have not been affected.

The applicant stated that the VVMs and concrete pads are not anticipated to be contaminated and will be left in place or removed as determined by the HI-STORE CIS Facility. NRC limits for unrestricted release using conventional decontamination techniques will be used to minimize the volume of waste. Any waste generated will be shipped to a licensed facility for disposal.

The applicant stated that, "A final Decommissioning Plan detailing the activities and procedures for decommissioning will be submitted to NRC after the spent nuclear fuel (i.e., canisters) is removed from the facility."

The staff reviewed the initial decommissioning plan and finds that it includes sufficient discussion of the applicant's proposed practices and procedures for minimizing contamination at the facility. The plan also includes sufficient discussion of the applicant's conceptual program for decommissioning the facility. The initial decommissioning plan sufficiently identifies and discusses the design and operational features of the ISFSI that facilitate its decontamination and decommissioning. Therefore, the staff finds that the initial decommissioning plan satisfies 10 CFR 72.30(a).

13.3.3.2 Records of Information Important to Decommissioning, 10 CFR 72.30(f)

The initial decommissioning plan includes a commitment by the applicant in section 7.0 to maintain the following records that are identified by 10 CFR 72.30(f) as important to, and required to be kept for, decommissioning:

- records of spills or other unusual occurrences involving the spread of contamination (required by 10 CFR 72.30(f)(1))
- as-built drawings and modifications of structures and equipment in restricted areas (required by 10 CFR 72.30(f)(2))
- a document, which is updated a minimum of every 2 years, listing all areas designated at any time as restricted areas and all areas outside of restricted areas involved in a spread of contamination (required by 10 CFR 72.30(f)(3))
- records of the cost estimate performed for the decommissioning funding plan (as required by 10 CFR 72.30(f)(4))

Based on the applicant's proposed practices and procedures for decommissioning recordkeeping, the staff finds that the initial decommissioning plan satisfies 10 CFR 72.30(f).

13.4 Evaluation Findings

The staff reviewed the SAR and decommissioning plan submitted by the applicant for the Holtec HI-STORE CIS Facility.

Based on a review of the information in the documents referenced above, the staff concludes the following:

- The applicant’s preliminary decommissioning plan provides reasonable assurance that decommissioning issues for the ISFSI facility have been adequately characterized so that the site will ultimately be available for unrestricted use for any private or public purpose and, therefore, meets the requirements in 10 CFR 72.30(a).
- The applicant’s preliminary decommissioning plan provides reasonable assurance the applicant will keep records of information important to the decommissioning of a facility in an identified location until the site is released for unrestricted use; therefore, the decommissioning plan meets the requirements of 10 CFR 72.30(f).

After the spent fuel (i.e., canisters) is removed from the facility, the applicant will submit a final decommissioning plan detailing activities and procedures for decommissioning the site for unrestricted use. The staff has determined that the initial decommissioning plan submitted by the applicant provides reasonable assurance that decommissioning issues for the Holtec HI-STORE CIS Facility have been adequately discussed. The staff, therefore, concludes that the proposed preliminary decommissioning plan complies with 10 CFR 72.30(a) and (f).

13.5 References

Code of Federal Regulations, Title 10, Part 20, “Standards for Protection Against Radiation.”

Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

Holtec International, Holtec Report No. HI-2115090, Revision 3, “Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System,” June 29, 2016. Agencywide Documents Access and Management System Accession No. ML16193A339.

Holtec International, Holtec Report No. HI-2177558, Revision 0, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Decommissioning Plan,” February 23, 2018. ML18058A606.

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Holtec Report No. HI-2167374, Revision 0T, Docket No. 72-1051, January 20, 2023. ML23025A112.

U.S. Nuclear Regulatory Commission (NRC), NUREG-1757, “Consolidated Decommissioning Guidance,” Volume 1, Revision 2, “Decommissioning Process for Materials Licensees,” September 2006. ML063000243.

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

In chapter 14, “Waste Confinement and Management,” of the safety analysis report (SAR), Revision 0T, dated January 20, 2023, Holtec International (the applicant) described its analysis of the waste management systems for the proposed HI-STORE Consolidated Interim Storage (CIS) Facility.

14.1 Scope of Review

The staff reviewed the applicant’s waste management analysis in SAR chapter 14 to determine whether the facility provides safe confinement and management of radioactive waste generated by site activities involving the handling and storage of spent nuclear fuel at the facility, and whether the generation of radioactive waste and release of the radioactive material to the environment meet regulatory requirements.

14.2 Regulatory Requirements

The U.S. Nuclear Regulatory Commission requirements relevant to the waste confinement and management systems evaluation of the proposed HI-STORE CIS Facility appear in the following parts and sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR Part 20, “Standards for Protection Against Radiation”
- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.40, “Issuance of license”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

14.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed and evaluated the waste management systems analysis provided by the applicant in Revision 0T of its SAR, documents cited in or attached to its application, and the applicant’s responses to the staff’s requests for supplemental and additional information. The staff conducted its review in accordance with “NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000.

14.3.1 Waste Sources and Waste Management Facilities

Chapter 14 of the SAR describes various sources of radioactive waste generated from the operation of the facility. As described in SAR Chapter 14, gaseous and liquid wastes will not be produced at the facility. Operations may produce solid or solidified wastes from unloading the multipurpose canisters (MPCs) from transportation casks, waste from routine contamination surveys, and waste from decontamination of transport casks and other associated equipment.

The applicant described the radioactive waste sources, which are limited to wastes produced during the operation of the facility. The applicant stated that a small quantity of low-level solid waste may be generated, consisting of anticontamination clothing, rags, and associated health physics materials used during the MPC unloading operations. The waste is packed in the low-level waste holding cell in the designated radiologically controlled area in the cask transfer building until the waste is characterized and transported offsite for disposal at a low--level disposal facility licensed in accordance with 10 CFR Part 61, and in compliance with other applicable Federal and State regulations.

Since the dry cask storage system to be implemented at the HI-STORE CIS Facility is a passive design requiring no active systems to ensure adequate decay heat removal and to ensure adequate confinement, a canister would not be accepted at the facility if surveys conducted upon receipt indicated surface contamination exceeding the acceptable limits. The canisters, which are sealed by welding, would also not be opened at the facility. As stated in SAR section 14.3, the welded canisters are designed such that they are not breached under normal, off-normal, or accident conditions of transfer, handling, and storage. Therefore, no release of radioactive materials is expected under these conditions.

Based on its review, the staff finds that the applicant adequately described the waste sources at the facility and that there are no routine effluents discharged to the environment due to the operation of the facility, including during normal and off-normal conditions. Since there are no liquid or gaseous effluents, the staff finds that the application meets the requirements of 10 CFR 72.122(b)(4) precluding the transport of radioactivity to an aquifer and the dose limits of 10 CFR 72.104(a) with respect to the release of effluents. Furthermore, the staff finds that the application meets the requirements of 10 CFR 72.128(a)(5) to limit the generation of radioactive waste because the canisters are surveyed for surface contamination before transport to the facility and upon arrival at the facility, the canisters are designed with a welded closure, and the canisters are not opened at the facility.

14.3.2 Combustion Products

In SAR section 14.1, the applicant stated that while an explosion within the protected area is unlikely, an explosion may result from the presence of combustible fluid in the vertical cask transporter (VCT) as described in SAR section 6.5.2. The HI-TRAC transfer cask protects a canister during an explosion during transfers with the VCT and HI-STORM UMAX storage system. The lid of the HI-STORM UMAX, the vertical ventilated module (VVM), and the independent spent fuel storage installation (ISFSI) pad provide further protection to the canister from explosions. The effect of fire would be minimal due to the quantity of combustible fluid used in the VCT and the nature of the material used in the HI-TRAC transfer cask, VVM, and canister as described in section 6.5.2 of the SAR.

14.3.3 Off-Gas Treatment and Ventilation

As described in SAR sections 6.5 and 9.2, the MPCs are designed to endure normal, off--normal, and accident conditions of storage with maximum decay heat loads without loss of confinement. Permanent area radiation and airborne radioactivity monitors are not needed at the ISFSI because MPCs have a welded design that makes leakage noncredible. As described in SAR section 14.2, there are no gaseous releases from the storage systems at the facility, as the facility will only handle sealed canisters.

Based on its review, the staff finds that under normal operations, no radioactive materials will be released to the environment as gaseous effluents; therefore, no special off-gas or ventilation systems are needed. Thus, the staff concludes that the application meets the requirements of 10 CFR 72.122(h)(3), 10 CFR 72.126(c)(1), 10 CFR 72.126(d), 10 CFR 72.128(a)(5), and 10 CFR 72.128(b). Since there are no gaseous effluents, the staff finds that the design and operation of the facility meet the requirements in 10 CFR 72.104(a), (b), and (c) for offsite doses from effluents.

14.3.4 Liquid Waste Treatment and Retention

In SAR section 6.3, the applicant discussed the facility’s liquid waste treatment and retention program. The applicant stated that no liquid radioactive wastes are generated by the receipt and transfer of canisters containing spent nuclear fuel or greater-than-Class-C waste at the HI-STORE CIS Facility.

In SAR section 14.3, the applicant discussed the facility’s liquid waste treatment and retention program. The applicant stated that no liquid radioactive wastes are generated by the receipt and transfer of canisters containing spent nuclear fuel. Liquid wastes will not be routinely generated at the facility under normal operations. ISFSI pads are designed to ensure drainage of rainwater or any liquid spills away from HI-STORM UMAX VVMs. The loaded canisters are inspected before transfer to the HI-TRAC transfer cask and finally to the HI-STORE CIS Facility.

If a leak or damage is detected, or if surface areas of a canister have higher surface contamination than allowed by regulatory requirements, the decontamination will be done at the originator site or other facility able to perform decontamination, preventing the generation of radioactive liquid waste or other liquid spills at the HI-STORE CIS Facility. Decontamination involves a washdown of the transport cask after unloading the canister. Thus, the staff finds that no liquid radioactive waste treatment and retention systems are needed at the ISFSI, and that the requirements of 10 CFR 72.128(b) are met for contaminated liquids. There are no liquid radioactive effluents that will be discharged to the environment under normal operations. Because there are no liquid effluents, the requirements of 10 CFR 72.122(b)(4) and 10 CFR 72.126(c)(1) and (d) are met with respect to possible release of radioactive material in liquid effluents. Since there are no liquid effluents, the staff finds that the design and operation of the ISFSI meet the requirements of 10 CFR 72.104(a), (b), and (c) with regard to doses from liquid effluents.

14.3.5 Solid Wastes

A small quantity of low-level solid waste may be generated during loading operations. Contamination surveys are done on transportation casks and canisters upon receiving them. Low-level waste may be generated during surface contamination surveys. The solid waste may include disposable anticontamination garments, paper, rags, tools, and other items that will be processed and transferred to offsite low--level waste facilities. No canister will be opened at any time, thereby preventing further generation of solid radioactive wastes. Since no liquid waste is generated at the ISFSI site, there is no solidified waste resulting from liquid waste.

The staff agrees that the provisions for handling solid wastes are appropriate and meet the requirements of 10 CFR 72.128(b). The method described would not be expected to produce radioactive effluents and therefore meets the requirements of 10 CFR 72.104(a), (b), and (c) with respect to doses from effluents.

14.3.6 Radiological Impact of Normal Operations

In SAR section 14.5, “Radiological Impact of Normal Operations Summary,” the applicant discussed the radiological impact of the facility’s activities during normal operations. The applicant summarized the radiological impact of normal operations and stated that under all normal and off-normal conditions of transfer, handling, and storage, the welded canisters will remain sealed, and no radioactive material will be released from inside the canister. Additionally, the applicant proposed practices and procedures to limit and control contamination at the facility to provide assurance that radiological impacts are minimized and that the principles of “as low as reasonably achievable” are maintained. The applicant stated that no release of radioactive material to the environment is expected during normal facility operations, and no liquid or gaseous effluents are anticipated from the facility. Thus, the applicant concluded that the radiological impacts to the environment from normal operations at the facility will be minimal.

Because the canisters would remain sealed, based on its review, the staff determined that the radiological impact of the facility under normal operations has been adequately and appropriately described, and that the radiological impacts from releases will be minimal and will not endanger public health and safety. Based on these considerations, the staff finds that the application meets the requirements of 10 CFR 72.40(a)(13)(i) and 10 CFR 20.1101, “Radiation protection programs,” with respect to potential releases of radioactive materials under normal operations; 10 CFR 20.1301, “Dose limits for individual members of the public”; and 10 CFR 20.1302, “Compliance with dose limits for individual members of the public,” with respect to doses to members of the public from potential releases of radioactive materials under normal operations. The staff reviews and evaluates the dose to members of the public in chapter 7, “Shielding Evaluation”; chapter 9, “Confinement Evaluation”; and chapter 11, “Radiation Protection Evaluation,” of this safety evaluation report.

14.4 Evaluation Findings

The staff finds the following regarding waste confinement and management at the HI-STORE CIS Facility, based on its review:

- The SAR adequately describes acceptable features of the ISFSI design and operating modes that reduce, to the extent practical, the radioactive waste volume generated by the installation, in compliance with 10 CFR 72.24(f) and 10 CFR 72.128(a)(5).

- The facility design meets the requirements of 10 CFR 20.2001, 10 CFR 20.2003, 10 CFR 72.104(b) and (c), 10 CFR 72.122(h)(3), 10 CFR 72.126(c)(1), and 10 CFR 72.126(d).

- The facility design satisfies the radioactive waste management and minimization requirements of 10 CFR 72.128(a).

- The facility design meets the dose limits to members of the public and satisfies the requirements of 10 CFR 20.1101, 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR 72.104(a).

- The design of the ISFSI provides an acceptable means to limit the release of radioactive materials in effluents during normal operation to levels as low as reasonably achievable and to control the release of radioactive materials under accident conditions in compliance with 10 CFR 72.126(d).

- The waste confinement and management activities described in the SAR support the conclusion that the activities authorized by the license can be conducted without endangering public health and safety in compliance with 10 CFR 72.40(a)(13).

- The SAR adequately describes acceptable equipment to be used to maintain control over radioactive materials in gaseous and liquid effluent produced during normal operations and expected operational occurrences in compliance with 10 CFR 72.24(l).

- The design of the facility includes radioactive waste treatment facilities that have the capability to pack site-generated, low-level wastes in a form suitable for storage on site while awaiting transfer to disposal sites, in compliance with 10 CFR 72.128(b).

14.5 References

Code of Federal Regulations, Title 10, Part 20, “Standards for Protection Against Radiation.”

Code of Federal Regulations, Title 10 Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste”

Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Holtec Report No. HI-2167374, Revision 0T, Docket No. 72-1051, January 20, 2023. Agencywide Documents Access and Management System Accession No. ML23025A112.

NRC, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” NUREG-1567, March 2000. ADAMS Accession No. ML003686776.

15 ACCIDENT ANALYSIS

In chapter 15, “Accident Analysis,” of the safety analysis report (SAR), Holtec International (the applicant) describes its engineering analyses to qualify the structures, systems, and components (SSCs) that are important to safety (ITS) to be constructed and operated at the HI-STORE Consolidated Interim Storage (CIS) Facility, including the storage and transportation systems that will transport, transfer, and store spent nuclear fuel (SNF) at the proposed site, for off-normal operating conditions and for a range of credible and hypothetical accident conditions. Consistent with the guidance in Regulatory Guide (RG) 3.48, “Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage),” Revision 1, issued August 1989, the applicant used the design events identified by the American National Standards Institute (ANSI) and the American Nuclear Society (ANS) in ANSI/ANS 57.9-1984, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type),” to form the basis for the accident analyses for the HI-STORE CIS Facility storage and transportation systems.

15.1 Scope of Review

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed and evaluated the accident analyses of the proposed facility discussed in chapter 15 of Revision 0T of the HI-STORE SAR, dated January 20, 2023; documents cited in or attached to the SAR; and the applicant’s responses to the staff’s requests for additional and supplemental information. In SAR section 15.0, “Introduction,” the applicant stated that the HI-STORE CIS Facility will use the NRC-certified HI-STORM UMAX canister storage system to store commercial SNF. The applicant incorporated by reference information regarding off-normal and accident events from the final safety analysis report (FSAR) for the HI-STORM UMAX canister storage system.

15.2 Regulatory Requirements

The regulatory requirements relevant to accident analyses for the proposed HI-STORE CIS Facility appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.90, “General considerations”
- 10 CFR 72.92, “Design basis external natural events”
- 10 CFR 72.94, “Design basis external man-induced events”
- 10 CFR 72.104, “Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS”
- 10 CFR 72.106, “Controlled area of an ISFSI or MRS”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.124, “Criteria for nuclear criticality safety”

- 10 CFR 72.126, “Criteria for radiological protection”
- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

These requirements ensure that an applicant identifies and evaluates the hazards for off-normal and accident or design-basis events involving SSCs that are ITS.

15.3 Staff Review and Analysis

The NRC staff evaluated the applicant’s analysis of off-normal events and postulated accident events for the HI-STORE CIS Facility by reviewing the information provided in chapter 15 of the HI-STORE SAR. The staff reviewed this information to establish whether the HI-STORE CIS Facility’s SSCs, including its storage system design, satisfy the applicable operational and safety requirements. In the SAR, the applicant stated that the HI-STORM UMAX FSAR identifies design-basis events that are classified as either normal, off-normal, or accidents.

15.3.1 Off-Normal Events

In SAR section 15.2, “Off-Normal Events,” the applicant considered and evaluated the following off-normal events for the HI-STORE CIS Facility:

- off-normal pressure
- off-normal environmental temperature
- leakage of one multipurpose canister (MPC) seal
- partial blockage of air inlet and outlet ducts
- hypothetical nonquiescent wind
- cask drop less than the design-allowable height
- off-normal events associated with pool facilities
- off-normal events associated with the cask transfer building (CTB)

In SAR table 15.0.1, “Material Incorporated by Reference in this Chapter,” the applicant incorporated by reference information regarding off-normal events from section 12.1, “Off-Normal Conditions,” of the HI-STORM UMAX FSAR.

15.3.1.1 Off-Normal Pressure

The applicant discussed the off-normal event of off-normal pressure in SAR section 15.2.1, “Off-Normal Pressure.”

The staff evaluates the off-normal pressure event in safety evaluation report (SER) sections 6.3.3 and 6.3.4.4. In these referenced SER sections, the staff determined that the off-normal pressure event does not affect the ability of the storage systems at the HI-STORE CIS Facility to perform their safety functions.

15.3.1.2 Off-Normal Environmental Temperature

The applicant discussed the off-normal event of off-normal environmental temperature in SAR section 15.2.2, “Off-Normal Environmental Temperature.”

The staff evaluates the off-normal event of off-normal environmental temperature in SER sections 6.3.3 and 6.3.4.4. In these referenced SER sections, the staff determined that the off-normal environmental temperature event does not affect the ability of the storage systems at the HI-STORE CIS Facility to perform their safety functions.

15.3.1.3 Leakage of One Multipurpose Canister Seal

The applicant discussed the off-normal event of leakage of one MPC seal in SAR section 15.2.3, “Leakage of One MPC Seal.” The applicant incorporated by reference section 12.1.3, “Leakage of One MPC Seal Weld,” of the HI-STORM UMAX FSAR, Revision 3, which references section 12.1.3, “Leakage of One Seal,” of the HI-STORM FW FSAR, which the staff previously reviewed and found acceptable. Section 12.1.3 notes that the MPC-37 and MPC-89 are designed without seals and concludes that this off-normal event is not credible.

15.3.1.4 Partial Blockage of Air Inlet and Outlet Ducts

The applicant discussed the off-normal event of partial blockage of air inlet and outlet ducts in SAR section 15.2.4, “Partial Blockage of Air Inlets and Outlets.”

The staff evaluates the partial blockage of air inlet and outlet ducts off-normal event in SER section 6.3.4.1. In this referenced SER section, the staff determined that the event of partial blockage of air inlet and outlet ducts of the HI-STORM UMAX system does not affect the ability of the dry cask storage system to perform its safety functions.

15.3.1.5 Hypothetical Nonquiescent Wind

The applicant discussed the hypothetical nonquiescent wind off-normal event in SAR section 15.2.5, “Hypothetical Non-Quiescent Wind.”

The staff evaluates the hypothetical nonquiescent wind off-normal event in SER section 6.3.3. In this referenced SER section, the staff determined that the analysis of the hypothetical nonquiescent wind off-normal event is adequately considered and evaluated in the HI-STORE SAR.

15.3.1.6 Cask Drop Less than the Design-Allowable Height

The applicant discussed the off-normal event of a cask drop less than the design-allowable height in SAR section 15.2.6, “Cask Drop Less Than Design Allowable Height.” In this section, the applicant stated that cask drops are not credible because all lifting operations at the HI-STORE CIS Facility are conducted with redundant drop protection.

As discussed in SER section 5.3 and as required in section 4.2.8 of the technical specifications, the staff confirmed that all heavy load lifts at the HI-STORE CIS Facility will be made with equipment with redundant drop protection that meets the criteria of NUREG-0612, “Control of

Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A 36," issued July 1980. Accordingly, the staff has determined that cask drops less that the design-allowable heights at the facility are not credible off-normal events.

15.3.1.7 Off-Normal Events Associated with Pool Facilities

The applicant discussed the off-normal event associated with pool facilities in SAR section 15.2.7, "Off-Normal Events Associated with Pool Facilities." The applicant confirmed that there are no pool facilities at the HI-STORE CIS Facility, and therefore the staff agrees that this is not a credible off-normal event.

15.3.1.8 Off-Normal Events Associated with the Cask Transfer Building

The applicant discussed the off-normal events associated with the CTB in SAR section 15.2.8, "Off-Normal Events Associated with Cask Transfer Building." The applicant stated that there are no credible off-normal events specific to operations at the CTB, beyond those already considered and evaluated in the SAR. Off-normal conditions result from variations in operational conditions not expected to occur frequently but have the potential for infrequent occurrence. These variations may lead to temporary exceedances in operating pressure or temperature. In the structural evaluation of the CTB in SAR section 5.3.2.4, "Structural Analysis," no such situations were identified to exist by the applicant. The staff did not find any instances in which the CTB would be exposed to off-normal conditions. The CTB was demonstrated to satisfy a no-collapse criteria under the site-specific natural phenomena hazards, which prevents burial of any cask under a debris field. All operational conditions inside the CTB that have the potential for reducing the airflow to a cask, such as those during the stack-up at the CTF, have been evaluated. Based on this review, the staff agrees that the applicant's conclusion that there are no credible off-normal events at the CTB is acceptable.

15.3.1.9 Off-Normal Events Summary

The staff found the applicant's evaluation of off-normal conditions acceptable because the applicant relied on analyses previously accepted by the NRC for the HI-STORM UMAX system, which will be used at this facility, and the HI-STORE CIS Facility site conditions are bounded by the conditions specified in the evaluation of off-normal events in the HI-STORM UMAX FSAR.

For the reasons stated above, the applicant's identification and assessment of off-normal events meet the requirements of 10 CFR 72.122 to protect public health and safety, 10 CFR 72.124 to maintain SNF in a subcritical condition, 10 CFR 72.126 for radiological protection, and 10 CFR 72.128 for SNF waste handling, storage, and retrievability.

15.3.2 Accidents

The applicant stated in SAR section 15.3, "Accidents," that its accident analysis for the HI-STORE CIS Facility considered accident events evaluated for the HI-STORM UMAX system. The analysis considered the following events:

- fire accident

- partial blockage of MPC basket vent holes
- tornado missiles
- flood
- earthquake
- 100 percent fuel rods rupture
- confinement boundary leakage
- explosion
- lightning
- 100 percent blockage of air inlet and outlet ducts
- burial under debris
- extreme environmental temperature
- cask tipover
- cask drop
- loss of shielding
- adiabatic heatup
- accidents at nearby sites
- aircraft crash accidents
- accidents associated with pool facilities
- building structural failure onto SSCs
- 100 percent rod rupture accident coincident with accident events

The applicant provided details of the accident analyses in SAR sections 15.3.1 through 15.3.20.

15.3.2.1 Fire Accident

The applicant discussed the fire accident in SAR section 15.3.1, “Fire Accident.”

The staff evaluates the fire accident in SER section 6.3.4.4. The staff’s evaluation in SER section 6.3.4.4 determined that the applicant adequately considered and evaluated all credible fire accident scenarios that may occur during operations at the HI-STORE CIS Facility. The staff concluded that the MPC and its contents remain below the temperature and pressure allowable values during all the postulated fire accident scenarios.

15.3.2.2 Partial Blockage of Multipurpose Canister Basket Vent Holes

The applicant discussed the partial blockage of MPC basket vent holes accident in SAR section 15.3.2, “Partial Blockage of MPC Basket Vent Holes.”

The staff evaluates the partial blockage of MPC basket vent holes accident in SER section 6.3.4.4. The staff’s evaluation in SER section 6.3.4.4 determined that the applicant incorporated by reference the evaluation of partial blockage from the HI-STORM UMAX and HI-STORM FW FSARs for the HI-STORE CIS Facility. In both FSARs, the applicant determined that the partial blockage condition is noncredible. The staff determined that the incorporation of the HI-STORM UMAX and HI-STORM FW evaluations and the conclusion that this scenario is not credible are acceptable.

15.3.2.3 Tornado Missiles

The applicant discussed the tornado missile accident in SAR section 15.3.3, “Tornado Missiles.”

The staff evaluates tornado missile impacts for the HI-STORE CIS Facility in SER sections 5.3.1.1, 5.3.2.1, 5.3.2.2, 5.3.3.1, 5.3.3.3, 5.3.4.4.2, 5.3.4.4.5, 5.3.5.1, and 5.3.5.2.

In SER chapter 5, the staff evaluates the effect of tornado missile impact on the different ITS SSCs used in the conduct of operations at the HI-STORE CIS Facility. These include strikes on empty casks, on casks loaded with MPCs, and during cask transfer operations. The staff considers the impact of individual missiles in the spectrum in the above-referenced sections.

The analyses of tornado missile impacts conducted by the applicant evaluated the effects of a tumbling automobile on a loaded transfer cask by modeling the impact energy absorption and the work done in displacing the cask without tipover of the cask. However, there are two instances during transfer operations when this modeling assumption is not applicable because the transfer cask is in a bolted position: (1) during the lifting of the MPC from the HI-STAR 190 transportation cask to the HI-TRAC CS transfer cask (inside the CTB) and (2) during the lowering of the MPC from the HI-TRAC CS transfer cask into the HI-STORM UMAX vertical ventilated module (VVM). For these two stack-up conditions, handling procedures may have short-term operational steps or sequences that are not evaluated or modeled for tornado missile impacts in the SAR. The applicant has chosen to implement administrative controls to preclude or address the situation where short-term operations could be conducted and the potential for severe weather exists. These administrative controls, which are described in SAR section 10.3.3.5, “Placement of Canisters in the CEC,” consist of operational requirements precluding the initiation of handling operations or completing ongoing transfer operations in the event of a severe weather warning. The applicant should use a credible weather source for the severe weather forecast and periodically check the weather forecast during short-term operations to verify there is no change.

The CTB is designed for tornado wind loads but not for tornado missile impacts. The staff finds that this is acceptable because the ITS SSCs used for canister transfer operations inside the CTB are evaluated for tornado missiles, except for the overhead crane. The staff assessed that only an automobile strike leading to a structural collapse of the steel framing of the CTB has the potential to affect crane operations. However, as discussed in SER sections 6.3.4.4, 5.3.4.4, and 15.3.2.11, Holtec Report No. HI-2210576, “Structural Analysis of the HI-STORE Cask Transfer Building (proprietary),” Revision 1, dated March 24, 2022, analyzed the response of the CTB to automobile strikes and found that the collapse or damage of walls between load-bearing columns would not result in CTB frame collapse.

The staff finds that the analysis of tornado missile impacts at the HI-STORE CIS Facility adequately considered all operational scenarios and conditions and determined that the MPC is protected from tornado missile impacts by the transportation cask, the transfer cask, and the UMAX independent spent fuel storage installation (ISFSI) pad during all these operational sequences. In addition, the staff has determined that the applicant’s proposed approach to implement administrative controls and operational limits during severe weather scenarios provides assurance that all short-term canister transfer operations at the HI-STORE CIS Facility will be adequately protected from tornado missile impacts.

15.3.2.4 Flood

The applicant discussed the flood accident in SAR section 15.3.4, “Flood.”

The staff evaluates flood accidents in SER section 2.3.4.2.

As discussed in SER section 2.3.4, the staff determined that the applicant adequately described and modeled the probable maximum flood levels resulting from precipitation events at the HI-STORE CIS Facility site. The staff determined that the applicant adequately considered and analyzed the effects of the probable maximum flood scenario on ITS structures and buildings at the site, specifically the CTB and the UMAX ISFSI pads. The staff confirmed that the UMAX ISFSI pads are designed to withstand flood levels higher that those that may occur at the HI-STORE site under the probable maximum flood scenario, and that the expected flood levels at the CTB location would not result in water infiltration into the building or otherwise affect its structural integrity. As discussed in SER section 2.3.4.2.4, “Hydraulic Modeling of Flood Water Depth and Velocity,” the applicant’s model for hydraulic flow of flood water relies on the presence of a continuous barrier as an obstruction to rainwater runoff flow entering the facility during design basis flood events. Accordingly, in SAR section 1.1, “General Description of Installation,” the applicant specifies that the design of the VBS must be constructed in a manner that consists of continuous runs with a minimum height of 3 feet. The staff determined that the proposed VBS design specification in SAR section 1.1 will provide the required protection to onsite structures from design basis flooding events.

The staff concludes that the probable maximum flood scenario will not affect the ability of ITS buildings and structures at the HI-STORE CIS Facility to meet their required safety functions.

15.3.2.5 Earthquake

The applicant discussed the earthquake accident in SAR section 15.3.5, “Earthquake.”

The staff evaluates the earthquake loads on the different SSCs used at the HI-STORE CIS Facility in SER chapter 5, presenting the details in the specific evaluation of individual ITS and non-ITS SSCs to be used at the HI-STORE CIS Facility. In its evaluation, the staff determined that all ITS and non-ITS SSCs at the HI-STORE site have been adequately designed and evaluated to perform their intended safety functions under the earthquake accident scenario. SER section 2.3.6 discusses the staff’s evaluation of the estimated ground motion for the earthquake event at the HI-STORE site. Accordingly, the staff concludes that the applicant adequately considered the effects of earthquakes on the design and operation of the HI-STORE CIS Facility and demonstrated that the facility will be adequately protected from earthquakes.

15.3.2.6 100 Percent Fuel Rods Rupture

The applicant discussed the 100 percent fuel rods rupture accident in SAR section 15.3.6, “100% Fuel Rods Rupture.”

The staff evaluates the 100 percent fuel rods rupture accident in SER section 6.3.3. In its evaluation, the staff confirmed that the increased pressure inside the MPC from the scenario of 100 percent rupture of fuel rods will not affect the integrity of the MPCs' confinement barrier.

15.3.2.7 Confinement Boundary Leakage

The applicant discussed the confinement boundary leakage accident in SAR section 15.3.7, "Confinement Boundary Leakage." The applicant incorporated by reference the evaluation of the confinement boundary leakage scenario from the FSAR for the HI-STORM UMAX canister storage system, which states that confinement boundary leakage is noncredible.

The staff evaluates radionuclide confinement under accident scenarios in SER section 9.3.2. In its evaluation, the staff determined that there are no credible accident scenarios or sequences during operations at the HI-STORE CIS Facility that can result in a release of radionuclides from the MPCs. Therefore, the staff concludes that the confinement boundary leakage scenario is not credible.

15.3.2.8 Explosion

The applicant discussed the potential explosion hazards at the proposed facility in SAR section 15.3.8, "Explosion," and SAR section 6.5.2.2, "Explosion Event." Additionally, SAR section 2.2.4, "Ground Transportation," covers the potential explosion hazards from ground transportation. In addition, the applicant incorporated by reference the explosion event analysis from the HI-STORM UMAX FSAR. The HI-STORM UMAX FSAR evaluated the effects of an explosion event on the UMAX system.

The staff has reviewed the information on explosion hazards at the proposed facility to ensure that ITS SSCs at the proposed facility are designed and located appropriately so that they can continue to perform their safety functions effectively under credible explosion-related scenarios. This review considered the safety of both transfer and interim storage conditions from offsite or onsite explosions that may damage ITS SSCs at the proposed facility.

The staff has determined that the evaluation presented in the HI-STORM UMAX FSAR is applicable because the proposed facility does not permit any explosives or combustible materials within the HI-STORE site boundary. In addition, no combustible materials are needed during loading and unloading operations at the proposed facility. The only exception is the combustible material associated with the cask transporters, which was described in SAR table 6.5.1, "Vertical Cask Transporter Combustible Quantities and Fire Duration," and SAR table 6.5.3, "HI-PORT Combustible Quantities and Fire Duration." As the fuel available in the transporter is limited, the effects of an explosion on a safety-related structure are minimal. In general, the geometry of the proposed cask storage area makes it intrinsically resistant to explosion overpressure. The canisters will be placed below grade and enclosed in the cavity of the VVM of the HI-STORM UMAX cask storage system at the storage pads, with only the closure lids exposed. In this geometry, the area exposed to the progressing blast (overpressure) wave from explosions would be significantly smaller than an above ground configuration and, consequently, the impact would be significantly less. This geometry reduces the consequences of explosion-related accidents, including those from explosion-generated missiles, for the proposed cask storage area. Additionally, the canister is composed of nonexplosive materials and maintains an inert gas environment. Therefore, an explosion event during long-term storage is also not credible. Table 2.3.1, “Loads, Criteria, Applicable Regulators, Reference Codes, and Standards for the VVM,” of the HI-STORM UMAX FSAR specifies an explosion overpressure of 67 kilopascals (kPa) (10 pounds per square inch (psi)) in terms of steady-state differential pressure for the design of the storage cask system. Analyses presented in the HI-STORM UMAX FSAR show that the overpressure does not cause lid separation of the UMAX cask system, and all lid stresses are a fraction of the allowable limits. Similarly, the canister transfer operations at the CTB would also be partially conducted below grade in the canister transfer facility pit. Consequently, the area of the CTB exposed to a progressing air overpressure wave from an explosion would be smaller than an above ground configuration, thereby reducing the consequences of explosion-related accidents. Therefore, the staff concludes that any potential onsite explosion will not pose a credible hazard to safe operations at the proposed facility.

The staff also considered explosions that can take place outside the boundaries of the proposed facility. For example, a natural gas pipeline can rupture, and the released natural gas may ignite and explode. A highway truck full of trinitrotoluene (TNT) can explode while transiting the nearby U.S. Highway 62/180. Similarly, a railway boxcar carrying TNT explosive can explode near the proposed HI-STORE CIS Facility. The evaluation of offsite accidents resulting in explosions mainly involved considered transient hazardous materials near the site. The potential scenarios near the proposed facility that can result in an offsite explosion include (1) nearby natural gas pipelines and (2) explosives and other flammable materials transported on U.S. Highway 62/180 and the nearby railroad tracks. The staff reviews and evaluates these potential scenarios in SER section 2.3.2.2 and presents only a brief discussion on each scenario here.

The applicant analyzed the potential scenarios and associated consequences from an accidental rupture of a pipeline in Holtec Report No. HI-2210487, “HI-STORE Gas Pipeline Risk Evaluation” (proprietary), Revision 2, dated August 12, 2022. A rupture of a natural gas pipeline would release the pressurized natural gas in an unconfined space. If ignited, the released natural gas could result in a jet fire, a flash fire, or, in an extremely unlikely scenario, a detonation within the jet, depending on the strength of the ignition source. As discussed in SER section 2.3.2.2, damage to the safety-related SSCs in the proposed ISFSI from an accidental rupture of a pipeline is extremely unlikely. Additionally, the applicant estimated the frequency of accidents involving large trucks in Holtec Report No. HI-2210620, “HI-STORE Highway 62/180 Hazardous Chemicals Risk Evaluation” (proprietary), Revision 2, dated August 12, 2022, and railway boxcars in Holtec Report No. HI-2210619, “HI-STORE Railway Hazardous Chemicals Risk Evaluation” (proprietary), Revision 2, dated March 17, 2022, releasing hazardous chemicals carried onboard near the proposed site and found the frequency to be extremely small, less than 1×10^{-6} releases per year. TNT is not a cargo regularly transported in large trucks on U.S. Highway 62/180 or in railcars on tracks near the proposed site. Using the maximum quantity of TNT that can be transported on any road and in any rail boxcar, as specified in RG 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants,” Revision 2, issued April 2013, the safe distance at which the overpressure resulting from detonation of the entire TNT load in a highway truck or a railway boxcar would be equal to 6.9 kPa (1 psi), calculated using equation (1) of RG 1.91, is smaller than the shortest distances between the proposed facility and U.S. Highway 62/180 and nearby railway lines. Therefore, the staff concludes that a potential offsite explosion is not a credible hazard to the proposed facility.

Based on its review, the staff finds that the applicant's analyses for explosion are acceptable because the analyses incorporated by reference from the HI-STORM UMAX FSAR bound the design criteria of the site. In addition, the analyses provided by the applicant for explosion hazards from nearby gas pipelines and from rail and highway transportation show that any offsite explosions are not a credible hazard to the facility and will not damage ITS SSCs.

15.3.2.9 Lightning

The applicant discussed the lightning accident in SAR section 15.3.9, "Lightning." The applicant incorporated by reference the evaluation of lightning strikes scenario from the HI-STORM UMAX FSAR, which states that lightning strikes do not affect the ability of the HI-STORM UMAX system to safely store spent fuel canisters in its enclosures. In its evaluation, the applicant stated that because the UMAX ISFSI pads sit in contact with the soil, any lightning strike would dissipate immediately through the surrounding steel of the lid and enclosure to the surrounding substrate.

The staff finds that a lightning strike accident will not affect the ability of ITS SSCs at the HI-STORE CIS Facility to perform their intended safety functions. As referenced by the applicant in the HI-STORM UMAX FSAR, a lightning strike onto the lid of the UMAX ISFSI will readily dissipate without affecting any of the materials of construction of the pads. In addition, in SAR section 15.3.8, "Explosion," SAR section 6.5.2.2, "Explosion Event," and SAR section 15.3.1, "Fire Accident," the applicant evaluated the consequences of potential fires and explosion accident scenarios that may be triggered by lightning strikes on other areas of the facility. As discussed in SER section 15.3.2.8, "Explosion," and section 15.3.2.1, "Fire Accident," the staff confirmed that a fire from combustible fuel or hydraulic fluids from the HI-PORT transporter vehicle or the vertical cask transporter would not affect or damage the confinement barrier of the MPCs. Therefore, the staff concludes that a lightning strike would not affect the ability of ITS SSCs to perform their intended safety functions at the HI-STORE CIS Facility.

15.3.2.10 100 Percent Blockage of Air Inlet and Outlet Ducts

The applicant discussed the 100 percent blockage of air inlet and outlet ducts accident in SAR section 15.3.10, "100% Blockage of Air Inlets and Outlets."

The staff evaluates the 100 percent blockage of air inlet and outlet ducts accident in SER sections 6.3.3 and 6.3.4.4. In these referenced SER sections, the staff determined that the applicant properly considered and modeled the scenario of complete blockage of air inlet and outlet ducts of the HI-STORM UMAX system. In its evaluation, the staff confirmed that the temperature and pressure increase in the MPC components from blockage would not exceed the allowable material limits within the analyzed transient period and, therefore, will not affect the ability of the storage systems at the HI-STORE CIS Facility to perform their safety functions.

15.3.2.11 Burial Under Debris

The applicant discussed the burial under debris accident in SAR section 15.3.11, "Burial Under Debris."

SER section 6.3.4.4 contains the staff's thermal evaluation of the burial under debris accident. The staff confirmed that the scenario of burial under debris resulting from the collapse of a building is not credible because no structures will be built above the UMAX ISFSI pads. In addition, in its evaluation of accident scenarios for the CTB in SER section 5.3.4.4, the staff confirmed that the CTB's design can withstand severe wind and seismic ground motion loads, and that collapse of the CTB is not credible. Therefore, the burial under debris accident scenario is not credible.

15.3.2.12 Extreme Environmental Temperature

The applicant discussed the extreme environmental temperature accident in SAR section 15.3.12, "Extreme Environmental Temperature." The applicant stated that the extreme environmental temperature accident scenario at the HI-STORE CIS Facility site is bounded by the generic analysis provided in the HI-STORM UMAX FSAR because it considers an extreme environmental temperature that is higher than that those expected at the HI-STORE site.

The staff evaluates the extreme environmental temperature accident in SER sections 6.3.3 and 6.3.4.4. In these referenced SER sections, the staff determined that the extreme environmental temperature scenario at the HI-STORE site is bounded by the generic UMAX FSAR analysis, and that the applicant adequately referenced the evaluation for the scenario of extreme environmental temperature from the HI-STORM UMAX FSAR. In its evaluation, the staff confirmed that the temperature and pressure increase in the MPC and UMAX components from extreme temperature would not exceed the allowable material limits and, therefore, not affect the ability of the storage systems at the HI-STORE CIS Facility to perform their safety functions.

15.3.2.13 Cask Tipover

The applicant discussed the cask tipover accident in SAR section 15.3.13, "Tip-over." The applicant stated that tipover is not a credible accident scenario at the HI-STORE CIS Facility.

The applicant analyzed the potential for cask tipover as a result of a tornado-generated automobile missile impact in SAR chapter 5, and SER section 15.3.2.3 contains the staff's evaluation.

For nonmechanistic tipover of an MPC in storage in a HI-STORM UMAX VVM, the staff confirmed in its SER for the initial certification of the HI-STORM UMAX system that tipover within the UMAX VVM was not credible and a nonmechanistic tipover analysis was not necessary.

In addition, the design controls for lifting equipment in section 4 of the technical specifications preclude the potential for tipover during cask transportation and handling operations.

15.3.2.14 Cask Drop

The applicant discussed the cask drop accident in SAR section 15.3.14, “Cask Drop.” The applicant stated that a cask drop is not credible because all heavy load lifts of ITS SSCs at the HI-STORE CIS Facility are made with redundant drop protection equipment.

As discussed in SER chapter 5 and as required in section 4.2.8 of the technical specifications, the staff confirmed that all heavy load lifts at the HI-STORE CIS Facility will be made with equipment with redundant drop protection that meets the criteria of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980. Accordingly, a cask drop accident at the HI-STORE site is not credible.

15.3.2.15 Loss of Shielding

The applicant discussed the loss of shielding accident in SAR section 15.3.15, “Loss of Shielding.”

The staff evaluates loss of shielding accidents in SER section 7.3.9. In its evaluation, the staff considered and evaluated several accident conditions that have the potential to result in a loss of shielding. The applicant evaluated the potential loss of concrete, including losses of the crystalized water and spallation of concrete, of the HI-TRAC CS under a fire accident scenario. The applicant considered this potential accident scenario in the dose rate calculation for the HI-TRAC CS, provided in Holtec Report No. HI-2177553, Revision 3, “Thermal Analysis of HI-TRAC CS Transfer Cask.” Chapter 7 of this SER provides a detailed discussion on how the shielding analyses for this accident scenario are considered. The staff reviewed this scenario and determined that the loss of concrete and degradation of shielding materials from this fire would not affect the ability of the HI-TRAC CS transfer cask to meet the radiation protection requirements in 10 CFR 72.106.

15.3.2.16 Adiabatic Heatup

The applicant discussed the adiabatic heatup accident in SAR section 15.3.16, “Adiabatic Heatup” and stated the accident was not credible because it postulates there are no means of heat dissipation by conduction, convection, and radiation heat transfer.

Staff notes that the HI-STORM UMAX FSAR also did not list an adiabatic heatup accident scenario, although it did include a transient adiabatic calculation to represent burial under debris in section 4.6.2.4. Likewise, HI-STORE CIS Facility SAR section 6.5.2.5 included a similar transient adiabatic analysis to model a situation where heat removal from the canister was prevented. Staff finds that these analyses provide a means to model adiabatic conditions and finds that adiabatic heatup is a bounding analysis because it conservatively assumes a scenario in which there are no means of heat transfer (including conduction, convection, and radiation) from a canister to surroundings (i.e., ambient, soil, debris).

15.3.2.17 Accidents at Nearby Sites

The applicant discussed the effects of accidents at nearby offsite facilities on operation of the proposed HI-STORE CIS Facility in SAR section 15.3.17, “Accidents at Nearby Sites.” It presented additional discussions and assessments in SAR section 2.1, “Geography and Demography,” and SAR section 2.2, “Nearby Industrial, Transportation, Military, and Nuclear facilities.” The staff reviewed this information to evaluate credible accidents at these nearby facilities that may endanger the radiological safety of the proposed facility.

Based on the information provided in the SAR, no military facilities are located within 8 kilometers (km) (5 miles (mi)) of the proposed facility. The industrial facilities within 8 km (5 mi) of the proposed facility include a land farm, Intrepid Potash Mines, a gas pipeline compressor station, and a caliche mine. As discussed in SAR section 2.2 and reviewed in SER section 2.3.2.1, due to the nature of the operations at the land farm and the caliche mine, the effects of an accident would be limited to the immediate vicinity and would not affect the proposed facility. A potential explosion of natural gas at the gas pipeline compressor station would not affect the proposed facility because the compressor station is located approximately 8.4 km (5.2 mi) away.

The Intrepid Potash North Mine is 6.8 km (4.2 mi) northwest and the Intrepid Potash East Mine is 7.9 km (4.9 mi) southwest of the proposed site. Underground mining has not occurred in the North Mine since 1982, which is currently used as a warehouse and distribution center. The proposed HI-STORE CIS Facility will be located entirely within Section 13, Township 20 South, Range 32 East, in the New Mexico Public Land Survey system. As discussed in SER section 2.3.2.1, it is extremely unlikely that the potash underneath Section 13 will be mined during the proposed licensed term of operation for the facility. As stated in SAR section 2.1.4, “Land and Water Use,” the potash is extracted using conventional mining machinery. Explosives are not used to blast the potash ore for mining. A collapse of a mine excavation would lead to a roof-fall accident. The excavation may grow upwards as the collapse progresses until an equilibrium is reached. The associated subsidence can reach the surface if the lateral extent of the collapse is large. As discussed in SER section 2.3.2.1, the proposed site is a significant distance away from existing underground potash mining areas and, consequently, will not be affected even if the subsidence reaches the surface. Based on the preceding discussion, the staff finds that an accident at nearby potash mining operations would not be a credible hazard to the proposed facility.

The SAR identifies nearby nuclear facilities, including the U.S. Department of Energy’s (DOE’s) Waste Isolation Pilot Plant (WIPP), the National Enrichment Facility (NEF) proposed fluorine extraction process and depleted uranium deconversion plant (which has not been constructed), and the Waste Control Specialists (WCS) CIS Facility. The WIPP facility, which is the underground repository for permanent disposal of transuranic and mixed waste generated through defense activities and programs, is located approximately 26 km (16 mi) southwest of the proposed site. The waste at the WIPP is disposed of in underground salt caverns at a depth of approximately 655 meters (2,150 feet). As of the end of 2014, approximately 90,983 cubic meters of transuranic waste has been disposed of at the WIPP. Although not the usual route to the WIPP, transuranic wastes have been transported on U.S. Highway 62/180 to the WIPP on occasion. U.S. Highway 62/180 is approximately 1.6 km (1 mi) from the proposed HI-STORE CIS Facility site. In SER section 2.3.2.1.2, the staff evaluated the hazards to the facility from transportation accidents involving hazardous materials along this highway. The staff verified that the estimated accident frequency adequately considered all relevant shipments of hazardous materials, including the small number of transuranic waste shipments to WIPP along this highway. Because the number of shipments of transuranic wastes is small and the shipments are made in NRC-certified transportation packages that meet the package performance criteria in 10 CFR Part 71, the staff concludes that the increase in accident frequency from these is negligible and will not pose a credible hazard to the HI-STORE CIS Facility.

The NEF is located approximately 61 km (38 mi) southeast of the proposed site near the New Mexico–Texas border. The NEF enriches uranium to a maximum of 5 percent uranium-235 for manufacturing nuclear fuel for commercial nuclear power reactors. The staff evaluated the safety of the NEF and documented its findings in NUREG-1827, “Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico,” issued June 2005. Appendix A, “Accident Analysis” of NUREG-1827 provides a discussion and evaluation of potential accidents that can occur at the NEF and the possible consequences of these on workers, the public, and the environment. The accidents considered include accidents initiated by natural phenomena, equipment failure, and operator error. In its evaluation of the accident analysis for the NEF, the staff concluded that possible accidents at the NEF will pose acceptably low risks to the public, workers, and the environment because of the NEF’s design features, its implementation of passive and active engineered controls, and implementation of administrative controls. The consequences of potential accidents are limited to risks to onsite workers or the immediate vicinity of the NEF site. Due to the substantial distance of the NEF site from the HI-STORE site, the staff concludes that any credible accidents at the NEF would not pose a credible hazard to the HI-STORE CIS Facility.

The WCS CIS Facility is located in Andrews County, Texas, approximately 63 km (39 mi) east of the HI-STORE site. The WCS CIS Facility is similar to the HI-STORE CIS Facility and will use several NRC-certified aboveground dry storage cask systems. The NRC documented potential environmental effects from the WCS CIS Facility in NUREG-2239, “Environmental Impact Statement for Interim Storage Partners LLC's License Application for a Consolidated Interim Storage Facility for Spent Nuclear Fuel in Andrews County, Texas: Final Report,” issued July 2021. As discussed in Chapter 16, “Accident Analyses” of the SER for the WCS CISF, the staff found that the ISP adequately considered and evaluated all credible off-normal and accident scenarios at the WCS CISF, and that none of those scenarios would affect the ability of its ITS SSCs to perform their safety functions. Due to the substantial distance of the WCS CISF site from the HI-STORE site, the staff concludes that any credible accidents at the WCS CISF site would not pose a credible hazard to the HI-STORE CIS Facility.

In summary, the staff finds that the applicant has adequately described and assessed the nearby industrial, transportation, and military facilities. Based on the preceding discussion, the staff concludes that an accident at a nearby site is not expected to affect the proposed facility.

15.3.2.18 Aircraft Crash Accidents

The applicant assessed the potential crash hazard from aircraft flying in the vicinity of the proposed site and documented the assessment in SAR section 2.2.3, “Air Transportation,” and in the calculation package in Holtec Report No. HI-2188201, “HI-STORE CIS Aircraft Crash Assessment” (proprietary), dated May 5, 2021.

The staff reviewed the information presented in the SAR, the calculation package, and the applicant’s May 24, 2018, response to the staff’s requests for additional information. The staff also reviewed pertinent information available in the literature. This staff review determined whether the hazards to the proposed HI-STORE CIS Facility from aircraft flying nearby have been appropriately characterized and considered in the design of the proposed facility. The staff reviewed the applicant's aircraft crash hazard assessment in accordance with section 3.5.1.6, "Aircraft Hazards," Revision 4, issued March 2010, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." The staff accepts the methodology in NUREG-0800 as applicable for reviewing the aircraft crash hazard to the proposed facility.

The applicant examined activities in connection with potential hazards from the crash of civilian and military aircraft flying in the vicinity of the proposed facility. The activities examined include aircraft taking off and landing at nearby airports; flying in nearby Federal airways, including missed approaches and holding patterns associated with nearby airports; and flying nearby military training routes while jettisoning ordnance.

SER section 2.3.2.3 discusses of the staff's evaluation of aircraft crash hazards at the HI-STORE CIS Facility. As described in that section, the staff concluded that the applicant adequately estimated the probability of an aircraft crash on the HI-STORE site and confirmed that the estimated probability is lower than the 1×10^{-6} threshold established by the NRC for hazards to spent fuel storage facilities. Accordingly, the staff finds that an aircraft crash accident is not credible.

15.3.2.19 Accidents Associated with Pool Facilities

This type of accident is not applicable to the HI-STORE CIS Facility because no pool facilities are present at the facility or are necessary to support facility operations.

15.3.2.20 Building Structural Failure onto Structures, Systems, and Components

The applicant discussed the fire accident in SAR section 15.3.19, "Building Structural Failure onto SSCs."

As discussed in SER sections 6.3.4.4, 5.3.4.4, and 15.3.2.11, the staff finds that building structural failure is not a credible accident scenario.

15.3.2.21 100 Percent Rod Rupture Accident Coincident with Accident Events

The applicant discussed this accident scenario in SAR section 15.3.20, "100% Rod Rupture Accident Coincident with Accident Events."

The staff evaluates the 100 percent rod rupture accident coincident with accident event scenario in SER section 6.3.4.4. The staff determined that the scenario was addressed in the HI-STORM UMAX FSAR and does not affect the ability of the storage systems at the HI-STORE CIS Facility to perform their safety functions.

15.3.2.22 Accidents Summary

For the reasons stated above, the applicant's identification and assessment of accidents meet the requirements of 10 CFR 72.122 to protect public health and safety, 10 CFR 72.124 to maintain SNF in a subcritical condition, 10 CFR 72.126 for radiological protection, and 10 CFR 72.128 for SNF and Greater than Class C waste handling, storage, and retrievability.

15.3.3 Other Nonspecified Accidents

The applicant did not identify additional accident scenarios applicable to the HI-STORE CIS Facility. Based on its review in other SER chapters, the staff finds that the design and operational characteristics of the HI-STORE CIS Facility will not result in any additional accident scenarios beyond those already identified by the applicant.

15.4 Evaluation Findings

The applicant identified and provided complete analyses of the credible off-normal and accident events for operations at the HI-STORE CIS Facility site. Based on the information in the application, the staff concludes the following:

- The SAR includes acceptable analyses of the design and performance of ITS SSCs under off-normal and accident scenarios. Applicable off-normal accidents analyzed in the SAR include off-normal pressure, off-normal environmental temperature, leakage of one MPC seal, partial blockage of air inlet and outlet ducts, hypothetical nonquiescent wind, and cask drop less than the design allowable height. Applicable accident events analyzed in the SAR included fire accident, partial blockage of MPC basket vent holes, tornado missiles, flood, earthquake, 100 percent fuel rods rupture, confinement boundary leakage, explosion, lightning, 100 percent blockage of air inlet and outlet ducts, burial under debris, extreme environmental temperature, cask tipover, cask drop, loss of shielding, adiabatic heatup, accidents at nearby sites, building structural failure onto SSCs, and 100 percent rod rupture accident coincident with accident events.

- The analyses of off-normal and accident events and conditions, and reasonable combinations of these and normal conditions, show that the design of the HI-STORE CIS Facility will acceptably meet the requirements without endangering public health and safety, in compliance with the overall requirements of 10 CFR 72.122.

- The analyses of off-normal and accident events and conditions, and reasonable combinations of these and normal conditions, show that the design of the HI-STORE CIS Facility will acceptably meet the requirements of 10 CFR 72.124 for the maintenance of the SNF in a subcritical condition.

- The analyses of off-normal and accident events and conditions, and reasonable combinations of these and normal conditions, show that the design of the HI-STORE CIS Facility will acceptably meet the requirements of 10 CFR 72.126 for criteria for radiological protection.

- The analyses of off-normal and accident events and conditions, and reasonable combinations of these and normal conditions, show that the design of the HI-STORE CIS Facility will acceptably meet the requirements of 10 CFR 72.128 for handling, storage, and retrievability of the SNF and other radioactive material.

15.5 References

American National Standards Institute and American Nuclear Society, ANSI/ANS 57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," 1984.

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

U.S. Department of Energy (DOE), DOE/EIS-0026-S2, "Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement," September 1997.

Holtec International, "Final Safety Analysis Report on the HI-STORM FW MPC Storage System," Revision 4, Holtec Report No. HI-2114830, Docket No. 72-1032, June 24, 2015. Agencywide Documents Access and Management System Accession No. ML15177A338.

Holtec International, Holtec Report No. HI-2115090, "Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System," Revision 3, June 29, 2016. ML16193A339.

Holtec International, Document ID 5025024, Attachment 1, "Safety Analysis Report (SAR), Chapter 2—'Site Characteristics,'" Responses to Requests for Additional Information, May 24, 2018. ML18150A330.

Holtec International, Holtec Report No. HI-2188201, "HI-STORE CIS Aircraft Crash Assessment" (proprietary), May 5, 2021. ML21131A201.

Holtec International, Holtec Report No. HI-2177553, "Thermal Analysis of HI-TRAC CS Transfer Cask," (proprietary), Revision 3, dated August 13, 2021. ML21228A214.

Holtec International, Holtec Report No. HI-2210619, "HI-STORE Railway Hazardous Chemicals Risk Evaluation" (proprietary), Revision 2, March 17, 2022. ML22108A123.

Holtec International, Holtec Report No. HI-2210576, "Structural Analysis of the HI-STORE Cask Transfer Building" (proprietary), Revision 1, March 24, 2022. ML22108A129.

Holtec International, Holtec Report No. HI-2210620, "HI-STORE Highway 62/180 Hazardous Chemicals Risk Evaluation" (proprietary), Revision 2, August 12, 2022. ML22227A164.

Holtec International, Holtec Report No. HI-2210487, "HI-STORE Gas Pipeline Risk Evaluation" (proprietary), Revision 2, August 12, 2022. ML22227A163.

Holtec International, Holtec Report No. HI-2210620, "HI-STORE Highway 62/180 Hazardous Chemicals Risk Evaluation" (proprietary), Revision 2, August 12, 2022. ML22227A164.

Holtec International, Holtec Report No. HI-2167374, "Licensing Report on the HI-STORE CIS Facility," Revision 0T, January 20, 2023. ML23025A112.

U.S. Nuclear Regulatory Commission (NRC), NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980. ML070250180.

NRC, NUREG-1827, “Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico,” June 2005. ML051780290.

NRC, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 3.5.1.6, “Aircraft Hazards,” Revision 4, March 2010. ML100331298.

NRC, Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants,” Revision 2, April 2013. ML12170A980.

NRC, “Safety Evaluation Report, Docket No. 72-1040, HI-STORM UMAX Canister Storage System, Holtec International, Inc., Certificate of Compliance No. 1040,” April 2, 2015. ML15093A510.

NRC, NUREG-1790, “Environmental Impact Statement for the Proposed Fluorine Extraction Process and Depleted Uranium Deconversion Plant in Lea County, New Mexico,” June 2005. ML15155B285, ML15155B289, ML15155B287, ML15155B290, ML15155B286, and ML051730292.

NRC, NUREG-2239, “Environmental Impact Statement for Interim Storage Partners LLC’s License Application for a Consolidated Interim Storage Facility for Spent Nuclear Fuel in Andrews County, Texas: Final Report,” July 2021. ML21209A955.

16 TECHNICAL SPECIFICATIONS

In Appendix A, “Proposed Technical Specifications,” of its proposed license, dated November 23, 2022, Holtec International (the applicant) provided the technical specifications (TS) that are applicable to the important-to-safety storage systems and other operational components at the proposed HI-STORE Consolidated Interim Storage (CIS) Facility. The proposed TS include functional and operating limits, monitoring instruments, limiting control settings, limiting conditions for operation (LCOs), surveillance requirements (SRs), design features, and administrative controls. The applicant identified the proposed TS necessary to maintain subcriticality, confinement, shielding, heat removal, and structural integrity under normal, off-normal, and accident conditions.

16.1 Scope of Review

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the proposed TS to ensure that those conditions and limits necessary for the design and operations of the HI-STORE CIS Facility meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.” The staff reviewed the proposed specifications to assess compliance with the regulatory requirements for the areas of functional and operating limits, LCOs and SRs, design features, and administrative controls.

16.2 Regulatory Requirements

The NRC requirements relevant to the overall function of an independent spent fuel storage installation and the design criteria and operation of certain separate functional subsystems at the proposed HI-STORE CIS Facility appear in the following 10 CFR sections:

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.26, “Contents of application: Technical specifications”
- 10 CFR 72.44, “License conditions”

16.3 Staff Review and Analysis

Unless otherwise stated, the staff reviewed and evaluated the TS proposed by the applicant in its proposed license, documents cited in or attached to its application, and the applicant’s responses to the staff’s requests for additional information.

16.3.1 Functional and Operating Limits

The staff reviewed the functional and operating limits specified by the applicant in sections 2.1 and 2.2 of the proposed HI-STORE CIS Facility TS. Safety evaluation report (SER) table 16-1 lists the proposed functional and operating limits TS and the associated SER sections that discuss them.

Table 16-1 HI-STORE CIS Facility TS Functional and Operating Limits and Associated SER Sections

TS Item	Functional and Operating Limit	Associated SER Sections
2.1	Approved Contents, Fuel Specifications and Loading Conditions	4.3.1, 6.3.3, 6.3.4
2.2	Violations	16.3.1

The staff confirmed that the functional and operating limits proposed by the applicant are consistent with corresponding limits in the certificate of compliance TS previously approved by the NRC. The staff also confirmed for TS 2.2, “Violations,” which contains requirements for correction, notification, and reporting in the event a canister is received that does not meet the fuel specifications or loading requirements established in Technical Specification 2.1, “Approved Contents, Fuel Specifications and Loading Conditions,” that the requirements proposed by the applicant are consistent with the corresponding requirements in the certificate of compliance TS previously approved by the NRC. The staff finds that the functional and operating limits listed in SER table 16-1 apply to the fuel proposed for storage at the HI-STORE CIS Facility and are necessary to protect the integrity of the stored fuel, to protect employees against occupational exposure, and to guard against the uncontrolled release of radioactive materials. Therefore, the staff concludes that the HI-STORE CIS Facility TS for functional and operating limits meet the requirements in 10 CFR 72.44(c)(1)(i).

16.3.2 Limiting Conditions for Operation and Surveillance Requirements

The staff reviewed the LCOs and SRs proposed by the applicant. SER table 16-2 lists the proposed LCOs and SRs included in the HI-STORE CIS Facility TS and the associated SER sections that discuss them.

Table 16-2 HI-STORE CIS Facility TS LCOs and SRs and Associated SER Sections

TS Item	LCO	Associated SR	Associated SER Sections
LCO 3.1.1	Spent Fuel Storage Cask Heat Removal System	SR 3.1.2	6.3.4
LCO 3.2.1	Canister Surface Contamination	SR 3.2.1	11.3.2.1

The staff confirmed that the LCOs listed in SER table 16-2 specify the lowest functional capability for the equipment required for safe operation and are consistent with corresponding LCOs in the certificate of compliance TS previously approved by the NRC. In addition, the staff confirmed that the SRs listed in table 16-2 provide for the necessary inspection and testing, confirm operation within appropriate functional and operating limits, and confirm that LCOs for safe storage are met. Therefore, the staff concludes that the HI-STORE CIS Facility TS meet the requirements in 10 CFR 72.44(c)(2) and (c)(3).

16.3.3 Design Features

The staff reviewed the design features portion of the TS, which includes items that could have a significant effect on safety if altered or modified, such as materials of construction or geometric arrangements. SER table 16-3 lists the design features included in the HI-STORE CIS Facility TS and the associated SER sections that discuss them.

Table 16-3 HI-STORE CIS Facility TS Design Features and Associated SER Sections

TS Item	Design Feature	Associated SER Sections
4.1	Site	2.3.1.1
4.2.1	Storage System	3.3.2
4.2.2	Storage Capacity	4.3.1
4.2.3	HI-STORM UMAX Vertical Ventilated Module Spacing	6.3.4.6
4.2.4	Site Temperature Limits	6.3.4.6, 17.3.5
4.2.5	Cask Transporter	5.3.5.1, 6.3.4
4.2.6	Cask Crane	5.3.4.4.4
4.2.7	Storage Pads	5.3.4.1
4.2.8	Special Lifting Devices	5.3.6
4.2.9	Canister Transfer Facility	5.3.4.3
4.2.10	Tilt Frame	5.3.7
4.3	Cask Transfer Building	5.3.4.4.1

The staff confirmed that the design features listed in SER table 16-3 are features that if altered could have a significant effect on safety. Therefore, the staff concludes that the HI-STORE CIS Facility TS meet the requirements in 10 CFR 72.44(c)(4).

16.3.4 Administrative Controls

The staff reviewed the administrative controls proposed in the HI-STORE CIS Facility TS. SER table 16-4 lists the administrative controls and the associated SER sections that discuss them.

Table 16-4 HI-STORE CIS Facility TS Administrative Controls and Associated SER Sections

TS Item	Administrative Controls	Associated SER Sections
5.1	Responsibility	10.3.1.1
5.2	Onsite and Offsite Organizations	10.3.1.1
5.3	HI-STORE CIS Facility Staff Qualification	10.3.4.1
5.4	Procedures	10.3.3.1
5.5.1	Radiological Effluent Control Program	9.3.3.1
5.5.2	Radiation Protection Program	11.3.4.3
5.5.3	Pre-operational Testing and Training Exercise of HI-STORE CIS Facility Systems and Equipment	10.3.2.2
5.5.4	Aging Management Program	17.3.16
5.5.5	Canister Acceptance Program	9.3.2.2
5.5.6	TS Bases Control Program	16.3.4

The staff confirmed that the administrative controls listed in SER table 16-4 are necessary to ensure that the operations involved in the storage of spent fuel at the facility are performed in a safe manner. Therefore, the staff concludes that the HI-STORE CIS Facility TS meet the requirements in 10 CFR 72.44(c)(5) and (d).

16.3.5 License Conditions

The regulations in 10 CFR 72.44 require that each license issued under 10 CFR Part 72 include license conditions that pertain to design, construction, and operation; the Commission may also include additional license conditions as it deems appropriate. The regulations also specify certain license conditions that apply to each license issued under 10 CFR Part 72, whether or not the license explicitly states those license conditions. The staff noted that the conditions specified in 10 CFR 72.44(b)(1) through (b)(6) are binding on the HI-STORE CIS Facility license but are not explicitly restated in that license.

SER table 16-5 lists the license conditions that the staff identified during its review of the HI-STORE CIS Facility application and associated documents, and the associated SER section detailing the staff's evaluation.

Table 16-5 HI-STORE CIS Facility License Conditions and Associated SER Sections

License Condition Number	License Condition Description	Associated SER Section
13	Emergency Plan	10.3.5
14	Physical Security Plan	10.3.6
15	Funding Operations	18.3.1.3
16	Financial Assurance	18.3.1.3
17	Startup Plan	10.3.2
18	Insurance Coverage	18.3.3
19	Decommissioning Funding Assurance	18.3.2

16.4 Evaluation Findings

Based on its review of the application, the staff concludes that applicant for the HI-STORE CIS Facility identified the necessary TS to satisfy the requirements of 10 CFR 72.44(c) and (d). The proposed TS provide reasonable assurance that the independent spent fuel storage installation will allow for the safe storage of spent fuel. This finding is based on the regulation itself, appropriate regulatory guides, and accepted practices.

16.5 References

Title 10, of the *Code of Federal Regulations*, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

Holtec International, “Attachment 2 - Proposed HI-STORE License/Technical Specifications,” November 23, 2022. ML22331A005.

17 MATERIALS EVALUATION

In chapter 17, “Material Evaluation,” of its safety analysis report (SAR), Holtec International (Holtec or the applicant) evaluated the materials considerations for the proposed HI-STORE Consolidated Interim Storage (CIS) Facility. The applicant stated that the materials assessment is largely provided by the evaluations previously performed for the HI-STORM UMAX and HI-STORM FW certificates of compliance (CoCs). The applicant also documented the evaluation of aging-related degradation of components in service beyond 20 years in SAR chapter 18, “Aging Management Program.” This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff’s review of all materials considerations, including aging-related issues.

17.1 Scope of Review

The staff’s review focused on the materials characteristics of the HI-STORE CIS Facility that are unique with respect to the materials designs previously reviewed and approved for the HI-STORM UMAX and HI-STORM FW CoCs.

For the component designs that the application incorporated by reference from the SARs of the previously approved storage systems, the staff evaluated the potential effects of site-specific environmental parameters and the longer service life with respect to the approved 20-year storage terms of the referenced storage systems. As described in SAR section 4.3, “Design Criteria for SSCs Important to Safety,” and SAR table 5.0.3, “Material Incorporated by Reference in this Chapter,” component design information incorporated by reference include the vertical ventilated modules (VVMs), multipurpose canisters (MPCs), and the independent spent fuel storage installation (ISFSI) pad and support foundation pad. In addition, the applicant relied on the HI-STAR 190 transportation cask (CoC No. 9373) to safely house the MPC before off-loading it in the canister transfer facility (CTF).

The remaining components unique to the HI-STORE CIS Facility, for which there is no prior staff evaluation, include the HI-TRAC CS transfer cask, vertical cask transporter (VCT), cask transfer building (CTB), CTB crane, CTF and its foundation, transport cask tilt frame and saddle, HI-PORT heavy haul trailer, and various special lifting devices.

17.2 Regulatory Requirements

The NRC requirements for the overall function of the ISFSI, the design criteria, and the operation of certain separate functional subsystems appear in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR):

- 10 CFR 72.24, “Contents of application: Technical information”
- 10 CFR 72.120, “General considerations”
- 10 CFR 72.122, “Overall requirements”
- 10 CFR 72.124, “Criteria for nuclear criticality safety”

- 10 CFR 72.128, “Criteria for spent fuel, high-level radioactive waste, reactor-related Greater than Class C waste, and other radioactive waste storage and handling”

17.3 Staff Review and Analysis

This SER chapter describes the staff’s review of systems associated with the receipt, transfer, storage, maintenance, and retrieval of the spent nuclear fuel (SNF). Unless otherwise stated, the staff reviewed and evaluated SAR chapters 10, 17 and 18, documents referenced in the SAR, the applicant’s responses to the staff’s requests for supplemental and additional information, and other relevant documents and literature. The staff reviewed and evaluated whether the materials for the structures, systems, and components (SSCs) meet applicable codes, standards, and specifications necessary to support the intended functions of the SSCs under all credible loads for normal, off-normal, and accident conditions, including the effects of environmental conditions and natural phenomena for the HI-STORE CIS Facility. The staff also evaluated operations that ensure adequate materials performance, including materials qualification, welding, drying of SNF, inerting of the confinement system, and the management of materials degradation.

The staff’s review followed the guidance in NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued March 2000, and Spent Fuel Project Office (SFST)-Interim Staff Guidance (ISG)-15, Revision 0, “Materials Evaluation,” dated January 10, 2001. In addition, for aging-related considerations for SSCs in service beyond 20 years, the staff considered the guidance in NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report,” issued July 2019. The staff notes that, although NUREG-2214 specifically addresses technical considerations associated with the renewal of an ISFSI license or CoC, the staff generally considers the information in that guidance to be informative of acceptable approaches to identify and manage aging-related degradation of materials such as those at the HI-STORE CIS Facility.

The staff’s review ensures that materials will support the functions of the storage system and site facility SSCs. Specifically, it ensures, among other things, that:

- The applicant provided information on materials of construction, including their fabrication, testing, and general arrangement, with sufficient detail to support a safety finding.

- Material properties have an adequate technical basis and demonstrate the ability to support the performance of the intended functions of SSCs under credible loads in normal, off-normal, and accident conditions.

- Materials will not undergo adverse environmental degradation, chemical reactions, or other reactions that could challenge the ability of SSCs to safety handle, package, transfer, and store SNF.

- The applicant ensures that the SNF cladding is protected against gross ruptures or is otherwise confined and that the SNF is always retrievable.

- Materials and special processes conform to all applicable codes and standards. Noncode materials have adequate controls for their qualification and fabrication.

17.3.1 Drawings

SAR section 1.5, “Licensing Drawings” (proprietary), includes the drawings for the HI-STORE CIS Facility components. The drawings include a bill of materials that states the material specifications and the safety category of each component. Additional drawing details and notes provide material alternatives, fabrication instructions, and additional material property requirements. The staff reviewed the drawings and found them to be acceptable because they contain an adequate description of the material specifications and fabrication instructions. Therefore, the staff determined that the materials information in the drawings is sufficient to meet the regulatory requirements in 10 CFR 72.24(c)(3) and 10 CFR 72.120(a).

17.3.2 Codes and Standards

SAR chapter 5, “Installation and Structural Evaluation,” SAR section 17.3, “Applicable Codes and Standards,” and the drawings in SAR section 1.5 (proprietary) describe the materials codes and standards. The staff previously reviewed and approved of the codes and standards for the MPC, VVM, ISFSI pad, and support foundation pad for the HI-STORM UMAX and HI-STORM FW CoCs; therefore, the staff finds them to be acceptable. For the components described below that are unique to the HI-STORE CIS Facility, the staff evaluated whether the materials are consistent with the design code or standard.

Cask Transfer Components

As described in SAR section 5.4, “Other SSCs Important to Safety,” and section 5.5, “Other SSCs,” the applicant designed the structural steel components of the HI-TRAC CS transfer cask, transport cask tilt frame and saddle, and CTF (including the HI-STAR 190SL pedestal) in accordance with Section III, Division 1, Subsection NF, “Supports,” of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, issued 2010. The structural components of the tilt fame and CTF are constructed with materials procured to ASME B&PV Code, Section II, “Materials.” Thus, they are consistent with the design code, and the staff finds them to be acceptable. The transfer cask is constructed primarily with materials procured to ASTM International (ASTM) standards. The staff finds the use of ASTM materials standards for the transfer cask to be acceptable, as these consensus industry standards provide adequate control for chemistry, strength, and ductility.

Regarding the VCT, SAR section 4.5.3, “Vertical Cask Transporter,” states that the jack/lifting towers shall be designed in accordance with ASME B&PV Code, Subsection NF, and ASME B30.1-2009, “Jacks, Industrial Rollers, Air Casters, and Hydraulic Gantries.” The VCT overhead beam and MPC downloader pully/pins shall be designed in accordance with NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” issued July 1980, and American National Standards Institute (ANSI) N14.6, “American National Standard for Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4 500 kg) or More,” issued 1993. The VCT drawings show that all materials of construction conform to ASTM standards, primarily ASTM A514 and A572 high strength steels.

The staff notes that the guidance in NUREG-1567 (through its reference to NUREG-1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility”) recognizes ASTM standards as appropriate for structures that do not perform a confinement function. Therefore, based on the applicant’s use of ASTM materials specifications for the construction of the VCT, the staff finds the materials standards to be acceptable.

SAR section 4.5.4, “HI-PORT,” states that the important-to-safety (ITS) drop deck of the HI-PORT heavy haul trailer is designed to meet the stress limits of ASME B&PV Code, Section III, Subsection NF. The applicant stated that the materials of construction will be structural or pressure vessel grade carbon steels manufactured to U.S. consensus codes and standards. The staff reviewed the proprietary Holtec Purchase Specification PS-5025-001, Revision 0, “Purchase Specification for the HI-STORE HI-PORT,” dated March 16, 2022, and verified that the specification has controls to ensure that the structural materials will be manufactured to consensus standards and will meet minimum required mechanical properties. In addition, the staff verified that the purchase specification includes documentation requirements to allow Holtec to verify the adequacy of the materials procurement and the associated examinations for materials integrity. Therefore, the staff finds the HI-PORT materials standards to be acceptable, as the applicant has adequate controls in place to ensure that the procured materials will be capable of fulfilling their structural functions.

Special Lifting Devices

As stated in SAR section 4.5.1, “Design Requirements Applicable to Lifting Devices and Special Lifting Devices,” the special lifting devices are designed to ANSI N14.6. Special lifting devices include the MPC lift attachment, transport cask horizontal lift beam, cask lift yokes, transfer cask lift link, and the MPC lifting device extension. The staff notes that ANSI N14.6 does not specify the use of a specific materials code or standard, but ANSI N14.6 includes qualification requirements for the materials, such as a drop weight test for ferritic steels to establish the nil ductility transition temperature. The staff reviewed the drawings for the lifting components and notes that the materials are specified in accordance with either ASME B&PV Code, Section II; ASTM standards; or the European Committee for Standardization standard EN-10025, “Hot Rolled Products for Structural Steels,” issued 2004. The staff finds the materials standards to be acceptable because the ASME, ASTM, and EN-10025 standards provide for adequate control of the material chemistry and mechanical properties of the special lifting devices.

Cask Transfer Building and Crane

In SAR section 4.6, “Design Criteria for the Cask Transfer Building (CTB),” the applicant stated that the steel framework of the CTB is designed to the requirements of ANSI/American Institute of Steel Construction (AISC) 360, “Specification for Structural Steel Buildings,” issued 2016. This standard defines acceptable ASTM structural steel grades. The staff reviewed the CTB drawing and confirmed that the applicant cited the use of steels that are consistent with the ANSI/AISC 360 specifications, including ASTM A325 or A490 high-strength alloy steel bolts and ASTM A572 grade 50 high-strength, low-alloy structural steel beams and girders. The applicant also included the optional use of ASTM A992 structural steel for the beams and girders. This grade is not identified in the ANSI/AISC standard. However, the staff notes that ASTM A992 is a commonly used steel grade for structural beams, and it is frequently used in applications that previously specified ASTM A572 because it provides similar, but better defined, mechanical property controls.

In SAR section 4.5.2, “Cask Transfer Building (CTB) Crane,” the applicant stated that the CTB crane shall be designed and built in accordance with the provisions of ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” issued 2015. In addition, SAR section 4.5.2.9, “Material Requirements,” states that construction materials will comply with ASME NOG-1, Subsection 4200, “Materials and Connections.” Subsection 4200 defines acceptable ASTM grades for structural components and includes fracture toughness testing requirements. The staff reviewed the CTB crane drawings and noted that several crane components are specified to structural steel grade EN-10025 S355J2, which is commonly used for structural applications.

As discussed above, the staff reviewed the materials for the CTB and CTB crane and finds the cited codes and standards to be acceptable. The design standards provide for adequate controls of material properties to ensure that the required materials performance is achieved.

Reinforced Concrete Components

SER section 17.3.10 documents the staff’s review of the reinforced concrete in the CTB and CTF foundation.

Based on the reviews above and SER section 17.3.10, the staff determined that the information provided by the applicant on the applicable codes and standards for the components unique to the HI-STORE CIS Facility is sufficient to meet the regulatory requirements in 10 CFR 72.24(c)(4) and 10 CFR 72.122(a).

17.3.3 Welding

SAR section 17.5, “Welding Material and Welding Specification,” and the proprietary drawings in SAR section 1.5 describe the welding materials and specifications for the HI-STORE facility components. The staff previously reviewed and approved the welding specifications for the MPC and VVM for HI-STORM UMAX CoC No. 72-1040; therefore, the staff finds them to be acceptable. For components unique to the HI-STORE CIS Facility, the staff evaluated whether the welding practices are consistent with the construction codes and standards.

Cask Transfer Components

The applicant designed the HI-TRAC CS transfer cask, transport cask tilt frame and saddle, and CTF in accordance with ASME B&PV Code, Section III, Division 1, Subsection NF. The majority of welds in the tilt frame and structural welds in the transfer cask and CTF are performed in accordance with Subsection NF. The staff finds these welds performed in accordance with Subsection NF to be acceptable.

Some welds in the transfer cask and CTF are not performed in accordance with Subsection NF of the ASME B&PV Code. The applicant stated in SAR section 17.5 that those welds are considered “non-structural,” as they are subject to minor stresses and are not critical to the safety of the part. The applicant stated that the criteria for determining whether a weld is nonstructural follows Subsection NF-1215 for secondary members (i.e., secondary members are those that do not sustain “significant” stresses in excess of 50 percent of the allowable stress). Notwithstanding the nonstructural designation, in SAR section 17.5, the applicant stated that weld procedures shall comply with ASME B&PV Code, Section IX, “Welding and Brazing Qualification,” or American Welding Society (AWS) D1.1, “Structural Welding Code—Steel,” issued 2008. In addition, the welds shall be made by a welder qualified to ASME B&PV Code, Section VIII, or AWS D1.1, and the welds shall be visually examined by personnel qualified to Level 1 in accordance with American Society for Nondestructive Testing (ASNT) SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” issued 1992. The staff reviewed the nonstructural welds and finds them to be acceptable because welding by qualified personnel in accordance with the ASME or AWS code is consistent with the guidance in NUREG-1567 and SFST-ISG-15, which provides assurance that these welds will fulfill their function.

SAR section 4.5.3.2, “Fabrication,” states that all welding associated with the VCT shall comply with ASME B&PV Code, Section IX, or AWS D1.1. The staff considers both the ASME and AWS industry consensus welding codes to provide appropriate control of welding processes for SSCs constructed of structural steels; therefore, the staff finds the VCT weld design and fabrication to be acceptable.

SAR section 4.5.4 states that all welding associated with the ITS drop deck of the HI-PORT heavy haul trailer shall conform to national consensus standards such as AWS D1.1. The staff reviewed the proprietary purchase specification for the HI-PORT and verified that the specification has controls in place to ensure that the design, qualification, and examination of welds conform to AWS D1.1 or other acceptable national standards defined in the specification. Therefore, the staff finds the HI-PORT weld criteria to be acceptable.

Special Lifting Devices

For the special lifting devices, ANSI N14.6 requires the qualification of welding procedures and welders in accordance with ASME B&PV Code, Section IX, or AWS D1.1. The staff reviewed the proprietary drawings in SAR section 1.5 for all the lifting components designed to ANSI N14.6 and notes that the drawings specify that welding is to be performed to ASME B&PV Code, Section IX, or AWS D1.1, consistent with the ANSI N14.6 requirements. Therefore, the staff finds the welding of the lifting devices to be acceptable.

Cask Transfer Building and Crane

The steel framework for the CTB is designed to ANSI/AISC 360. The staff notes that this standard cites the use of AWS D1.1 for all weld fabrication, with some exceptions. In SAR section 4.5.2, the applicant stated that the CTB crane is designed to ASME NOG-1, which requires welding to be performed in accordance with AWS D1.1, with some exceptions.

The applicant stated in SAR section 17.5 that the weld procedures shall comply with ASME B&PV Code, Section IX, or AWS D1.1; the welds shall be made by a welder qualified to ASME B&PV Code, Section VIII, or AWS D1.1; and the welds shall be visually examined by personnel qualified to Level 1 under ASNT SNT-TC-1A. Because the CTB and CTB crane will be welded in a manner consistent with these construction standards, the staff finds the welding practices to be acceptable.

Based on the reviews above, the staff determined that the information provided by the applicant on the welding specifications for the components unique to the HI-STORE CIS Facility demonstrates adequate control of fabrication processes and meets the regulatory requirements in 10 CFR 72.24(c)(4) and 10 CFR 72.122(a).

17.3.4 Mechanical Properties of Metallic Materials

17.3.4.1 Component Designs Incorporated by Reference

The staff reviewed SAR chapter 4, “Design Criteria for the HI-STORE CIS Systems, Structures, and Components,” and SAR chapter 5, “Installation and Structural Evaluation,” and verified that the materials and mechanical properties used in the design of the VVMs, MPCs, ISFSI pad, and support foundation pad have been previously reviewed and approved in the HI-STORM UMAX CoC. As stated in SAR table 17.0.2, “Material Incorporated by Reference,” the mechanical properties are incorporated by reference from the HI-STORM UMAX SAR, Revision 3, dated June 29, 2016. For the MPCs, the HI-STORM UMAX SAR, in turn, incorporates by reference the mechanical properties in the HI-STORM FW SAR, Revision 4, dated June 24, 2015.

Although the materials and mechanical properties for components incorporated by reference are identical to those previously approved in the HI-STORM UMAX CoC, the staff evaluated whether the properties remain valid in consideration of site-specific temperatures and the components’ longer service life relative to the approved 20-year storage terms of the referenced storage system designs.

Regarding site-specific temperatures, the staff reviewed SAR table 6.0.1, “Material Incorporated by Reference in this Chapter,” SAR table 6.3.1, “Thermally Significant Parameters for the HI-STORM UMAX ISFSI at HI-STORE and Corresponding Certified Value in the System FSAR,” and the associated references and verified that the thermal conditions of the components at the HI-STORE CIS Facility are bounded by those evaluated in the HI-STORM UMAX SAR. As a result, the staff finds that the material properties specified in the HI-STORM UMAX SAR appropriately account for the effects of the HI-STORE site-specific temperatures.

Regarding the service life, the staff evaluated the mechanical properties for the proposed 40-year HI-STORE CIS Facility license duration, which contrasts with the 20-year duration of the previously approved HI-STORM UMAX CoC. In addition, the staff considered that the first MPC-37 and MPC-89 canisters were loaded at their original ISFSI sites in 2015 (UxC, 2022), such that their total service life could approach 50 years in the proposed storage term of the HI-STORE CIS Facility. As a result, the staff considered this potential extended service time in the evaluation of the mechanical properties for the MPCs. Below, the staff documents its evaluation of the long-term mechanical properties for the components incorporated by reference.

Multipurpose Canister Confinement Boundary

The staff verified that the materials and mechanical properties used in the MPCs are identical to those used in the HI-STORM UMAX and HI-STORM FW designs. The MPC confinement boundary is constructed of either ASME SA-240 (plate) or SA-336 (forged) Alloy 304, 304LN, 316, or 316LN stainless steels. An option exists to construct the MPC lid with either solid stainless steel or ASTM A36 or ASME SA-516 carbon steel with a stainless-steel veneer.

ASME B&PV Code, Section II, Part D, “Material Properties,” defines the maximum allowable operating temperatures of the steel and stainless-steel grades used in the MPCs, below which time-dependent mechanical properties need not be considered (i.e., the design does not need to account for a change in properties over time). The staff reviewed SAR table 6.4.3, “Normal Long-Term Storage Temperatures for MPC-37 in HI-STORM UMAX at HI-STORE CIS,” as well as the results of the applicant’s thermal analyses of several shorter-term operations in SAR section 6.4.3, “Calculations and Results,” and verified that the maximum operating temperature of the MPC confinement boundary remains below the ASME B&PV Code temperature thresholds. Therefore, the mechanical properties used for the structural analyses of the confinement boundary for the HI-STORM UMAX and HI-STORM FW designs remain valid for the potential MPC service life at the HI-STORE CIS Facility. Regarding the effects of prolonged exposure to radiation, the staff notes that NUREG-2214 estimates that the maximum cumulative neutron fluence levels at the middle of the MPC basket over 100 years is about three orders of magnitude below the levels that would be expected to degrade the mechanical properties (specifically, fracture resistance) of carbon and stainless steels. As a result, because the steel and stainless steel MPC components are not exposed to temperatures or neutron fluence levels that are expected to affect mechanical properties, the staff finds the use of the properties incorporated by reference from the HI-STORM UMAX and HI-STORM FW SARs to be acceptable.

Multipurpose Canister Fuel Basket

HI-STORM FW SAR table 1.2.8a, “Minimum Guaranteed Values of Metamic-HT Primary Properties,” provides the minimum mechanical properties of the Metamic-HT basket material, which were derived from testing described in proprietary Holtec Report No. HI-2084122, Revision 13, “Metamic-HT Qualification Sourcebook,” issued 2017. The staff reviewed the Metamic-HT test results and notes that test coupons exposed to thermal aging and irradiation treatments to simulate 40 years of service continued to meet the minimum properties in table 1.2.8a. Because the MPCs stored at the HI-STORE CIS Facility may exceed 45 years of service (including storage before arriving at the site), the staff evaluated whether those same test results provide a sufficient technical basis to conclude that the baskets remain capable of fulfilling their structural function at the HI-STORE CIS Facility.

The staff notes that the Metamic-HT test program simulated continuous exposure to heat and radiation for 40 years, assuming that the exposure levels present at the time of cask loading remain constant and do not decay over time. The staff evaluated the test results with respect to a more realistic service exposure, in which the temperature and radiation exposure are known to be greatest early in the MPC's service life and are significantly reduced at the longer MPC service times that may be reached at the HI-STORE CIS Facility (IAEA, 2003). Based on the staff's extrapolation of the applicant's test data to the more realistic decay profile and potentially longer-term service exposure at the HI-STORE CIS Facility, the staff determined that the testing adequately demonstrates that the minimum mechanical properties will continue to be met. Therefore, the staff finds the use of the previously approved Metamic-HT mechanical properties from the HI-STORM FW SAR to be acceptable.

HI-STORM FW SAR section 3.3.2, "Nonstructural Materials," states that the aluminum used in the basket shims is not a structural material because it does not withstand any tensile loads and it resides in a confined space that prevents uncontrolled deformation. The HI-STORM FW SAR states that the yield strength is a critical characteristic in that system design because it is an input into the cask tipover analysis. However, the staff notes that the HI-STORE design does not include cask tipover as a credible event, as documented in SAR section 15.3.13, "Tip-over." As a result, the aluminum shim mechanical properties are not relied on to sustain applied loads. Therefore, the impact of increased service time does not need to be evaluated.

Vertical Ventilated Modules

In SAR table 17.0.2, the applicant incorporated by reference the materials and mechanical properties of the VVMs from the drawings and section 3.3, "Mechanical Properties of Materials," of the HI-STORM UMAX FSAR, Revision 3. The applicant included in HI-STORE SAR section 1.5 the drawings for the specific version of the VVMs (Version C) to be used at the HI-STORE CIS Facility. The majority of the ITS structural components of the VVMs are constructed of ASTM A516 Grade 70 carbon steel. Stainless steel is used for a small number of components (e.g., ASME SA-240 304 divider shell guide cover). The VVM structures include the cavity enclosure container (CEC), the divider shell, and the concrete-filled closure lid.

The staff reviewed the mechanical properties of the VVM structural materials to verify that they are adequate for the proposed 40-year license term. Similar to the discussion above for the MPC materials, the temperatures of the A516 Grade 70 carbon steel and the SA-240 304 stainless steels used in the construction of the VVM remain below the ASME B&PV Code thresholds for which time-dependent properties need to be considered. For example, as shown in table 4.4.9, "Normal Long-Term Storage Temperatures for Limiting MPC-37 with Short Fuel (Under Heat Load Chart 1) at an Elevated Site," in the HI-STORM UMAX FSAR, the VVM divider shell is expected to be less than 177 degrees Celcius (°C) (350 degrees Fahrenheit (°F)), compared to the 371°C (700°F) maximum operating temperature described for ASME A516 Grade 70 in ASME B&PV Code, Section II, Part D. Also, as discussed above for the MPCs, the neutron fluence exposure of the VVM components is expected to be well below levels that may affect mechanical properties. As a result, because the steel and stainless steel VVM components are not exposed to temperatures or neutron fluence levels that are expected to affect mechanical properties, the staff finds the use of the referenced mechanical properties from the HI-STORM UMAX SAR to be acceptable.

17.3.4.2 HI-STORE Site-Specific Components

For the components described below that were not previously reviewed by the staff for the HI-STORM UMAX or HI-STORM FW CoCs, the staff evaluated the mechanical properties of materials used in the structural analyses in consideration of the service environment and the proposed 40-year license of the HI-STORE CIS Facility.

HI-TRAC CS Transfer Cask

In SAR section 5.4.2.3, “Material Properties,” the applicant stated that the mechanical properties of the materials for the HI-TRAC CS transfer cask are those in section 3.3, “Mechanical Properties of Materials,” of the HI-STORM FW final safety analysis report (FSAR). The staff notes that, although the design of the HI-STORE transfer cask is a variation of the HI-STORM FW transfer cask, the two designs use the same structural steels. The HI-TRAC CS is constructed of ASTM A516 Grade 70 carbon steel inner and outer shells, with concrete in the intervening space. The bottom of the transfer cask has ASTM A516 Grade 70 carbon steel gates that allow passage of the MPC. A set of trunnions at the top of the cask is used for lifting, and a set of bottom trunnions is used for tilting operations. The trunnions are constructed of either ASME SA-564 Grade 630 precipitation-hardened martensitic stainless steel or ASME SB-637 Grade 718 nickel-based alloy.

The staff reviewed the transfer cask drawings in SAR section 1.5 and mechanical properties in section 3.3 of the HI-STORM FW SAR and confirmed that the mechanical properties are in accordance with the values in ASME B&PV Code, Section II, “Properties,” and that the exposure temperatures do not exceed the ASME thresholds for which time-dependent properties are required. Therefore, the staff finds the mechanical properties used in the structural analysis of the transfer cask to be acceptable.

Canister Transfer Facility (Including HI-STAR 190SL Pedestal)

SAR section 5.4.7, “Cask Transfer Facility Steel Structure,” states that the steel structure of the CTF is designed to the stress limits of ASME B&PV Code, Subsection NF, and the drawings in SAR section 1.5 show that the materials of construction for the ITS components include ASME SA-516 Grade 70 carbon steel plates, SA-193 Grade B7 alloy steel bolting, SA-350 Grade LF2 carbon steel forging, SA-240 Grade 304 stainless steel plate, and commercial steel piping. The pedestal used to raise the height of the shorter HI-STAR 190SL transportation cask in the CTF is constructed of ASME SA-516 Grade 70 carbon steel. The SAR states that the minimum strength properties for these materials are obtained from the applicable ASTM standards or ASME B&PV Code, Section II, Part D.

The staff reviewed the structural analysis for the CTF and confirmed that the mechanical properties used in the analyses are in accordance with the applicable materials standards. The staff also verified that the service temperatures provided in the thermal analyses for the transfer facility do not exceed the ASME thresholds for which time-dependent properties are required. Therefore, the staff finds the mechanical properties used in the structural analysis of the CTF to be acceptable.

Tilt Frame

SAR section 5.5.1.3, "Material Properties," states that the transport cask tilt frame is constructed of ASME SA-516 Grade 70, ASTM A572, and ASTM A500 Grade B carbon steels. The SAR states that the minimum strength properties of these materials are obtained directly from the applicable ASME and ASTM specifications. The staff reviewed the mechanical properties for the tilt frame and saddle materials and confirmed that they conform with the applicable materials standards; therefore, the staff finds them to be acceptable.

Vertical Cask Transporter

SAR section 4.5.3.9, "Material Failure Modes," states that material properties and allowable stress values for structural steel members of the VCT shall be taken from the applicable national consensus standard. The VCT drawings show that all materials of construction conform to ASME and ASTM materials standards. The staff finds the mechanical properties of materials used in the VCT design to be acceptable because the use of properties from ASME and ASTM consensus standards is consistent with the guidance in SFST-ISG-15 for the design of major structural components.

Cask Transfer Building and Crane

The staff reviewed SAR section 4.6, "Design Criteria for the Cask Transfer Building (CTB)", and proprietary Holtec Report No. HI-2210576, Revision 1, "Structural Analysis of the HI-STORE Cask Transfer Building," dated March 24, 2022, and verified that the mechanical properties used for the CTB steels in the structural analysis are consistent with those defined in the steel construction standard AISI/AISC 360 (and the ASTM materials standards referenced therein). Therefore, the staff finds the CTB steel properties to be acceptable.

For the CTB crane, SAR section 4.5.2.9, "Material Requirements," states that the crane construction materials will comply with ASME NOG-1, Subsection 4200. The staff notes that ASME NOG-1 defines acceptable ASTM grades for structural components and includes fracture toughness testing requirements. Although the applicant did not identify the mechanical properties of the steels to be used for the CTB crane (as the manufacturer is to perform the structural qualification of the crane), the design standard provides for controls of materials strength and fracture toughness. In addition, the use of ASTM steel grades will govern materials procurement to ensure that the required materials performance is achieved. Therefore, the staff finds the applicant has adequate controls in place to ensure that materials with appropriate mechanical properties are used in the construction of the CTB crane.

Special Lifting Devices

In SAR section 5.4, "Other SSCs Important to Safety," and the drawings in SAR section 1.5, the applicant indicated that the cask lifting components are constructed of structural steels (ASTM A514, A336 F6NM, A53, A500 Grade B, and ASME SA-516 Grade B) and nickel-based alloys (SB-637 Grade 718). SAR section 5.4.6.3, "Material Properties," states that the material properties are in accordance with the applicable ASTM specifications and the ASME B&PV Code. The staff reviewed the mechanical properties for the special lifting devices and confirmed that they conform with the applicable materials standards; therefore, the staff finds them to be acceptable.

HI-STAR 190 Transportation Cask

SAR table 4.4.4, “HI-STAR 190 Materials Temperature Limits,” provides the materials temperature limits of the HI-STAR 190 transportation cask (including the contained MPCs) under short-term operations and accident conditions. The staff notes that the material temperature limits of the MPC subcomponents while housed within the HI-STAR 190 transportation cask at the HI-STORE CIS Facility are bounded by those described in HI-STORM UMAX FSAR table 2.3.7, “Design Temperatures,” for short-term operations (e.g., MPC drying and onsite transport). The staff also notes that the material temperature limits for the HI-STAR 190 subcomponents at the HI-STORE CIS Facility are bounded by those described in HI-STAR 190 transportation SAR table 3.2.10, “HI-STAR 190 Materials Temperature Limits” (Holtec Report No. HI-2146214, Revision 3). Because the materials and mechanical properties of the HI-STAR 190 transportation cask are identical to those used in the approved HI-STAR 190 transportation package, and because the conditions at the HI-STORE CIS Facility are bounded by those considered in the HI-STAR 190 transportation SAR, the staff find the mechanical properties to be acceptable.

Based on the reviews above, the staff determined that the mechanical properties used in the design of the HI-STORE CIS Facility are appropriate, and the information in the SAR therefore meets the regulatory requirements in 10 CFR 72.24(d), 10 CFR 72.122(a) and (b), and 10 CFR 72.124(b).

17.3.5 Fracture Toughness of Ferritic Steels

In SAR section 17.4.3, “Protection Against Brittle Fracture of Ferritic Steel Parts,” the applicant stated that the risk of brittle fracture of ferritic steel subcomponents is eliminated by using materials that maintain high fracture toughness at temperatures down to -40°F (-40°C). The discussion below documents the staff’s review of the brittle fracture performance of each component at the HI-STORE CIS Facility that relies on the structural performance of ferritic steels.

Component Design Incorporated by Reference (Vertical Ventilated Modules)

The applicant stated that the fracture criteria for ferritic steels used to construct the VVMs are provided in HI-STORM UMAX SAR table 3.1.9, “Fracture Toughness Test Requirements,” which includes impact testing in accordance with ASME B&PV Code, Subsection NF. The staff verified that the thermal conditions of the VVM components at the HI-STORE CIS Facility are bounded by those evaluated in the HI-STORM UMAX SAR. As a result, the staff finds that the fracture testing criteria used in the HI-STORM UMAX SAR appropriately account for the effects of the HI-STORE site-specific temperatures. Therefore, the staff finds the approach to demonstrate adequate fracture performance of the VVMs to be acceptable.

Transfer Equipment and Cask Transfer Facility

In SAR section 17.4.3, the applicant defined the lowest service temperature used to establish fracture toughness testing requirements for components used in transfer operations (HI-TRAC CS, tilt frame, VCT, CTF) as 0°F (-18°C), which section 4.2.4, “Site Temperature Limits,” of the technical specifications establishes as the lowest allowable temperature for short-term transfer operations. SAR table 17.4.1, “Fracture Toughness Test Requirements for HI-TRAC CS,” through table 17.4.4, “Fracture Toughness Test Requirements for Vertical Cask Transporter,” provide the detailed fracture testing requirements and acceptance criteria for the transfer components. The staff reviewed the ferritic steel fracture toughness requirements and verified that they are in accordance with ASME B&PV Code, Subsection NF, which is the design and fabrication code for the subject components. Therefore, the staff finds the applicant’s criteria to ensure adequate fracture toughness of the HI-TRAC CS, tilt frame, VCT, and CTF to be acceptable.

Special Lifting Devices

SAR section 4.5.1.2, “Stress Compliance Criteria Applicable to Special Lifting Devices (SLDs),” provides two options to demonstrate adequate fracture performance of ferritic steels used to construct special lifting devices. The steels may be impact tested in accordance with the requirements and acceptance criteria of ASME B&PV Code, Subsection NF, or they may be impact tested with acceptance criteria defined by a methodology that is based on NUREG/CR-1815, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick,” issued August 1981.

For the first option, the staff finds that the ASME B&PV Code provides sufficient controls to ensure adequate fracture performance, similar to its findings in the staff’s review of the VVM and transfer components given above. For the second option, the measured impact toughness is compared against a toughness threshold defined in NUREG/CR-1815 by material thickness, applied stress, and fracture performance data for structural steels. In its evaluation of the second option, the staff considered potential uncertainties with respect to how the toughness acceptance criteria are calculated, including the fact that the materials data that form the basis for the NUREG/CR-1815 methodology were gathered from steels with a lower strength than may be used for special lifting devices. However, the staff notes that additional assurance of adequate performance is provided by the fact that special lifting devices will also be inspected and tested in accordance with ANSI N14.6, as described in SAR table 10.2.1, “Pre-Operational, Startup, and Other Tests,” and table 10.3.1, “Maintenance and Inspection Activities for the HI-STORM UMAX VVM Systems.” The additional ANSI N14.6 activities include initial-use load tests (at 150 percent of the maximum service load) as well as annual load testing or inspections, or both, to verify the devices are free from damage. As a result, based on the use of impact tests to verify adequate fracture toughness of the lifting device steels and the performance of load testing and inspections in accordance with ANSI N14.6, the staff finds the applicant’s activities to demonstrate adequate performance of ferritic steel special lifting devices to be acceptable.

Based on the reviews above, the staff determined that the information provided by the applicant on the fracture performance of the structural steels is sufficient to meet the regulatory requirements in 10 CFR 72.24(d) and 10 CFR 72.122(a) and (b).

17.3.6 Creep

In SAR section 17.4.4, the applicant referenced the analysis in the HI-STORM FW FSAR to address the potential for creep of the Metamic-HT fuel basket. The staff notes that the fuel basket components are considered the only SSCs that experience temperatures that are sufficiently elevated to warrant the assessment of creep. HI-STORM FW FSAR section 8.4.4.1 provides an analysis to estimate the degree of creep that may occur in the vertically positioned fuel basket over 60 years of service. The analysis assumed that the basket temperature is initially 350°C) (662°F), then decreases linearly to 150°C (302°F) over 60 years. The applicant calculated that the basket would shrink 0.044 inches over 60 years. In its initial review of the HI-STORM FW CoC, the staff concluded that the applicant adequately demonstrated acceptable creep performance for the Metamic-HT fuel baskets (NRC, 2011). The staff notes that creep is most relevant in the initial years of storage when there is fuel decay heat to drive the creep mechanism. Extended storage terms, even beyond the 60-year analysis provided in the HI-STORM FW FSAR, would not be expected to introduce the potential for significant additional creep strain. For these reasons, the staff finds the applicant's evaluation for creep to be acceptable to demonstrate adequate performance of the fuel baskets. Based on its review, the staff determined that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 72.24(d), 10 CFR 72.124(b), and 10 CFR 72.128(a).

17.3.7 Thermal Properties of Materials

In SAR table 6.0.1, the applicant stated that the thermal properties of the materials used to construct the MPC, VVM, and HI-TRAC CS transfer cask are incorporated by reference from the HI-STORM UMAX SAR. In addition, the thermal performance of the HI-STAR 190 transportation cask used the thermal properties specified in the HI-STAR 190 transportation SAR. The staff reviewed the thermal properties in the referenced SARs (e.g., thermal conductivity, thermal expansion, heat capacity) and finds the use of the previously approved properties to be acceptable.

In SAR section 17.2.2, "Non-Structural Materials," the applicant stated that the VVM divider shell insulation shall be suitable for high temperature and high humidity and will be foil faced or jacketed to ensure that it is water resistant. SAR table 17.2.2, "Acceptance Criteria for the Selection of the Insulation Material," describes the required characteristics of the insulation material and notes that Kaowool ceramic fiber insulation is one material that satisfies the requirements and has been used at HI-STORM UMAX ISFSIs to date. The staff reviewed the divider shell insulation requirements and verified that they are identical to those in the previously reviewed and approved HI-STORM UMAX CoC. The staff did not identify any insulation properties that would be expected to change by extending the service life to 40 years at the HI-STORE CIS Facility, versus the 20-year license for which the insulation was initially reviewed. In addition, the staff notes that the VVM maintenance and aging management programs (AMPs), discussed in SER section 17.3.16, include periodic inspections that are capable of monitoring for the condition of the insulation jacketing. Therefore, the staff determined the use of the previously reviewed divider insulation properties to be acceptable and that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 72.24(d) and 10 CFR 72.122(b).

17.3.8 Radiation Shielding Properties

In SAR table 7.0.1, “Material Incorporated by Reference in this Chapter,” the applicant stated that the storage system incorporates by reference the same shielding design and evaluation methodology as that of the HI-STORM UMAX storage system. Sections 1.2.3.1, “Shielding Materials,” and 7.3, “Confinement Requirements for Hypothetical Accident Conditions,” of the HI-STORE SAR describe the principal shielding materials as steel, concrete, and the subgrade. The composition and density of the materials used in the shielding model are provided in table 7.3.1, “Composition of the Materials - HI-STORE CIS Facility,” of the HI-STORE SAR (plain concrete, reinforced concrete, and soil) and table 5.3.2, “Composition of the Materials in the HI-STORM FW System,” of the HI-STORM UMAX FSAR (steel and Metamic-HT).

The staff reviewed the description of the shielding materials in the SAR and verified that the material density characteristics used in the shielding analyses are appropriate. The staff also verified that manufacturing controls are in place to ensure that the concrete used in the VVM and HI-TRAC CS does not include gaps or other imperfections that could impact shielding effectiveness. For the plain concrete used in the transfer cask, SAR section 1.2.4, “Operational Characteristics of the HI-STORM UMAX,” states that the concrete must meet the requirements in appendix 1.D to the HI-STORM 100 Cask System SAR (Holtec Report No. HI-2002444, Revision 14). The staff notes that the referenced appendix 1.D includes construction requirements to ensure placement without voids, such as limiting the size of the concrete lifts (pours) and the use of vibrating equipment.[2]

The applicant’s HI-TRAC CS shielding analysis also accounted for potential concrete degradation in a high-temperature accident event. As stated in SAR table 4.4.1, “Permissible Temperature Limits for HI-TRAC CS and CTF Materials,” any concrete that exceeds 1,100°F (593°C) is assumed to be unavailable for shielding. In addition, as described in SAR table 7.3.1, the remainder of the concrete is assumed to have a reduced density (approximately 21 percent) relative to the design requirements and a total loss of hydrogen in a high-temperature accident. The staff reviewed the assumed reductions in concrete performance against the technical literature on laboratory testing and the operating experience of concrete exposed to high temperatures. The staff considered data on mechanical performance, weight (water) loss, and degradation (e.g., Hager, 2013; Arioz, 2007), changes in shielding performance (Peterson, 1960), and cracking and spalling in fire events (e.g., Jansson and Boström, 2008). Based on the staff’s review of the technical literature, the staff finds that the applicant’s shielding

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[2] The staff compared appendix 1.D in Revision 14 of the HI-STORM 100 SAR, which NRC has not approved, with Revision 15, which NRC approved in Amendment 10 to the HI-STORM 100 CoC, issued on May 25, 2016. The construction requirements to ensure placement without voids are identical in the two revisions.

analyses adequately accounted for potential reductions in concrete properties at high temperature. The staff considers the penalties Holtec applied to the concrete shielding performance to reasonably bound potential water loss and cracking phenomena under high-temperature exposures.

The subgrade, which fills the space between the VVMs, is also credited in the applicant's shielding analysis. The subgrade is constructed of a cementitious material, either controlled low strength material (defined in American Concrete Institute (ACI) 229R-99, "Controlled Low-Strength Materials"), issued in 1999, or lean concrete (defined in ACI 116R-00, "Cement and Concrete Terminology"). The staff notes that the subgrade material specifications are identical to those in the HI-STORM UMAX system CoC, for which the staff previously found the subgrade is capable of performing its shielding function in the below-grade service environment. In NUREG-2214, the staff generically concluded that neither the normal long-term temperatures nor the radiation environments associated with storage systems are expected to degrade adjacent cement-based materials. The staff notes that the subgrade service temperature (less than 111°F (44°C) per SAR table 6.4.3) and radiation exposure adjacent to the HI-STORE VVMs are bounded by the parameters considered in the NUREG-2214 analyses. Therefore, the staff finds that the applicant adequately accounted for any potential reduction in shielding performance in the subgrade design.

Therefore, based on the staff's review of the material density characteristics, fabrication controls, and the applicant's consideration of potential concrete degradation at high temperatures, the staff finds the materials properties used in the shielding analyses to be acceptable and that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 72.24(d), 10 CFR 72.122(a), and (b)(1), and 10 CFR 72.128(a).

17.3.9 Criticality Control Materials

In SAR section 17.9, "Neutron Absorbing Materials," the applicant stated that the MPC fuel basket is constructed of the Metamic-HT neutron absorber, which is a composite of aluminum oxide and boron carbide particles in an aluminum matrix. The staff notes that Metamic-HT fulfills both structural and criticality control functions, and it has been approved previously in the HI-STORM FW and HI-STORM UMAX CoCs for use for a 20-year service life. In SAR table 17.0.2, the applicant incorporated by reference the Metamic-HT properties described in HI-STORM FW FSAR section 1.2.1.4, "Shielding Materials." Section 17.3.4.1 of this SER documents the staff's review of the structural properties of Metamic-HT, while the sections below give the staff's evaluation of the neutron absorption properties of Metamic-HT over the potential length of service of the MPCs.

In NUREG-2214, the staff evaluated the potential for long-term degradation to affect the criticality control performance of aluminum-based neutron absorber materials. The staff's evaluation concluded that the potential for boron depletion over a 60-year life would be negligible, or about 0.02 percent of the available boron-10 atoms. [

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Therefore, based on the prior staff evaluation of boron depletion and the applicant's testing for long-term radiation exposure, the staff finds the use of the Metamic-HT material for criticality control at the HI-STORE CIS Facility to be acceptable and that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 72.24(d), 10 CFR 72.120(d), and 10 CFR 72.124(b).

17.3.10 Reinforced Concrete Structures

17.3.10.1 Independent Spent Fuel Storage Installation Pad and Support Foundation Pad

In SAR section 5.3, "Reinforced Concrete Structures," and section 17.2.1.2, "Synopsis of Structural Materials," the applicant stated that all reinforced concrete load-bearing structures in the ISFSI are designed in accordance with ACI 318-05, "Building Code Requirements for Structural Concrete," issued in 2005. The applicant also stated that the concrete and its reinforcement are identical to those described in the drawings and section 3.3 of the HI-STORM UMAX FSAR. The compressive strength, density, and reinforcement requirements are described in HI-STORM UMAX FSAR table 2.3.2, "Design Data for HI-STORM UMAX ISFSI," (reproduced in HI-STORE SAR table 4.3.3). Finally, in SAR section 6.1, "Decay Heat Removal Systems," the applicant stated that the design-basis heat load and ambient conditions are lower than those for the UMAX system.

The staff reviewed the VVM drawings in HI-STORE SAR section 1.5 and verified that the drawings cite the concrete strength and reinforcement requirements in HI-STORM UMAX FSAR table 2.3.2. The staff also compared the canister enclosure container temperature at the HI-STORE site, as listed in SAR table 6.4.3, "Normal Long-Term Storage Temperatures for MPC-37 in HI-STORM UMAX at HI-STORE CIS," to that in the HI-STORM UMAX system FSAR table 4.4.2) and confirmed that the reinforced concrete structures at the HI-STORE CIS Facility are exposed to lower temperatures than those in the approved HI-STORM UMAX CoC. The staff also notes that the long-term canister enclosure container temperatures are lower than the 200°F (93°C) threshold cited in ACI 349, "Code Requirements for Nuclear Safety-Related Structures," which also does not require any tests to prove concrete elevated-temperature performance.

Regarding potential radiation effects on concrete, the staff notes that NUREG-2214 evaluated the effects of long-term neutron radiation exposure on concrete and determined that the accumulated neutron fluence over 100 years of storage in the most limiting cask location is three orders of magnitude lower than the levels reported to potentially degrade concrete. As a result, NUREG-2214 concluded that the radiation-induced degradation of concrete is not credible. In addition, the applicant stated that the radiation source term in the HI-STORE storage systems is less severe than would be expected in the HI-STORM UMAX systems at the original ISFSI sites, due to the more limiting allowable heat load of the HI-STAR 190 transportation casks that will bring the fuel to the HI-STORE CIS Facility.

As a result, because concrete exposure temperatures and radiation levels are below levels that are expected to degrade concrete over the proposed 40-year license, the staff find the design of the ISFSI pad and support foundation pad, which is consistent with the HI-STORM UMAX CoC, to be acceptable.

17.3.10.2 Cask Transfer Building Structures and Canister Transfer Facility Foundation

The CTB slab, walls, and roof and the CTF foundation are designed to the specifications in ACI 318-05, similar to the ISFSI pad design. The minimum concrete strength for the CTB floor slab is defined in SAR table 4.6.2, "Reference Design Data for the CTB Slab," SAR table 5.3.1, "Material Properties for CTB Slab & CTF Foundation," and the slab drawing in SAR section 1.5. The strength is identical to that of the ISFSI pad and support foundation pad. As stated in SAR section 5.3.3, "Canister Transfer Facility Foundation," and tables 4.6.2 and 5.3.1, the CTF foundation is designed to the same requirements as the CTB floor slab. The staff finds the concrete design codes to be acceptable because they are consistent with consensus national standards for structural concrete.

Regarding potential temperature effects on concrete, SAR table 6.4.5, "Maximum Component Temperatures and MPC Cavity Pressure for HI-STAR 190 in CTF Short-Term Operation," shows that the enclosure shell of the transportation cask may reach 160°C (336°F) while positioned inside the CTF. The staff notes that the exposure of concrete to this temperature can lead to a loss of concrete strength. Carette and Malhotra (1985) found that the compressive strength of concrete can drop by almost 30 percent after 4 months of continuous exposure near 150°C (302°F). In a response to a request for additional information (Holtec, 2019), the applicant stated that the temperature of the concrete beneath the transportation cask could reach as high as 150°C (302°F). The applicant cited its structural analyses that show the concrete beneath the transportation cask could reduce in strength by as much as 50 percent and still be capable of supporting the cask. The staff reviewed the applicant's structural analyses and verified that the potential decreases in concrete yield strength would not prevent the CTF foundation from supporting the transportation cask; therefore, the staff finds that the applicant adequately accounted for temperature exposure in the reinforced concrete design.

Based on the reviews above in SER sections 17.3.10.1 and 17.3.10.2, the staff determined that the information provided by the applicant on the performance of reinforced concrete components is sufficient to meet the regulatory requirements in 10 CFR 72.24(c)(3), 10 CFR 72.24(d), and 10 CFR 72.120(d).

17.3.11 Bolt Applications

In SAR section 17.6, "Bolts and Fasteners," the applicant stated that ITS bolting is present in the MPC lift attachment and in the connections that secure the transfer cask against tipover during MPC transfer in the VVM and CTF. The MPC lift attachment bolts connect to the MPC lid and allow the MPC to be raised and lowered during canister transfer operations. These bolts are constructed of ASME SB-637 Grade 718 nickel-based alloy (INCONEL® Alloy 718). The bolting that secures the transfer cask against tipover is constructed of ASME SA-193 Grade B7 alloy steel.

The staff notes that neither of the bolting applications discussed above would be expected to be subject to thermal expansion differences that could cause excessive loading. In the case of the

MPC attachment bolts, the Alloy 718 bolting attaches the ASTM A336-F6NM martensitic stainless steel lifting lug to the relatively hot MPC outer lid. Because Alloy 718 has a coefficient of thermal expansion slightly larger than that of stainless steel (Desai and Yo, 1978; Special Metals, 2007), no loading of the bolt is expected due to expansion of the lifting lug as it heats up. The alloy steel bolts that secure the transfer cask against tipover are not exposed to significant elevated temperatures, and they are also secured to similar materials (carbon steel). Because of the temporary nature of bolt service in the MPC lifting attachment and the transfer cask tipover anchor, neither creep nor corrosion is expected to impact the bolt functions.

The applicant addressed the fracture performance of the bolts by either specifying a material that is not subject to brittle fracture (nickel alloy MPC lift attachment) or requiring Charpy impact testing in accordance with the criteria in ASME B&PV Code, Section III, Division 1, Subsection NF, as specified in SAR table 17.4.1, “Fracture Toughness Test Requirements for HI-TRAC CS,” and SAR table 17.4.2, “Fracture Toughness Test Requirements for Cask Transfer Facility.”

Based on the staff’s verification that the bolting is not subject to substantial thermal expansion stresses, creep deformation, corrosion, or brittle fracture, the staff finds the applicant’s analysis of bolting performance to be acceptable and that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 72.24(d) and 10 CFR 72.122(a) and (b).

17.3.12 Seals

In SAR section 17.10, “Seals,” the applicant stated that the only seals present at the HI-STORE CIS Facility are associated with the VVM that was previously approved in the HI-STORM UMAX CoC. These seals are at the interface between the closure lid and the CEC and between the closure lid and the divider shell. The staff reviewed the VVM drawings in section 1.5, “Figures and Drawings,” of the HI-STORM UMAX SAR and notes that the VVM parts list defines the seals as not ITS. The staff also reviewed the HI-STORE SAR and verified that no ITS seals are used elsewhere in the HI-STORE CIS Facility.

17.3.13 Corrosion Resistance

In the materials evaluation discussion in SAR section 17.0, “Introduction,” the applicant stated that the environmental conditions at the HI-STORE CIS Facility are completely bounded by those evaluated for the HI-STORM FW and HI-STORM UMAX CoCs. Thus, the applicant stated that the use of materials that are identical to those in the HI-STORM FW and HI-STORM UMAX systems demonstrates their adequacy.

The staff evaluated whether the applicant accurately described the environment at the HI-STORE site and appropriately considered whether exposure of ITS SSCs to the environment could impact component functions over the 40-year license term. For this evaluation, the staff also reviewed the information in SAR chapter 18 and proprietary Holtec Report No. HI-2167378, Revision 5, “Aging Assessment and Management Program for HI-STORE CIS,” dated June 25, 2021, which includes an aging management review of all SSCs to identify credible aging-related degradation mechanisms.

Description of Site Environment

In SAR chapter 2, “Site Characteristics,” the applicant described the environmental conditions at the HI-STORE site as semiarid with low precipitation and low humidity. Precipitation measured 30 miles (48 kilometers) from the site at the Lea County Regional Airport averages 10.2 inches (26 centimeters) per year. Relative humidity ranges from 45 to 61 percent. The area around the site contains several playas, or transitory shallow lakes, that contain accumulations of halite (sodium chloride) and gypsum (calcium sulfate dehydrate). The surrounding area historically has been mined for potash. The staff notes that sylvinite, a mixture of sylvite (potassium chloride) and halite, is the typical potash ore mined in the Carlsbad Potash District in southeastern New Mexico (Barker and Austin, 1993).

The applicant estimated a primary water table depth of approximately 253 to 265 feet (77 to 81 meters) at the site, based on well drilling data. However, isolated pockets of saturated horizons were found near the surface, approximately 35 to 50 feet (11 to 15 meters) deep, in areas associated with the playas in the region surrounding the HI-STORE CIS Facility site. The applicant stated that the shallow ground water in the region was influenced by the brine discharges from potash refining or oil and gas production. In SAR section 2.5, “Subsurface Hydrology,” the applicant noted that no near-surface ground water was found in boreholes at the site. Consequently, the applicant stated that no impacts to the facility would be expected from near-surface ground water. Based on water table elevations in areas adjacent to the site and elevations of the Laguna Gatuna, the staff could not preclude the possibility of saturated lenses of ground water impacting the subgrade, as described in SER section 2.3.5. In SAR section 18.4, “Unique Aspects of the HI-STORE CIS With Nexus to Its AMP,” the applicant described the environment at the HI-STORE site as having a minuscule amount of salts and other airborne particulates that are known to be injurious to stainless steels.

In its review of the corrosion resistance of the HI-STORE CIS Facility components documented below, and the evaluation of maintenance activities in SER section 17.3.16, “Maintenance and Aging Management,” the staff considered the implications of a potential salt-bearing environment and the potential for groundwater to be present in the subgrade. Although the applicant stated that the salt presence was too low to be detrimental and that no impacts from near-surface groundwater were expected, the staff concluded that the uncertainties associated with groundwater and the airborne transport of area deposits warranted an assumption for conservatism that these environmental conditions could be present.

Ferritic Steel Components

Ferritic steels (e.g., carbon steels, low-alloy steels) are the primary materials of construction for all ITS site components other than the MPC and concrete pads. These components include the VVM CEC, HI-TRAC CS transfer cask, CTF, CTB, transport cask tilt frame, VCT, HI-PORT heavy haul trailer, and lifting components (CTB crane, cask lifting yokes, transport cask horizontal lift beam, MPC lift attachment, MPC lifting device extension, and transfer cask lift link). As stated in SAR tables 17.1.2, “Considerations Germane to the HI-STORM UMAX VVM Material Performance,” and 17.1.3, “Considerations Germane to Other SSC’s Material

Performance," service environments for these components include the ambient air (indoor and outdoor), the subgrade, and embedment in concrete.

<u>Atmosphere Exposure</u>

In SAR section 17.7, "Coating and Corrosion Mitigation," the applicant stated that all carbon steel surfaces of ITS SSCs exposed to the atmosphere or subgrade are coated to avoid the formation of corrosion products. Nevertheless, in SAR section 18.3, "Mechanisms for Aging of SSCs," the applicant stated that all painted carbon steel surfaces are potentially susceptible to general corrosion, and this is managed with a maintenance program. Holtec Report No. HI-2167378 cited localized corrosion mechanisms (pitting and crevice corrosion) as credible as well.

The staff notes that the applicant's evaluation of steel corrosion in the ambient environment is consistent with the guidance in NUREG-2214, which evaluated steel corrosion for service in storage up to 60 years. Although the SSCs are coated to prevent corrosion, the applicant proposed maintenance activities and AMPs (evaluated in SER section 17.3.16) that will inspect for all forms of corrosion. As a result, the staff finds the applicant's evaluation of steel corrosion to be acceptable.

<u>Subgrade Exposure (Vertical Ventilated Module Components)</u>

In SAR section 17.7, the applicant discussed the potential for corrosion of the buried VVM components. The SAR describes the corrosion preventive measures incorporated into the design of the VVM CEC, including encasing this carbon steel structure in self-hardening engineered subgrade (controlled low-strength material or "lean" concrete), limiting potential water ingress by the reinforced concrete enclosure wall and support foundation pad, and coating the CEC with surface preservatives for corrosion resistance. SAR table 17.2.1, "Additional CLSM Performance Properties," contains performance requirements for the controlled low-strength material, which include minimum electrical resistivity and pH to limit corrosion.

The staff notes that NUREG-2214 identifies general, pitting, crevice, galvanic, and microbiologically influenced corrosion as credible aging mechanisms for buried carbon steel components during storage from 20 to 60 years. The applicant did not identify specific corrosion mechanisms that could impact the buried steel components of the VVM. However, the VVM maintenance activities described in SAR section 10.3.4, "Maintenance Program for the HI-STORM UMAX VVM Systems," and the VVM AMP described in SAR section 18.7, "VVM Aging Management Program," include inspections of the internal surfaces of the VVM to identify corrosion. SER section 17.3.16 documents the staff's review of those activities, while SER section 17.3.14 describes the staff's review of the CEC coating. Because the VVM design includes the use of an engineered subgrade, a reinforced concrete enclosure wall and support foundation pad limiting potential water ingress, and coatings for corrosion prevention, and the applicant will use inspections to identify indications of corrosion, the staff finds that the applicant has appropriately evaluated the potential for corrosion of buried steel components.

Embedment in Concrete

In SAR section 18.8, “Reinforced Concrete Aging Management Program,” and Holtec Report No. HI-2167378, the applicant described activities to manage concrete degradation after 20 years of operation. The staff notes that the AMP includes visual inspections to identify evidence of embedded steel corrosion. SER section 17.3.16 documents the staff’s review of these concrete inspection and monitoring activities.

The staff notes that NUREG-2214 identifies corrosion of reinforcing steel as a credible aging mechanism in outdoor and ground water environments during storage from 20 to 60 years. Because the applicant has identified the need for inspections to identify indications of corrosion of steel embedded in concrete, and because of the staff’s review in SER section 17.3.16, the staff finds that the applicant has appropriately evaluated for the potential for corrosion of embedded steel.

Stainless Steel Components

As discussed in SER section 17.3.4.1, the MPC confinement boundary is constructed of “Alloy X” austenitic stainless steels, which include ASME grades SA-316, SA-316LN, SA-304, and SA-304LN. The MPC has several additional stainless-steel components, such as the vent and drain blocks and ports, lifting lugs, and port covers. The stainless steel MPC subcomponents are exposed to either the MPC internal helium environment or the external atmosphere, which is sheltered within the VVM from direct contact with precipitation.

In SAR section 18.3, the applicant stated that stress corrosion cracking is only a remote possibility for the stainless-steel components that are exposed to the ambient environment because the halide content in the air at the HI-STORE site is negligible. However, because the applicant stated that stress corrosion cracking could not be ruled out entirely, the applicant proposed inspections to identify stress corrosion cracking for MPCs with more than 20 years of service. SER section 17.3.16 documents the staff’s review of these MPC inspection activities.

The staff notes that NUREG-2214 identifies stress corrosion cracking of austenitic stainless steels as a credible aging mechanism in outdoor and sheltered environments within overpacks during storage from 20 to 60 years. Because the applicant has identified the need for inspections to identify indications of corrosion and cracking of the MPC confinement boundary, the staff finds that the applicant has appropriately evaluated the potential for corrosion-related aging of stainless steels.

Based on the reviews above, the staff determined that the information provided by the applicant on the corrosion resistance of the ITS SSCs is sufficient to meet the regulatory requirements in 10 CFR 72.120(d) and 10 CFR 72.122(b).

17.3.14 Protective Coatings

In SAR section 17.7, the applicant stated that all carbon steel surfaces of ITS SSCs exposed to the atmosphere or subgrade are coated with a zinc-rich preservative to avoid corrosion. The SAR incorporates by reference the coatings given in HI-STORM FW FSAR section 8.7.2, “Acceptable Coatings,” and appendix 8A, which include Carboguard 890 (cycloaliphatic amine epoxy), Thermaline 450 (amine-cured novolac expoxy), CarboZinc 11 (solvent-based inorganic zinc), and Sherman Williams Zinc Clad II coatings (inorganic ethyl silicate, zinc-rich coating). The coating on the interior surface of the VVM divider shell has an additional requirement for emissivity, which is incorporated by reference from HI-STORM UMAX FSAR table 4.2.4, “Summary of Materials Surface Emissivity Data.”

The CEC exterior coating has unique requirements. As described in SAR section 17.7.1, “Exterior Coating,” the CEC exterior is coated with a preservative designed for below-grade or immersion service. This coating is treated as a Service Level II coating (as defined in ASTM D5144, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” issued 2008), which is subject to the quality assurance guidelines in ASTM D3843-00, “Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities.” The SAR describes the acceptance criteria for this coating selection, such as suitability for immersion/below-grade service, compatibility with cathodic protection systems, and compatibility with concrete encasement. The staff notes that the CEC exterior coating criteria are identical to those in the HI-STORM UMAX FSAR.

The staff reviewed the coating criteria and determined that they are acceptable because they are consistent with the coatings of the previously approved HI-STORM UMAX and HI-STORM FW systems. As such, they provide for the adequate control of chemistry and properties to ensure that the coating can protect the steel from corrosion. Therefore, the staff determined that the information provided by the applicant on the use of coatings to mitigate corrosion of the ITS SSCs is sufficient to meet the regulatory requirements in 10 CFR 72.24(c)(4), 10 CFR 72.120(d), and 10 CFR 72.122(b). To verify that the coatings remain effective in preventing corrosion, the applicant proposed maintenance activities and AMPs. SER section 17.3.16 contains the staff’s evaluation of those activities.

17.3.15 Adverse Content Reactions

The staff reviewed the SAR to verify that the SNF contents are stable, such that there will be no flammable or explosive reactions during fuel loading or during the storage period. The staff has previously evaluated the potential for adverse content reactions with the MPC-37 and MPC-89 canisters in its reviews of the HI-STORM FW and HI-STORM UMAX systems. The staff notes that zirconium cladding materials and stainless steel and zirconium noncladding hardware components are considered to be compatible with the fuel pool and dry storage environment, in accordance with the guidance in section 8.5.13, “Content Reactions,” of NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities,” issued April 2020. The MPC lid welding procedures in HI-STORM FW SAR section 9.2.4, “MPC Closure,” also account for the potential presence of hydrogen gas due to the aqueous corrosion of contents (including the aluminum fuel basket) or coatings by flooding the area between the MPC lid and the submerged fuel assemblies with inert gas and monitoring for combustible gases. As a result, the staff finds that the materials and processes are appropriate to avoid adverse content reactions because the stainless steel and zirconium contents are compatible with the wet and dry MPC environments. Also, any potential presence of flammable gases due to corrosion in the pool water during fuel loading at the original MPC ISFSI site is addressed by monitoring and inerting the internal MPC environment during lid welding. Therefore, the staff determined that the information provided by the applicant to address potential adverse content reactions is sufficient to meet the regulatory requirements in 10 CFR 72.120(d).

17.3.16 Maintenance and Aging Degradation

In SAR section 18.14, “Timing of Aging Management Implementation,” the applicant stated that SSCs will be maintained by a combination of two site programs:

(1) Maintenance (SAR chapter 10). Maintenance inspections and tests will be performed on all equipment constructed exclusively for the HI-STORE CIS Facility, and these activities begin at the start of operation and continue throughout the term of the license.

(2) Aging management (SAR chapter 18): After any SSC reaches 20 years of service (including the prior time in storage of MPCs at their original ISFSI sites), additional aging management activities will be performed. SAR chapter 18 summarizes the aging management programs (AMPs), which are described in greater detail in proprietary Holtec Report No. HI-2167378.

Additionally, the applicant included a requirement for the AMP in Technical Specification 5.5.4, “Aging Management Program.”

As described in SAR section 18.14, “Timing of Aging Management Implementation,” after 20 years of service, SSCs are managed by a combination of maintenance activities and AMPs. AMPs may take credit for maintenance inspections if (1) the maintenance activity takes place within 1 year of the scheduled AMP activity, and (2) the maintenance activity uses AMP acceptance criteria for the condition of the SSCs.

The staff notes that the applicant used the term “aging management,” although that phrase is specifically related to regulatory requirements for license renewal (i.e., 10 CFR 72.42, “Duration of license; renewal”), rather than the initial issuance of a site license to store SNF. Notwithstanding the applicant’s terminology, the staff considered the totality of the applicant’s proposed inspections and monitoring activities in SAR chapters 10 and 18 to evaluate whether the applicant established activities to effectively maintain the functions of the ITS SSCs during the proposed 40-year facility license, in accordance with 10 CFR 72.120(a).

The staff’s evaluations below considered the guidance in NUREG-2214, which assesses aging-related degradation mechanisms for components in service up to 60 years. NUREG-2214 also includes example AMPs that the staff regards as generally acceptable for managing the effects of aging during extended storage. While NUREG-2214 specifically addresses license renewal requirements to extend service from 20 to 60 years, the staff considers the technical guidance to be generally relevant to an evaluation of monitoring and inspection activities for the proposed 40-year HI-STORE CIS Facility license as well.

Vertical Ventilated Module (Cavity Enclosure Container, Lid, and Divider Shell)

In SAR section 10.3.4 and table 10.3.1, the applicant stated that annual visual inspections of accessible areas will be performed on every loaded VVM to verify that vent screen coatings are intact and that visible exterior surfaces are free of significant corrosion.

The applicant also stated that, after 20 years of service, the Vertical Ventilated Module AMP will use condition monitoring to inspect for corrosion and integrity of the CEC, lid, and divider shell, as set forth in the maintenance program in SAR chapter 10. As described in SAR section 18.7 and table 18.6.1, “Periodic Inspection Frequency of HI-STORE CIS ISFSI Components,” the VVM aging management inspections will be performed every 5 years on each VVM that houses an MPC that undergoes an aging management inspection, as described in the staff’s review of the MPC maintenance activities below. As one MPC will be inspected from each originating reactor ISFSI site, the number of VVMs inspected will equal the number of sites that send MPCs to the HI-STORE CIS Facility.

The staff notes that the applicant’s proposed inspection approach is consistent with the NRC guidance in NUREG-2214 for managing the aging of metallic overpack components for storage terms up to 60 years. As a result, the staff finds the applicant’s activities to maintain the VVMs to be acceptable because the use of periodic remote visual inspections to monitor for corrosion of the internal surfaces of a sample of VVMs on the site is considered capable of ensuring that the functions of the VVMs are maintained.

Independent Spent Fuel Storage Installation Pad, Support Foundation Pad, and Cask Transfer Building Slab

In SAR table 10.3.1, the applicant stated that the maintenance program will include an annual inspection of the ISFSI pad to ensure it is free of cracks and that subgrade settlement is minimal. Every 5 years, the applicant will confirm that VVM settlement is within the range of the design basis. In addition, the CTF floor slab (i.e., the CTB slab) will be inspected annually for concrete degradation.

In SAR section 18.8, the applicant stated that the Reinforced Concrete AMP will include annual visual inspections to inspect for cracking, loss of material, permeability, and integrity of the ISFSI pad, support foundation pad, and CTB slab after 20 years of service.

The staff reviewed the applicant’s activities to maintain the reinforced concrete structures and finds them to be acceptable because, consistent with the NRC guidance in NUREG-2214, periodic visual inspections are capable of assessing the condition of reinforced concrete and ensuring the concrete’s structural performance is maintained. The staff notes that the applicant proposed to inspect the concrete at a greater frequency (annually) than suggested by the guidance in NUREG-2214 and the relevant standard, ACI 349.3R, “Evaluation of Existing Nuclear Safety-Related Concrete Structures” (every 5 years).

Multipurpose Canisters

In SAR section 10.3.5, “Maintenance Program for the Canister,” the applicant stated that the stainless steel MPCs do not require inservice maintenance, absent a disruptive occurrence such as flooding contamination. However, as described below, the SAR includes requirements to inspect MPCs before shipment to the HI-STORE CIS Facility site, leak-test the MPCs when they are received, and continue to inspect a sample of MPCs during storage at the site under an AMP.

In SAR section 10.3.3.1, “Receipt and Inspection of Transportation Cask and Canister,” the applicant stated that all MPCs being shipped to the HI-STORE CIS Facility must first be evaluated based on HI-STAR 190 transportation package SAR table 8.A.1, “MPC Transportability Checklist.” The staff notes that the cited checklist includes sampling-based inspections to confirm the integrity of the MPC containment boundaries. As described in HI-STAR 190 SAR section 8.1.8, “MPC Enclosure Vessel Shell Surface Defect Inspection,” MPCs stored for more than 5 years before shipment must first undergo eddy current examinations to verify that the MPCs are free of flaws that could develop into cracks during hypothetical accident conditions of transport. The inspection program does not require the inspection of all MPCs, but rather a sample of the MPC population at each originating ISFSI site will be inspected to provide reasonable assurance that the containment boundary of all MPCs will be maintained as they are transported. The applicant also clarified that the inspection program will be applied to all MPCs containing fuel of all levels of burnup.

SAR section 10.3.3.1 also states that, upon arrival at the HI-STORE CIS Facility, each MPC is leak tested in accordance with ANSI N14.5, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment,” to verify confinement performance before being placed into storage.

Finally, in SAR section 18.5, “MPC Aging Management Program,” the applicant stated that the MPCs need monitoring or inspections while in storage at the HI-STORE CIS Facility to detect potential chloride-induced stress corrosion cracking (CISCC). For those canisters in service for more than 20 years (including service time at the original ISFSI), the MPC AMP includes monitoring the exterior of at least one MPC from each original ISFSI site every 5 years, using visual inspections to identify signs of degradation. All accessible weld areas will be inspected for signs of corrosion pits, stress corrosion cracking, etching, deposits, and colored corrosion products. In order to prioritize the MPCs for inspection, the applicant stated that the canisters are ranked in accordance with Electric Power Research Institute (EPRI) Report No. 3002005371, “Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems,” issued September 2015. The canister ranking criteria include, but are not limited to, anticipated level of chloride accumulation, material alloy susceptibility, and decay heat. Suspected areas of localized corrosion are assessed with surface or volumetric examinations.

The Canister AMP also includes coupon testing as a defense-in-depth strategy to provide an early warning of the onset of CISCC. Prestressed U-bend coupons will be prepared in accordance with ASTM G30, “Standard Practice for Making and Using U-Bend Stress-Corrosion

Test Specimens," and will be placed in the inlet region of select VVMs that hold canisters in service for more than 20 years (including service time at the original ISFSI). The coupons will be monitored every 5 years with optical microscopy and dye penetrant testing. The applicant stated that, if the coupon testing detects a defect or anomaly in the coupon, the external surface of a representative canister may be examined with eddy current testing.

The staff reviewed the applicant's MPC inspection and monitoring activities and finds them to be acceptable because the combination of preshipment inspections, receipt leak testing, and ongoing aging management inspections are considered adequate to ensure that the MPC confinement boundary will be maintained. The staff notes that the aging management inspections are consistent with the NRC guidance in NUREG-2214, as the periodic remote visual inspections for corrosion, with follow-up surface or volumetric examinations in suspected areas of corrosion, are capable of identifying the presence of CISCC and ensuring confinement is maintained. The staff also notes that the applicant's proposal to inspect one canister from each originating reactor ISFSI site every 5 years is consistent with the guidance in NUREG-2214 for managing the aging of MPCs in service up to 60 years.

<u>HI-TRAC CS Transfer Cask</u>

The applicant provided the maintenance activities for the transfer cask in SAR section 10.3.4.1, "Structural Capacity Verification," and table 10.3.1. The SAR states that, before each handling campaign, the surface coatings of the interior and exterior surfaces of the cask will be verified to be intact, and the closure lid lift lugs, trunnions, bottom lid bolts, bolt holes, shield gates, and tie down studs will be visually inspected for degradation. In addition, the upper trunnions will be tested and inspected in accordance with ANSI N14.6 to identify cracks and deformation. ANSI N14.6 requires either annual testing to 150 percent of the maximum service load or inspection (visual and penetrant) to verify the capability of lifting devices.

In SAR section 18.6, "HI-TRAC CS Transfer Cask Aging Management Program," the applicant stated that the HI-TRAC CS AMP will use visual inspections to detect degradation of external surfaces and trunnions after the transfer cask reaches 20 years of service. Painted surfaces (including the inside surface of the cask cavity) will be inspected for corrosion and paint integrity, all surfaces will be inspected for damage, and trunnions will be inspected for deformation, cracks, damage, corrosion, and galling. These inspections will be performed before use and once every year while the transfer cask is in use.

The staff reviewed the applicant's activities to maintain the transfer cask and finds them to be acceptable because, consistent with the guidance in NUREG-2214, periodic visual inspections are capable of assessing coating integrity, the presence of corrosion, and the general condition of the transfer cask components. In addition, annual testing or inspections of the upper trunnions in accordance with ANSI N14.6 are capable of verifying the continued load-bearing capacity of lifting devices.

Special Lifting Devices

In SAR section 10.3.6, “Maintenance Programs for ITS Lifting and Handling Equipment, Including VCT,” and table 10.3.1, the applicant stated that the maintenance of the special lifting devices designed to ANSI N14.6 will be in accordance with the requirements of that standard. The staff notes that section 6.3 of ANSI N14.6 describes the tests that must be performed to verify continued compliance with that standard, including either annual load testing or nondestructive examinations (e.g., visual and dye penetrant/magnetic particle). In addition, the standard includes visual examinations for damage and deformation by maintenance and operating personnel at shorter intervals.

In SAR section 18.10, “Lifting Device Aging Management Program,” the applicant stated that the Lifting Device AMP uses condition monitoring to manage the effects of aging for special lifting devices after 20 years of service. The program includes visual inspections for corrosion and damage before each use and once every year while in use for the MPC lift attachment, MPC lifting device extension, HI-TRAC CS lift yoke, HI-TRAC CS lift link, transport task lift yoke, and transport cask horizontal lift beam.

The staff reviewed the applicant’s activities to maintain the special lifting devices and finds them to be acceptable because they are consistent with the applicable industry consensus standard for testing and inspections, and they are considered to be capable of verifying the continued load-bearing capacity of special lifting devices throughout the 40-year license term.

Cask Transfer Components (Vertical Cask Transporter, Transport Cask Tilt Frame, Canister Transfer Facility)

In SAR section 10.3.6, the applicant stated that the maintenance of the lifting and handling equipment designed to ASME B&PV Code, Section III, Subsection NF, includes functional testing before use and visual inspections for degradation and damage before each cask transfer. The staff notes that this includes portions of the VCT, the transport cask tilt frame, and the metallic subcomponents of the CTF. In addition, the transport cask tilt frame is visually inspected annually for corrosion and general condition, and the CTF floor slab is visually inspected annually for concrete degradation.

In SAR section 18.10, section 18.11, “Tilt Frame Aging Management Program,” and section 18.12, “CTF Aging Management Program,” the applicant stated that these AMPs use condition monitoring to manage the effects of aging for the VCT, transport cask tilt frame, and CTF, respectively, after those components reach 20 years of service. Each of these programs includes visual inspections to evaluate for corrosion, component integrity, and damage. The VCT and tilt frame are inspected before each use and once every year while in use; the CTF is inspected every 5 years.

The staff reviewed the applicant’s activities to maintain the cask transfer components and finds them to be acceptable because the periodic visual inspections are capable of assessing the presence of corrosion and general condition of the components, such that the structural function can be adequately maintained. The staff also notes that the use of periodic general visual inspections for corrosion of metallic components is consistent with the guidance in NUREG-2214.

Cask Transfer Building Crane

In SAR section 10.3.7, “Maintenance Programs for ITS Crane Systems,” and table 10.3.1, the applicant stated that the maintenance, inspection, and testing of crane systems designed to ASME NOG-1 shall be performed in accordance with the manufacturer’s recommendations and the requirements of ASME B30.2, “Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist),” issued 1976. The staff notes that ASME B30.2 establishes inspection frequencies based on the severity of service, as defined by the number and magnitude of lifts. Visual inspections are performed before use for general functionality, monthly during normal service for proper operation, and annually during normal service to identify corrosion, deformation, cracking, and wear.

In SAR section 18.10, the applicant stated that the Lifting Device AMP uses condition monitoring to manage the effects of aging for the CTB crane after it reaches 20 years of service. The program includes visual inspections for corrosion, component integrity, and damage before each use and once every year while in use.

The staff reviewed the applicant’s activities to maintain the crane and finds them to be acceptable because the inspections in accordance with ASME B30.2 are consistent with the consensus industry guidance on crane maintenance. The periodic visual inspections identify the potential degradation mechanisms before a loss of function.

Spent Nuclear Fuel Assemblies

SER section 17.3.17 documents the staff’s review of the maintenance of spent fuel assemblies.

Assessment of Aging Management Effectiveness (Tollgates)

In SAR section 18.13, “Learning Based AMP,” and table 18.13.1, “Tollgate Assessments for HI-STORE ISFSI,” the applicant described its proposed “tollgate” process for formally assessing aggregated aging management feedback at specific points in time and performing a safety assessment that confirms the safe storage of SNF.

The applicant’s tollgate assessment includes an evaluation of industry and site-specific operating experience to determine whether adjustments should be made to AMPs (e.g., modifying inspection frequencies, adding mitigative activities). The tollgate process begins the year the first MPC completes 20 years of service (including service at the prior reactor ISFSI site) or the year an MPC is first placed at the HI-STORE CIS Facility if that MPC already exceeds 20 years of service. Tollgate assessments of the efficacy of MPC aging management continue every 5 years thereafter.

The staff notes that the applicant’s tollgate methodology is consistent with NRC Regulatory Guide 3.76, Revision 0, “Implementation of Aging Management Requirements for Spent Fuel Storage Renewals,” issued July 2021. Regulatory Guide 3.76 endorses Nuclear Energy Institute

(NEI) 14-03, Revision 2, “Format, Content and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management,” issued December 2016, which recommends a periodic review of inspection results, operating experience, and other available information to adjust aging management activities to ensure their continued effectiveness. As a result, the staff finds that the implementation of periodic tollgate assessments, in addition to other periodic operating experience reviews consistent with the site quality assurance program, provides an acceptable approach to ensure that the condition of the HI-STORE CIS Facility SSCs will be adequately maintained throughout the 40-year license term.

Based on the reviews above, the staff determined that the information provided by the applicant on the activities to effectively maintain the functions of the ITS SSCs is sufficient to meet the regulatory requirements in 10 CFR 72.120(a).

17.3.17 Spent Nuclear Fuel

In SAR section 17.12, “Fuel Cladding Integrity,” the applicant stated that the evaluation of fuel cladding integrity is incorporated by reference from HI-STORM FW SAR section 8.13, “Fuel Cladding Integrity.” In the HI-STORM FW SAR, the applicant stated that the functions of the fuel cladding are fulfilled by holding cladding temperatures to the limits specified in SFST-ISG-11, Revision 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” dated November 17, 2003; limiting thermal cycling during drying operations; and maintaining an inert environment within the cask by MPC drying and helium backfilling, or by forced helium dehydration.

In SAR section 18.9, “HBF Aging Management Program,” the applicant stated that it does not expect that there will be aging effects that could lead to fuel reconfiguration, provided that drying practices are consistent with the recommendations in SFST-ISG-11. Also, in SAR section 18.3 the applicant provided an analysis to demonstrate that the fuel cladding creep rate in the HI-STORM UMAX system would be expected to be a small fraction of the creep rate that may be anticipated for fuel that reached the SFST-ISG-11 cladding temperature limit (400°C (752°F)). However, the applicant proposed to use the High Burnup Fuel AMP to monitor and assess data to verify the performance of high burnup fuel during storage beyond 20 years. The program relies on the EPRI/U.S. Department of Energy High Burnup Dry Storage Cask Research and Development Project (HDRP) to provide data on high burnup fuel performance (EPRI, 2014). The staff notes that the HDRP has placed into service a demonstration cask containing high burnup fuel. After at least 10 years of storage, the cask is planned be opened to allow the examination of the fuel and the characterization of the internal cask environment.

In its prior reviews of the HI-STORM UMAX and HI-STORM FW CoCs, the staff concluded that the spent fuel in the MPCs will remain in the configuration analyzed in the SAR over the 20-year term of those licenses, as the applicant employed drying and inerting practices that will protect the fuel from degradation and potential reconfiguration under normal and accident conditions.

For high burnup fuel stored longer than 20 years, the staff notes that NUREG-2214 concluded that, while hydride reorientation and creep of cladding may occur these aging mechanisms are not expected to result in cladding failures and reconfiguration of the fuel. However, since the conclusions in NUREG-2214 were based on relatively short-term testing, the guidance recommends a surveillance program to confirm the expected fuel conditions. The staff finds the applicant's use of the High Burnup Fuel AMP to assess potential fuel assembly degradation to be acceptable because, consistent with the guidance in NUREG-2214, the activities to evaluate the condition of the fuel and internal environment within the HDRP demonstration cask are considered capable of verifying the absence of aging-related issues associated with the extended storage of high burnup SNF. Therefore, the staff determined that the information provided by the applicant on the performance of the fuel cladding is sufficient to meet the regulatory requirements in 10 CFR 72.120(a) and 10 CFR 72.122(h)(1). The staff's evaluation of the storage facility design to support ready retrieval of the SNF is documented in SER section 4.3.3.8.

17.4 Evaluation Findings

The staff makes the following evaluation findings:

- The applicant has met the requirements in 10 CFR 72.24(c)(3) and 10 CFR 72.120(a). The applicant described the materials used for ITS SSCs in sufficient detail to support a safety finding.

- The applicant has met the requirements in 10 CFR 72.24(d) and 10 CFR 72.128(a). The properties of the materials in the storage facility design have been demonstrated to support the safe storage and handling of storage systems for SNF for the life of the facility under normal, off-normal, and accident conditions.

- The applicant has met the requirements in 10 CFR 72.124(b). Neutron absorbing materials are demonstrated to effectively control criticality without significant degradation over the life of the facility.

- The applicant has met the requirements in 10 CFR 72.120(d), 10 CFR 72.122(b)(1), and 10 CFR 72.124(b). Materials and storage contents are compatible with their operating environment such that there will be no adverse degradation or significant chemical or other reactions.

- The applicant has met the requirements in 10 CFR 72.122(h)(1). The SNF cladding has been demonstrated to be adequately protected against gross ruptures, or the fuel has been demonstrated to be otherwise confined.

- The applicant has met the requirements in 10 CFR 72.24(c)(4) and 10 CFR 72.122(a). The use of codes and standards, quality assurance programs, and control of special processes is demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

17.5 References

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18 FINANCIAL QUALIFICATIONS

By letter dated March 30, 2017, Holtec International (Holtec or the applicant) submitted a license application for a consolidated interim storage (CIS) facility for spent nuclear fuel (SNF), to be located at an away-from-reactor site in Lea County, New Mexico. Holtec submitted the application pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." Holtec is requesting authorization to construct and operate the consolidated interim storage facility (CISF), which it refers to as the HI-STORE CIS Facility, for an initial 500 canisters of SNF, containing up to 8,680 metric tons of uranium (MTU), and a license duration of 40 years.

In this safety evaluation report (SER) chapter, the staff has summarized its analysis as to the adequacy of Holtec's financial qualifications, including its plans for meeting the decommissioning financial assurance requirements found in 10 CFR Part 72 and discussed in greater detail in SER section 18.2. SER chapter 13 gives the staff's evaluation of Holtec's plans for decommissioning not related to decommissioning financial assurance.

18.1 Scope of Review

The staff reviewed the applicant's description of its financial qualifications to demonstrate its current and continuing access to the financial resources necessary to engage in the proposed activity. As part of this review, the staff evaluated the estimated construction and operations costs provided by the applicant for reasonableness and evaluated the applicant's qualifications in order to assess its ability to finance the construction and operation of the facility.

For decommissioning, the staff reviewed information provided by the applicant to determine how decontamination and decommissioning of the facility will be addressed. As part of this review, the staff evaluated the applicant's decommissioning cost estimate (DCE) for reasonableness and evaluated the applicant's method of assuring funds for independent spent fuel storage installation (ISFSI) decommissioning to determine whether the U.S. Nuclear Regulatory Commission's (NRC's) requirements for decommissioning financial assurance are met.

The staff also reviewed the applicant's plans for providing site specific insurance and indemnity coverage.

18.2 Regulatory Requirements

The following sections of Title 10 of the *Code of Federal Regulations* contain the NRC requirements relevant to the staff's evaluation of the applicant's financial qualifications and adherence to decommissioning financial assurance criteria for the proposed facility:

- 10 CFR 72.22, "Contents of application: General and financial information"
- 10 CFR 72.30, "Financial assurance and recordkeeping for decommissioning"

With respect to the NRC's financial qualifications requirements for a CISF for SNF, in accordance with 10 CFR 72.22(e), an applicant for an ISFSI license, except for the U.S.

Department of Energy (DOE), must submit sufficient information to demonstrate its financial qualifications to carry out the activities for which the license is sought. The information must show that the applicant either possesses the necessary funds, or that the applicant has reasonable assurance of obtaining the necessary funds, or that by a combination of the two, the applicant will have the necessary funds available to cover the following:

- estimated construction costs
- estimated operating costs over the planned life of the ISFSI
- estimated decommissioning costs, and the necessary financial arrangements to provide reasonable assurance before licensing that decommissioning will be carried out after the removal of spent fuel, high-level radioactive waste, and reactor-related Greater than Class C (GTCC) waste from storage

With respect to the NRC's decommissioning planning and financial assurance requirements for an SNF CISF, in accordance with 10 CFR 72.30(a), an applicant for an ISFSI must provide a proposed decommissioning plan that describes its intended practices and procedures for decontamination and decommissioning of the site. Further, under 10 CFR 72.30(b), an applicant must submit a decommissioning funding plan (DFP) containing information on how reasonable assurance will be provided that funds will be available to decommission the ISFSI, a detailed cost estimate for decommissioning (the DCE), key assumptions used in the DCE, and a description of the method of assuring funds for decommissioning, including a means of adjusting cost estimates and associated funding levels periodically over the life of the ISFSI.

An applicant must address provisions for site-specific insurance, as needed, on a case-specific basis. While NRC application requirements under 10 CFR Part 72 do not detail specific financial protection amounts, the Commission determined in CLI-00-13, for the Private Fuel Storage licensing proceeding, that 10 CFR Part 72 affords the NRC flexibility in reviewing whether an application provides adequate financial assurance, according to the risks of the proposed activity. As such, the NRC has previously exercised its discretion on a case-specific basis to condition the approval of an SNF storage license on the provision of onsite and offsite financial protection by the applicant.

18.3 Staff Review and Evaluation

The staff reviewed and evaluated the applicant's financial qualifications and decommissioning plan included in its March 30, 2017, license application and in documents cited in or attached to the application, along with the updated DCE and DFP submitted on November 16, 2022. The staff also reviewed and evaluated additional information provided in the applicant's October 9, 2020, response to the staff's requests for additional information, as the staff needed additional information to fully assess the applicant's financial qualifications and to ensure that the NRC's decommissioning funding assurance requirements are met.

In its review and evaluation of application information, the staff also applied guidance provided in the following two NRC documents:

- NUREG-1757, “Consolidated Decommissioning Guidance,” Volume 3, Revision 1, “Financial Assurance, Recordkeeping, and Timeliness,” issued February 2012

- NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” issued February 2000

In its evaluation of the applicant’s DFP, the staff conducted its review consistent with the guidance in NUREG-1757, Volume 3, Revision 1, which notes the requirement under 10 CFR 72.30(b)(2) that the DFP must contain a detailed DCE giving an amount reflecting (1) the cost of an independent contractor to perform all decommissioning activities, (2) an adequate contingency factor, and (3) the cost of meeting the unrestricted use criteria in 10 CFR 20.1402, “Radiological criteria for unrestricted use” (or the cost of meeting the restricted use criteria in 10 CFR 20.1403, “Criteria for license termination under restricted conditions,” provided the licensee can demonstrate its ability to meet these criteria).

The licensee’s DFP must also identify and justify using the key assumptions contained in the DCE, as required by 10 CFR 72.30(b)(3).

The DFP must describe the method of assuring funds for ISFSI decommissioning, including a means for adjusting cost estimates and associated funding levels periodically over the life of the ISFSI, as required by 10 CFR 72.30(b)(4).

The DFP must specify the volume of onsite subsurface material containing residual radioactivity that will require remediation to meet the criteria for license termination and include a certification that financial assurance for ISFSI decommissioning has been provided in the amount of the DCE, as required by 10 CFR 72.30(b)(5) and (6).

18.3.1 Financial Qualifications for Independent Spent Fuel Storage Installations

The regulation at 10 CFR 72.22(e) requires, in part, that the applicant must provide information on the cost of facility construction and operation, as well as information showing it meets the financial qualifications requirements for funding construction and operation at the site. Specifically, the requirement states the applicant is to provide “information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the license is sought. …The information must show that the applicant either possesses the necessary funds, or that the applicant has reasonable assurance of obtaining the necessary funds or that by a combination of the two, the applicant will have the necessary funds available to cover…” estimated construction costs and estimated operating costs over the planned life of the ISFSI.

Moreover, 10 CFR 72.40(a) states that “…the Commission will issue a license under this part upon a determination that the application for a license meets the standards and requirements of the Act and the regulations of the Commission, and upon finding that…the applicant for an ISFSI or MRS [monitored retrievable storage installation] is financially qualified to engage in the proposed activities in accordance with the regulations in this part.”

The staff reviewed the reasonableness of the estimated construction and operations costs provided by the applicant and evaluated the applicant's qualifications as to its ability to fund facility construction and operation. The following sections present the staff's analyses.

18.3.1.1 Estimated Construction Costs

As described in chapter 1, "Introduction," of the applicant's safety analysis report, Holtec plans up to 20 stages of construction of the facility, increasing storage capacity in stages to correspond with anticipated demand from U.S. nuclear power producers and the U.S. Government. The first planned stage involves the containment of 500 storage canisters holding up to 8,680 metric tons of uranium (MTU) of SNF. The applicant indicated that following construction of the first stage, 19 subsequent expansion phases will be constructed over the course of 20 years. Holtec indicated that it will seek authorization to possess and store additional SNF waste for each of the 19 subsequent expansion phases by requesting amendments to its license. In all, each of the 20 stages will contain 500 storage canisters, with a total capacity of 10,000 canisters holding 173,600 MTU. As stated by the applicant in the DFP, the HI-STORE CIS Facility will rely on the HI-STORM UMAX canister storage system, which stores SNF in a subterranean configuration.

In Holtec Report No. HI-2177593, Revision 2, "HI-STORE CIS Facility Financial Assurance & Project Life Cycle Cost Estimates" (proprietary), dated August 9, 2022, the applicant estimated construction costs of $223,300,000 for the facility. The cost estimate covers the first phase of the facility as proposed by the applicant. The facility may be expanded by future license amendments to include 19 additional phases. The first phase will have a capacity of 500 HI-STORM UMAX canisters for the storage of up to 8,680 MTU of SNF. Holtec identified key construction cost elements as follows:

- land acquisition: $1,000,000
- security building: $35,410,000
- administrative building: $4,090,000
- storage building: $2,510,000
- cask transfer building: $25,900,000
- batch plant: $1,010,000
- rail lines: $12,780,000
- site work: $51,200,000
- UMAX manufacturing: $89,400,000

The staff found Holtec's detailed estimate to be thorough, as described in the summary below. In order to evaluate the reasonableness of this estimate, the staff reviewed estimated ISFSI construction costs presented in a 2009 study performed by the Electric Power Research Institute (EPRI) for a generic away-from-reactor ISFSI (GISF). The study, "Cost Estimate for an Away-from-Reactor Generic Interim Storage Facility (GISF) for Spent Nuclear Fuel," issued in 2009, provides detailed evaluations of capital costs required to construct a GISF, including those for establishing the necessary transportation infrastructure required to serve the GISF; GISF buildings, equipment, and infrastructure; and equipment needed to load and transport

SNF from nuclear power plants. The costs presented are representative of capital costs associated with designing, engineering, and licensing a GISF.

Based on the study, EPRI estimated capital expenditure costs of $270 million (2009 dollars) for a facility designed to store 20,000 MTU. In order to scale this estimate to reflect Holtec's plan for an initial license to store 8,680 MTU, the staff reduced the EPRI estimate by 50 percent and applied a 15 percent increase to that cost estimate to account for inflation between January 2009 (the year of the report's publication) and December 2016 (the NRC received Holtec's application in March 2017). Based on this information, the staff concluded that the estimated cost to construct a facility of the size for which Holtec is requesting approval would equal approximately $156 million (2017 dollars).

The staff also evaluated data from the November 2009 U.S. Government Accountability Office (GAO) report, "Nuclear Waste Management: Key Attributes, Challenges, and Costs for the Yucca Mountain Repository and Two Potential Alternatives." In that report, the GAO cited[3] estimates for "centralized facility licensing and construction" component costs and the cost of annual operations and maintenance for a centralized storage facility. The GAO estimated licensing and construction costs of $218 million (2009 dollars) for a facility designed to store 70,000 MTU. In order to scale this estimate to reflect Holtec's plan for an initial license to store 8,680 MTU, the staff reduced the GAO estimate by 85 percent and applied a 15 percent increase to that cost estimate to account for inflation between January 2009 (the year of the report's publication) and December 2016. Based on this information, the staff concluded that the estimated cost to construct a facility of the size for which Holtec is requesting approval would equal approximately $38 million (2017 dollars).

As presented in its application, the Holtec construction cost estimate of $182,849,000 is thorough and, after scaling to reflect Holtec's plans for a facility to store 8,680 MTU, lies above the two cost estimates in the EPRI and GAO reports. Accordingly, the staff concludes that Holtec's construction cost estimate for a consolidated storage facility appears reasonable.

18.3.1.2 Estimated Operations Costs

In section 2.1, "Annual Operating Costs," of its cost estimate, Holtec estimated annual operating costs of $10 million for the facility. The operating cost estimate reflects only the first phase of the facility. As proposed by the applicant, the facility may be expanded by future license amendments to comprise 19 additional phases, and thus the operating cost may require adjustments in the future based on an expanded operational footprint.

In order to evaluate the reasonableness of this estimate, the staff reviewed the estimated ISFSI operations costs identified in two GAO reports. The 2009 GAO report cited above includes annual operations and maintenance costs for a centralized storage facility. The second GAO report, "Spent Nuclear Fuel Management: Outreach Needed to Help Gain Public Acceptance for

[3] Table 7, "Initial Assumptions and Component Cost Estimates for Our Centralized Storage and On-site Storage Alternatives and Modifications Made Based on Experts' Responses to Our Data Collection Instrument," pages 52–56.

Federal Activities that Address Liability," issued October 2014, provides safety and security system and annual operational costs for dry storage at a shutdown reactor site.

The 2009 GAO publication (page 54) estimates $8.8 million in operations costs for a CISF. By applying a 15 percent increase to account for inflation, this estimate results in an approximate $10 million annual cost in 2017 dollars. The 2014 GAO publication (page 52) estimates $2.5 million to $6.5 million in annual operations and other related costs; inflation from 2014 through 2017 is negligible, at approximately 5 percent during that time period.

As presented in its application, the Holtec operations cost estimate of $10 million lies above the two cost estimates developed in the two cited GAO reports. Accordingly, the staff concludes that Holtec's operations cost estimate for a centralized storage facility appears reasonable.

18.3.1.3 Funding Construction and Operations

In addition to its review of financial information provided by the applicant, the staff considered conclusions in the Private Fuel Storage, LLC (PFS), licensing proceeding, which are described in portions of the PFS SER and in Commission decisions in the proceeding, germane to financial qualifications. The staff notes that the Commission found in CLI-00-13 that an ISFSI presents risks less than those of a reactor, and that a 10 CFR Part 72 applicant need not meet the detailed financial qualifications requirements found in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Furthermore, the Commission found that a precisely drawn license condition, so that verification of compliance is a largely ministerial act, is an appropriate way of meeting the financial qualifications requirements found in 10 CFR Part 72. The applicant proposed license conditions 17 and 18 (which NRC renumbered in the final license as conditions 15 and 16) that address financial qualifications requirements at 10 CFR 72.22 and financial and legal provisions in service contracts between Holtec and its customers.

<u>Evaluation of Financial Information Provided by the Applicant</u>

Holtec proposed to fund construction of the first phase of storage capacity at the facility using its own funds. In its cost estimate, it states that Holtec's—

> …fully resourced capability infrastructure…imputes Holtec with a unique profile of predictable performance within the nuclear industry…
>
> Holtec International is well positioned to provide the financial assurance for the construction and oversight of Phase 1 of the CISF facility to include 500 HI-STORM UMAX canisters for the storage of Spent Nuclear Fuel (SNF) waste from commercial reactors. Our commitment is based on the willingness and capability of Holtec to fund the construction efforts of the CISF estimated to be in the range of ~$180 million.

As Holtec is a private company, publicly available information regarding its finances and financial performance is limited. In the proprietary version of its cost estimate, Holtec Report No. HI-2177593, Holtec provided some financial information that could be used to support analyses of its ability to fund construction and operations. Holtec provided net asset data as of the end of 2016; earnings before interest, taxes, depreciation, and amortization (EBITDA) data for fiscal year 2016; and projected cumulative EBITDA data for 2016 through 2020. In addition, Holtec provided a general statement that it has been profitable each year over the 30-year period since the company's inception.

In its proprietary submittal dated October 9, 2020, in which the applicant provided additional information, Holtec stated that it maintains

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In that submittal, Holtec also provided additional proprietary financial information in the form of audited balance sheets, statements of income, statements of changes in equity, and statements of cash flows, supplementing information initially provided with its earlier submitted cost estimate. The staff reviewed the financial statements provided by Holtec and identified several key financial data points for analysis. Specifically, staff concluded that Holtec's financial statements for years 2017 and 2018, reflect consistent sales revenue, positive net income, and a positive trend in growing both retained earnings and total equity, coupled with total liabilities remaining relatively flat, as identified below. Moreover, the measure, or magnitude, of Holtec's retained earnings, coupled with available credit, provide evidence that it will have adequate capital to undertake the construction of the facility based upon the estimated construction costs presented earlier in this analysis.

Regarding the funding of operations at the proposed facility, Holtec's plans to require financial provisions in service contracts with its customers, coupled with its own financial resources, as necessary, adequately address operational financial requirements of the facility, and thus provide evidence of the financial resources required to cover operational expenses.

Finally, in section 1.0, "Financial Profile of Holtec International," of its cost estimate, Holtec described how, as a matter of financial prudence, Holtec will require user agreements with customers that will justify the required capital expenditures by the company. License conditions addressing such agreements between Holtec and prospective customers are discussed below.

Accordingly, regarding whether the applicant, in accordance with 10 CFR 72.22, "…either possesses the necessary funds, or…has reasonable assurance of obtaining the necessary funds or that by a combination of the two,…will have the necessary funds available to cover…" estimated construction and operating costs over the planned life of the ISFSI, the staff concludes that evidence of such funding, for both construction and operations, has been provided by the applicant in information submitted as part of its application and in additional supplementary information.

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Holtec's Use of License Conditions for Facility Construction and Operations

While Holtec has proposed to fund construction of the first phase of the facility using its own funds, like PFS, Holtec has proposed license conditions 17 and 18 (which NRC revised and renumbered in the final license to conditions 15 and 16) that pertain to and further support the financial qualifications of the applicant.

As stated in section 1.0 of its cost estimate, "…if the…necessary contractual instruments are established insuring the minimum revenue stream needed to justify the facility…," Holtec will begin construction of the facility. Additionally, section 2.1 of the cost estimate states that "…[a]ll financial commitments related to annual operations will be tied to the sponsoring party's agreement with Holtec ..." Thus, final license to conditions 15 and 16 contain language that refers to the establishment of definitive agreements with customers and required provisions in its service contracts with customers, and these agreements, instruments, and associated provisions relate to the applicant's decision to fund and initiate construction and fund operations.

Staff Conclusions

Based on information provided by Holtec, its plans to self-fund construction of the facility appear reasonable. Holtec International and its subsidiaries maintain low levels of debt and associated interest expenses, and its balance of retained earnings and total equity, coupled with access to a senior credit facility, indicate that the applicant's plans to self-fund the project reflect that it possesses the necessary funds, or that it has reasonable assurance of obtaining the necessary funds, to cover estimated construction costs for the proposed facility.

Additionally, staff's license condition 15, based on the applicant's proposed license condition 17, states the following:

> In accordance with 10 CFR 72.22, the construction program will be undertaken only after a definitive agreement with the prospective customer for storing the used fuel at the HI-STORE CIS Facility has been established. Construction of any additional capacity beyond the initial capacity of 500 canisters shall commence only after funding is fully committed that is adequate to construct such additional capacity.

Thus, based on the financial information reflected in Holtec's financial reports and in its application, in combination with this license condition, the staff concludes that the applicant possesses the necessary funds, or has reasonable assurance of obtaining the necessary funds, or a combination of the two, to cover the cost of construction.

Similarly, regarding operational costs at the facility, based on the financial information reflected in Holtec's financial reports and in information provided in its application, Holtec appears to possess the necessary funds, or has reasonable assurance of obtaining the necessary funds, or a combination of the two, to cover the cost of operations, estimated at $10 million per year. As Holtec indicates in its application, financial commitments related to its annual operations will be tied to agreements with its customers. Moreover, as part of license condition 18, Holtec will "…include in its service contracts provisions requiring customers to provide periodically credit information, and, where necessary, additional financial assurances such as guarantees, prepayment, or payment bond," which gives additional assurance that operational costs of the licensed facility will be covered.

18.3.1.4 Financial Qualifications Summary

Based on its evaluation discussed above, the staff finds that the applicant provided reasonable construction and operations cost estimates for the proposed facility. Based upon financial data provided by the applicant and further supported by proposed conditions to be set forth in the license, the staff also concludes that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction and operations costs associated with the proposed facility. Accordingly, the staff concludes that the applicant is financially qualified to construct and operate the proposed facility and thus meets the requirements in 10 CFR 72.22 and conforms to the guidance of NUREG-1757.

18.3.2 Decommissioning Funding Assurance

The regulation at 10 CFR 72.30(b) requires an applicant to provide a DFP containing information on how reasonable assurance will be provided that funds will be available to decommission the facility. The DFP must contain a detailed DCE in an amount reflecting (1) the cost of an independent contractor to perform all decommissioning activities, (2) an adequate contingency factor, and (3) the cost of meeting the 10 CFR 20.1402 unrestricted use criteria (or the cost of meeting the 10 CFR 20.1403 restricted use criteria, provided the licensee can demonstrate its ability to meet these criteria). The licensee's DFP must also identify and justify using the key assumptions contained in the DCE. Further, the DFP must describe the method of assuring funds for ISFSI decommissioning, including means for adjusting cost estimates and associated funding levels periodically over the life of the ISFSI. Additionally, the DFP must specify the volume of onsite subsurface material containing residual radioactivity that will require remediation to meet the criteria for license termination and contain a certification that financial assurance for ISFSI decommissioning has been provided in the amount of the DCE.

18.3.2.1 Holtec Consolidated Interim Storage Facility Estimated Decommissioning Costs

Holtec's estimated total cost to decommission the ISFSI at the HI-STORE CIS Facility for unrestricted use, as presented in its DFP, is $23,716,355 (2017 dollars), which includes a 25 percent contingency. The cost estimate reflects decommissioning of the first phase of the facility; as proposed by the applicant, the facility may be expanded, through future license amendments, to comprise 19 additional phases. The first phase includes 500 HI-STORM UMAX canisters for the storage of SNF waste, to contain 8,680 MTU. In DFP table 9.14, "Total Decommissioning Costs," and table 2.2, "Estimated Project Decommissioning Expenditures," of "HI-STORE CIS Facility Financial Assurance & Project Life Cycle Cost Estimates," Holtec identified key decommissioning cost element expenditures as follows:

- planning and preparation: $410,000
- decontamination or dismantling of radioactive facility components: $672,000
- packaging, shipping, and disposal of radioactive waste: $14,082,000
- final radiation survey: $2,700,000
- site stabilization and long-term surveillance: $356,000
- security: $750,000
- contingency (25 percent): $4,743,000

Based on its analysis of Holtec's application, the staff concluded that the submitted DCE reflects reasonable costs of a third-party contractor, includes an adequate contingency factor, and is based on reasonable and documented assumptions. Absent the 25 percent contingency, the costs of packaging, shipping, and disposal of radioactive waste represent nearly 75 percent of the DCE, with final radiation survey activity costs amounting to an additional 14 percent of the DCE. These costs and percentages appear reasonable considering the nature of the storage and handling activities anticipated by the licensee as reflected in the application. Furthermore, as a basis for comparison, the staff also evaluated ISFSI DCEs provided by current 10 CFR Part 50 reactor licensees in recent submittals to the NRC provided with their site-specific DCEs for complete reactor facility decommissioning, as well as those provided by 10 CFR Part 50 licensees as part of their compliance with the reporting requirements in 10 CFR 72.30(c). Based on the details provided by the applicant and considering DCEs provided for onsite ISFSIs coupled with cost-scaling assumptions in line with the first phase of the Holtec facility (500 canisters containing up to 8,680 MTU), the staff determined that the applicant's cost estimate of $23,716,355 (2017 dollars) for decommissioning is reasonable. Accordingly, the staff finds that the DCE adequately estimates the cost to carry out required ISFSI decommissioning activities before license termination, and that the DCE is acceptable.

18.3.2.2 Holtec Decommissioning Funding Assurance

The Commission's regulations require that financial assurance for decommissioning a 10 CFR Part 72 facility such as that proposed by Holtec must be provided by one or more of the following three methods, pursuant to 10 CFR 72.30(e):

(1) prepayment before the start of operations in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities

(2) a surety method, insurance, or other method to guarantee that decommissioning costs will be paid (e.g., a surety method in the form of a surety bond, letter of credit, or line of credit)

(3) an external sinking fund in which deposits are made at least annually, coupled with a surety method or insurance, the value of which may decrease by the amount being accumulated in the sinking fund

In section 2.2, “Decommissioning Funding Assurance,” of the “HI-STORE CIS Facility Financial Assurance & Project Life Cycle Cost Estimates,” Holtec proposed to provide financial assurance in the following manner:

> A decommissioning fund will be established by setting aside $840 per MTU stored at the HI-STORE facility. These funds, plus earnings on such funds calculated at not greater than a 3 percent real rate of return over the 40-year license life of the facility, will cover the estimated cost to complete decommissioning.

This funding approach appears to reflect the prepayment method listed in 10 CFR 72.30(e)(1). The applicant indicated that the total it plans to set aside for decommissioning funding assurance (of $840 per MTU, applied to 8,680 MTU, equaling $7,291,200), compounded over a 40-year life at 3 percent, would provide adequate funding to meet the DCE of $23,784,170 (2017 dollars). The NRC staff observes that while the applicant’s calculation of funding available for decommissioning relies on prepayment, or “set-aside,” fees of $7,291,200 compounded over a 40-year period, the particular timing of early prepayment deposits is uncertain. However, based on additional information regarding Holtec’s financial resources as discussed throughout this SER, and the potential to rely on additional surety mechanisms for decommissioning as may be required (as further discussed below), the applicant’s prepayment method appears to provide reasonable assurance of decommissioning funding as required by 10 CFR Part 72.

In its submittal dated October 9, 2020, Holtec provided additional information regarding its use of a 3 percent real rate of return on funds set aside for decommissioning funding assurance. Unlike regulations governing 10 CFR Part 50 licensee decommissioning financial assurance (see 10 CFR 50.75(e)(1)(i)), which allow the applicant/licensee to “…take credit for projected earnings on the prepaid decommissioning trust funds, using up to a 2 percent annual real rate of return from the time of future funds’ collection through the projected decommissioning period,” 10 CFR Part 72 financial assurance regulations do not specify such a real rate of return on trust fund assets. Holtec indicated that it initially considered application of a 2 percent real rate of return on prepaid decommissioning funds, but that upon further analysis, it determined that returns not greater than a 3 percent real rate of return were achievable. Absent a requirement in 10 CFR Part 72 identifying a specific real rate of return allowance for such funds, the applicant’s assertion that its investment advisors’ analysis supports a 3 percent real rate of return, and recognition by the staff that some public utility commissions have identified assumed rates of return up to and exceeding 3 percent, the staff concludes that a 3 percent real rate of return on decommissioning assets for this 10 CFR Part 72 applicant is reasonable and, hence, acceptable.

Additionally in its October 9, 2020, submittal, the applicant indicated that "[t]he company is positioned to be able to use any of the three (3) methods, surety bond, letter of credit or insurance for *coupling with the external sinking fund* [emphasis added] to provide for the needed financial assurances." The applicant indicates these additional funding mechanisms, which are acceptable to the NRC per 10 CFR 72.30(e), may be used to fully meet decommissioning funding assurance requirements as may be necessary. To ensure that the applicant will have a funding mechanism in place prior to the receipt of spent fuel, the staff is imposing the following license condition 19:

> Prior to receipt of the material identified in sections 6.A and 7.A of this license, the Licensee shall have a decommissioning financial assurance instrument, in a form of one or more of the methods described in 10 CFR 72.30(e), reflecting the current decommissioning cost estimate.

Based on the applicant's prepayment approach, coupled with other sources of funding, as needed, and the staff's license condition, the staff has reasonable assurance that the funds required for decommissioning would be available to meet the decommissioning funding requirements of 10 CFR 72.30.

18.3.2.3 Decommissioning Funding Assurance Summary

In consideration of the above, the staff determined that the applicant, in accordance with requirements in 10 CFR 72.30, has provided a thorough and reasonable DCE for decommissioning the proposed facility. The staff also determined that the applicant's approach to decommissioning funding assurance, that of prepayment coupled with other surety instruments, as needed, provides reasonable assurance that funding will be available to decommission the proposed facility. Moreover, license condition 19 provides additional assurance that such funding will be in place prior to receipt of SNF at the site. Accordingly, the staff concludes that the applicant meets the requirements of decommissioning funding assurance at 10 CFR 72.30.

18.3.3 Holtec Onsite and Offsite Nuclear Insurance

In its submittal dated October 9, 2020, Holtec provided additional information regarding offsite nuclear liability insurance, stating that, if the DOE had not already taken possession of the SNF, Holtec plans to obtain offsite insurance by means of a $100 million insurance policy, or a smaller policy if the NRC formally approves a different amount for ISFSI-only sites in future NRC rulemaking.

Regarding onsite nuclear liability insurance, in its response submittal dated August 15, 2022, Holtec revised its proprietary project life-cycle cost estimate document, Holtec Report No. HI-2177593, thereby reflecting plans to obtain $50 million for onsite coverage. In the August 15, 2022, submittal, Holtec also reiterated its plan to obtain offsite insurance as reflected in its earlier response submittal. Holtec also included in its draft license submitted on August 15, 2022, proposed license condition 20, committing to obtaining insurance coverage as stated in the project life-cycle cost estimate.

The staff notes that a prior 10 CFR Part 72 application for an offsite, or away-from-reactor, ISFSI committed to acquiring and maintaining nuclear liability insurance. Specifically, the NRC-approved PFS license included a condition to maintain onsite and offsite nuclear liability insurance.

Holtec has committed to obtain and maintain nuclear liability insurance in the amount of $100 million to address offsite liability and $50 million for onsite liability. The staff considers these amounts reasonable based on the risk profile of the proposed activities and previous NRC determinations regarding the adequacy of financial assurance for SNF facilities. Therefore, Holtec's commitment to provide nuclear liability insurance in accordance with proposed license condition 20 (which NRC renumbered in the final license to condition 18) is acceptable to the staff.

18.4 Evaluation Findings

Based on information presented in the original application and additional information provided in subsequent submittals by the applicant, the NRC staff concludes that Holtec meets the financial qualifications requirements in 10 CFR 72.22 and the decommissioning funding assurance requirements in 10 CFR 72.30, and that information presented by the applicant adheres to NRC guidance in NUREG-1757 and NUREG-1567.

18.5 References

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities."

Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

Electric Power Research Institute, "Cost Estimate for an Away-from-Reactor Generic Interim Storage Facility (GISF) for Spent Nuclear Fuel," 2009.

Holtec International, "Holtec International HI-STORE CIS (Consolidated Interim Storage Facility) License Application," Docket No. 72-1051, March 30, 2017. Agencywide Documents Access and Management System Accession No. ML17115A431.

Holtec International, "Holtec International HI-STORE CIS (Consolidated Interim Storage Facility) License Application Responses to Requests for Additional Information—Part 6, Supporting Information" (proprietary), October 9, 2020. ML20290A465 (public version).

Holtec International, "HI-STORE Consolidated Interim Storage Facility, Requests for Additional Information Part 6, Attachment 2 Holtec Letter 5025059," October 9, 2020. ML20283A792.

Holtec International, Holtec Report No. HI-2177593, Revision 2, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground CISF—Financial Assurance & Project Life Cycle Cost Estimates” (proprietary), August 9, 2022. ML22227A162. Public version ML22331A008.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) CIS Facility - Decommissioning Cost Estimate and Funding Plan,” Holtec Report No. HI-2177565, Revision 1 (proprietary), November 16, 2022. Public version ML22331A012.

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Revision 0T, Report No. HI-2167374, January 20, 2023, Docket No. 72-1051. ML23025A112.

U.S. Government Accountability Office, “Nuclear Waste Management: Key Attributes, Challenges, and Costs for the Yucca Mountain Repository and Two Potential Alternatives,” November 2009.

U.S. Government Accountability Office, “Spent Nuclear Fuel Management: Outreach Needed to Help Gain Public Acceptance for Federal Activities that Address Liability,” October 2014.

U.S. Nuclear Regulatory Commission (NRC), NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” February 2000. ML003686776.

NRC, Memorandum and Order CLI-00-13, Docket No. 72-22, August 1, 2000. ML003737270.

NRC, “Consolidated Safety Evaluation Report Concerning the Private Fuel Storage Facility,” Chapter 17, “Financial Qualifications and Decommissioning Funding Assurance,” Docket No. 72-22, March 2002. ML020660749.

NRC, License No. SNM 2513, “License for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” issued to Private Fuel Storage, LLC, for the Private Fuel Storage Facility, February 21, 2006. ML060450438.

NRC, NUREG-1757, “Consolidated Decommissioning Guidance,” Volume 3, Revision 1, “Financial Assurance, Recordkeeping, and Timeliness,” February 2012. ML12048A683.

19 CONCLUSION

The staff has reviewed the design, testing, operations, maintenance, and other safety-related activities and features for the HI-STORE Consolidated Interim Storage (CIS) Facility, as described in the following documents submitted by the applicant:

- the License Application, which contains general and financial information, the applicant's technical qualifications, technical specifications, and a preliminary decommissioning plan;
- the safety analysis report for the HI-STORE CIS Facility;
- the emergency plan for the HI-STORE CIS Facility; and
- the physical security plan for the HI-STORE CIS Facility, which includes the safeguards contingency plan.

Based on the information provided in the above documents, the conditions specified in the proposed technical specifications and the license conditions identified in this safety evaluation report (SER), and the use of previously approved HI-STORM UMAX and HI-STORM FW storage systems and the respective final safety analysis reports that the applicant incorporated by reference, the staff concludes that the HI-STORE CIS Facility meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." Pursuant to 10 CFR 72.40(a), the staff has made the following findings:

- 10 CFR 72.40(a)(1) - Based on the evaluation throughout this SER, the staff finds that the applicant's proposed ISFSI design complies with Subpart F, "General Design Criteria," of 10 CFR Part 72.
- 10 CFR 72.40(a)(2) - Based on the evaluation in chapters 2, 4, 7, 11, and 14 of this SER, the staff finds that the proposed site complies with the criteria in Subpart E, "Siting Evaluation Factors," of 10 CFR Part 72.
- 10 CFR 72.40(a)(4) - Based on the evaluation in chapter 10 of this SER, the staff finds that the applicant is qualified by reason of training and experience to conduct the operation covered by the regulations in this part.
- 10 CFR 72.40(a)(5) - Based on the evaluation in chapter 3 of this SER, the staff finds that the applicant's description of its proposed operating procedures to protect health and to minimize danger to life or property are adequate.
- 10 CFR 72.40(a)(6) - Based on the evaluation in chapter 18 of this SER, the staff finds that the applicant for the ISFSI is financially qualified to engage in the proposed activities in accordance with the regulations in this part.

- 10 CFR 72.40(a)(7) - Based on the evaluation in chapter 12 of this SER, the staff finds that the applicant's quality assurance plan complies with Subpart G, “Quality Assurance,” of 10 CFR Part 72.

- 10 CFR 72.40(a)(8) - Based on the evaluation in chapter 10 of this SER, the staff finds that the applicant's physical protection provisions comply with Subpart H, “Physical Protection,” of 10 CFR Part 72.

- 10 CFR 72.40(a)(9) - Based on the evaluation in chapter 10 of this SER, the staff finds that the applicant's personnel training program complies with Subpart I, “Training and Certification of Personnel,” of 10 CFR Part 72.

- 10 CFR 72.40(a)(10) - Based on the evaluation in chapter 13 of this SER, the staff finds that the applicant's preliminary decommissioning plan, pursuant to 10 CFR 72.30, provides reasonable assurance that decontamination and decommissioning of the ISFSI at the end of its useful life will provide adequate protection to the health and safety of the public.

- 10 CFR 72.40(a)(11) - Based on the evaluation in chapter 10 of this SER, the staff finds that the applicant's emergency plan complies with 10 CFR 72.32.

- 10 CFR 72.40(a)(12) - This regulatory requirement is outside the scope of this SER.

- 10 CFR 72.40(a)(13) - Based on the evaluation throughout this SER, the staff finds that there is reasonable assurance that: (i) The activities authorized by the license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of Chapter 10 of the *Code of Federal Regulations*.

- 10 CFR 72.40(a)(14) - Based on the evaluation in chapter 10 of this SER, the staff finds that the issuance of a license for the HI-STORE CIS Facility will not be inimical to the common defense and security.

19.1 References

Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

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Holtec International, “Final Safety Analysis Report on the HI-STORM UMAX Canister Storage System,” Holtec Report No. HI-2115090, Revision 3, June 29, 2016, Docket No. 72-1040. ML16193A339.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) CISF—Security Training and Qualification Plan,” Revision 2 (nonpublic), Report No. HI-2177561, H-037-17, transmittal dated March 30, 2019. ML19094A271.

Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) Underground Consolidated Interim Storage Facility—Physical Security Plan,” Revision 3 (nonpublic), Report No. HI-2177559, H-011-20, transmittal dated March 2, 2020. ML20065H155.

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Holtec International, “Holtec International & Eddy Lea Energy Alliance (ELEA) CIS Facility - Decommissioning Cost Estimate and Funding Plan,” Holtec Report No. HI-2177565, Revision 1 (proprietary), November 16, 2022. Public version ML22331A012.

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Holtec International, “Attachment 2 - Proposed HI-STORE License/Technical Specifications,” November 23, 2022. ML22331A005.

Holtec International, “Licensing Report on the HI-STORE CIS Facility,” Revision 0T, Report No. HI-2167374, January 20, 2023, Docket No. 72-1051. ML23020A133.