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Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site

October 2023

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U.S. Department of Energy

EXECUTIVE SUMMARY

This *Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site* (Draft CSSF 3116 Basis Document) concerns the Calcined Solids Storage Facility (CSSF) at the Idaho National Laboratory Site near Arco, Idaho. The CSSF currently stores calcine that was produced by calcining liquid waste from the prior reprocessing of spent nuclear fuel and non-reprocessing waste into a granular solid (called calcine).[1] The U.S. Department of Energy (DOE) will retrieve calcine from the CSSF and then grout and close the CSSF.

In accordance with Section 3116(a) of the "Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005" (NDAA) [hereinafter referred to as NDAA Section 3116(a)], high-level radioactive waste does not include certain waste from the reprocessing of spent nuclear fuel in Idaho or South Carolina that the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission (NRC), determines meets the criteria in NDAA Section 3116(a). The DOE is issuing this Draft CSSF 3116 Basis Document to demonstrate that those criteria will be satisfied. The Final CSSF 3116 Basis Document will support a potential determination that the Secretary of Energy, in consultation with the NRC, may make pursuant to NDAA Section 3116(a).

NDAA Section 3116(a) states:

> In General – Notwithstanding the provisions of the Nuclear Waste Policy Act of 1982, the requirements of section 202 of the Energy Reorganization Act of 1974, and other laws that define classes of radioactive waste, with respect to material stored at a Department of Energy site at which activities are regulated by a covered State pursuant to approved closure plans or permits issued by the State, the term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy (in this section referred to as the "Secretary"), in consultation with the Nuclear Regulatory Commission (in this section referred to as the "Commission"), determines –
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical; and
>
> (3) (A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will be disposed of –
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of –

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1. At the Idaho National Laboratory Site, reprocessing of spent nuclear fuel occurred from 1952 to 1992, and calcine was generated from 1963 to 2000. Calcination converted liquid wastes into a solid form in a high-temperature fluidized bed. During calcination, liquid radioactive wastes were atomized with air and sprayed into a heated bed, and the principal calcination reactions were evaporation and thermal decomposition. The CSSF currently stores approximately 4,400 m^3 (155,300 ft^3) of calcined solids. The CSSF stores the calcine in stainless-steel bins that are housed in a series of six discrete, reinforced-concrete vaults. These vaults are known as CSSFs 1 through 6 and collectively as "the CSSF," with each of the CSSF vaults containing three to 12 stainless-steel storage bins.

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and

(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and

(iii) pursuant to plans developed by the Secretary in consultation with the Commission.

As to the first Section 3116(a) criterion, this Draft CSSF 3116 Basis Document shows that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein will meet all other Section 3116 criteria and do not raise any unique considerations that would require permanent isolation in a deep geologic repository.

This Draft CSSF 3116 Basis Document also demonstrates that highly radioactive radionuclides will be removed from the CSSF to the maximum extent practical, thereby satisfying the second criterion in Section 3116(a). DOE will remove calcine and other nonradioactive material (primarily startup bed nonradioactive material)[2] from the CSSF using pneumatic retrieval technology. Retrieval operations using pneumatic technologies have been proven effective by historic and ongoing retrieval demonstrations. DOE anticipates, and has demonstrated (with retrieval demonstrations), that approximately 99% or more of the calcine (by volume) and approximately 99% of the radioactivity[3] attributable to highly radioactive radionuclides will be removed from the CSSF. Following removal of calcine from the CSSF, DOE will stabilize the CSSF bins (including integral equipment), transport lines, and any residual calcine therein with grout.[4]

Additionally, as demonstrated in this Draft CSSF 3116 Basis Document, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of CSSF closure are anticipated to meet concentration limits for Class C low-level radioactive waste as set out in 10 *Code of Federal Regulations* (CFR) 61.55, thus meeting the third criterion in Section 3116(a). DOE also is consulting with the NRC on DOE's disposal plans for CSSF pursuant to NDAA Section 3116(a)(3)(B)(iii) to take full advantage of the consultation process established by Section 3116(a).

This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet the performance objectives in 10 CFR 61, Subpart C, in accordance with the criteria in Section 3116(a). These performance objectives address protection of the general population from radioactive releases,

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2. Small amounts of nonradioactive startup bed material and nonradioactive liquid feed used to transition from the startup bed to radioactive waste were sent to the CSSF. In addition, most of the calcine solids formed in the calcination process were nonradioactive oxides of aluminum and zirconium from the fuel cladding and calcium fluoride formed from the calcium nitrate added to fluoride-bearing wastes during the calcination process to prevent fluoride volatility. Only a small fraction (less than 1 wt%) of the calcine was composed of radioactive elements from reprocessing of spent nuclear fuel (Staiger and Swenson 2021).

3. While 99% is the overall anticipated removal efficiency, actual removal efficiency may vary slightly between each CSSF. Under current plans, CSSF 1 calcine will be transferred to CSSF 6 in the first phase of closure of the CSSF. For additional information, DOE will then transfer the retrieved calcine from CSSF 6 and the remaining CSSFs that store calcine to appropriate facilities based on future decisions.

4. This Draft CSSF 3116 Basis Document addresses residual calcine and CSSF structures and components that have had contact with calcine and will be grouted and disposed of in situ. These structures and components include (1) the waste storage and transfer equipment that comprises the bins, distributor lines, cyclones, and transport lines, (2) the off-gas system, access risers, and rod-out lines, and (3) components contained within the bins such as the thermowells and corrosion coupons. CSSF structures, components, and residual waste included in this Draft CSSF 3116 Basis Document are referred to as "CSSF bins (including integral equipment), transport lines, and any residual calcine therein."

protection of individuals from inadvertent intrusion on the disposal site, protection of individuals during disposal facility operations, and stability of the disposal site after closure.

DOE has analyzed potential doses to a future member of the public and inadvertent human intruder from the CSSF bins (including integral equipment), transport lines, and any residual calcine therein after closure. Results from the *Performance Assessment and Composite Analysis for the INTEC Calcined Solids Storage Facility at the INL Site* show that there is reasonable assurance that the peak annual all-pathways effective dose for a future hypothetical member of the public and a total effective dose for a hypothetical inadvertent human intruder will remain well below 25 mrem and 500 mrem, respectively, in compliance with the performance objectives in 10 CFR 61.41 and 61.42.

DOE has programs in place to ensure the protection of workers and the public during facility operations. As demonstrated in this Draft CSSF 3116 Basis Document, DOE requirements for occupational radiological protection and those for radiological protection of the public are comparable to the relevant requirements in the performance objective in 10 CFR 61.43.

DOE reviewed site characteristics, including demography, geography, meteorology, climatology, ecology, geology, seismology, and hydrogeology. As demonstrated in this Draft CSSF 3116 Basis Document, site conditions do not present hazards that impact CSSF stability. In addition, CSSF closure methods will result in a facility closure that does not require ongoing maintenance. As such, this Draft CSSF 3116 Basis Document demonstrates that the CSSF at the time of closure meets the performance objective in 10 CFR 61.44 concerning long-term site stability.

In accord with the third Section 3116(a) criterion, the CSSF will be removed from service (operationally closed) and stabilized pursuant to State-approved closure plans, consistent with requirements in the *Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL* under the Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act. (See Appendix B.)

Pursuant to Section 3116(a), DOE is consulting with NRC concerning this Draft CSSF 3116 Basis Document. DOE is also making this Draft CSSF 3116 Basis Document available to states, Tribal Nations, stakeholders, and the public for comment. After careful consideration of NRC comments and comments received from states, Tribal Nations, stakeholders, and the public, DOE will perform any necessary revisions of analyses and technical documents and will prepare a Final CSSF 3116 Basis Document. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in consultation with the NRC, may potentially determine in the future whether the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are not high-level radioactive waste and may be disposed of (closed) in place at the Idaho National Laboratory Site as low-level radioactive waste.

CONTENTS

FIGURES

TABLES

ACRONYMS

ALARA	as low as reasonably achievable
AoA	analysis of alternatives
BEA	Battelle Energy Alliance, LLC
BLM	U.S. Bureau of Land Management
bls	below land surface
CA	composite analysis
CERCLA	Comprehensive Environmental Response, Compensation and Liability Act
CFA	Central Facilities Area
CFR	*Code of Federal Regulations*
CSSF	Calcined Solids Storage Facility
DEQ	Department of Environmental Quality
DOE	U.S. Department of Energy
DOE-EM	U.S. Department of Energy Office of Environmental Management
ED	effective dose
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ESRP	Eastern Snake River Plain
FFA/CO	Federal Facility Agreement and Consent Order
HIP	hot isostatic pressing
HLW	high-level radioactive waste
HPM	Historical Processing Model
HRR	highly radioactive radionuclide
HWMA	Hazardous Waste Management Act
ICDF	Idaho CERCLA Disposal Facility
ICP	Idaho Cleanup Project
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LLW	low-level radioactive waste
MCL	maximum contaminant level
MEI	maximally exposed individual
MFC	Materials Fuels Complex

NDAA	National Defense Authorization Act
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
NWCF	New Waste Calcining Facility
OFAT	one factor at a time
OSHA	Occupational Safety and Health Administration
OU	operable unit
PA	performance assessment
RBA	radioactive buffer area
RCRA	Resource Conservation and Recovery Act
ROD	record of decision
ROM	rough order of magnitude
RWMC	Radioactive Waste Management Complex
SNF	spent nuclear fuel
SOF	sum of fractions
SRPA	Snake River Plain Aquifer
STP	Site Treatment Plan
TAN	Test Area North
TEDE	total effective dose equivalent
TFF	Tank Farm Facility
USGS	U.S. Geological Survey
WCF	Waste Calcining Facility

NOMENCLATURE

Chemical Element Symbol and Name

Ac	actinium
Ag	silver
Al	aluminum
Am	americium
Ar	argon
At	astatine
Ba	barium
Be	beryllium
Bi	bismuth
C	carbon
Ca	calcium
Cd	cadmium
Ce	cerium
Cf	californium
Cm	curium
Co	cobalt
Cr	chromium
Cs	cesium
Eu	europium
Fe	iron
Fr	francium
Gd	gadolinium
H	hydrogen
Ho	holmium
I	iodine
In	indium
K	potassium
Kr	krypton
La	lanthanum
Mg	magnesium
Na	sodium
Nb	niobium
Nd	neodymium
Ni	nickel
Np	neptunium
Pa	protactinium
Pb	lead
Pd	palladium
Pm	promethium
Po	polonium
Pr	praseodymium
Pu	plutonium
Ra	radium
Rb	rubidium
Rh	rhodium
Rn	radon
Ru	ruthenium
Sb	antimony
Se	selenium
Sm	samarium
Sn	tin
Sr	strontium
Tb	terbium
Tc	technetium
Te	tellurium
Th	thorium
Tl	thallium
Tm	thulium
U	uranium
Y	yttrium
Zr	zirconium

Chemical Formula Symbol and Name

$CaCO_3$	calcium carbonate or calcite
$CaCl_2$	calcium chloride
$Ca\text{-}Mg\text{-}HCO_3$	calcium bicarbonate
K_2O	potassium oxide
NaK	sodium-potassium (alloy)

Isotope and Name

H-3	tritium

Equipment Numbering

CYC	cyclone
TAV	transport air valve
VES	vessel
WCS	waste calcine storage

Scientific Abbreviation and Description

μg	microgram
Bq	becquerel
cfs	cubic feet per second
Ci	curie
cm	centimeter
cms	cubic meters per second
ft	feet
g	gram
gal	gallon
gpm	gallon per minute
ha	hectare
hr	hour
in.	inch
ka	kiloannum
K_d	soil-to-water partition coefficient (ratio of solid phase to solute concentrations)
km	kilometer
L	liter
lb	pound
m	meter
Ma	megaannum
mg	milligram
mi	mile
mL	milliliter
mm	millimeter
mph	miles per hour
mrem	millirem
mSv	millisievert
pCi	picocurie
R	roentgen
rem	roentgen equivalent man
s	second
ß	beta
Sv	sievert
yr	year
γ	gamma

Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site

1. INTRODUCTION

Section 1 Purpose

The purpose of this section is to provide introductory information that lays the foundation for detailed discussions in later sections of this document, *Draft Basis for Section 3116 Determination for Closure of the Calcined Solids Storage Facility at the Idaho National Laboratory Site,* hereinafter referred to as the Draft CSSF 3116 Basis Document.

Section 1 Contents

This section describes the purpose and scope of the Draft CSSF 3116 Basis Document, provides background information concerning the Calcined Solids Storage Facility (CSSF), identifies technical requirements on which this Draft CSSF 3116 Basis Document is based, and outlines the contents of the remainder of the document.

Section 1 Key Points

- The U.S. Department of Energy (DOE) is issuing this Draft CSSF 3116 Basis Document to provide a basis, after DOE issues a Final CSSF 3116 Basis Document, for a potential determination by the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission (NRC), pursuant to Section 3116(a) of the "Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005" (NDAA), hereinafter referred to as Section 3116(a). This Draft CSSF Basis Document demonstrates that the criteria in NDAA Section 3116(a) will have been met at closure of the CSSF, and accordingly, the closed CSSF may be disposed of (closed) in place as low-level radioactive waste (LLW).
- This Draft CSSF 3116 Basis Document concerns the CSSF located at Idaho Nuclear Technology and Engineering Center (INTEC) on the Idaho National Laboratory (INL) Site. The CSSF stores solid high-level radioactive waste (HLW) (referred to as "calcine") and other nonradioactive material (primarily startup bed nonradioactive material) in stainless-steel bins housed in six discrete reinforced-concrete vaults, known as CSSFs 1 through 6, each containing three to 12 stainless-steel storage bins. Calcine was generated from 1963 to 2000 by converting (calcining) liquid HLW from the reprocessing of spent nuclear fuel and non-reprocessing waste stored in tanks at the INTEC Tank Farm Facility into a granular solid (i.e., calcine). The liquid HLW resulted from reprocessing of spent nuclear fuel by DOE and its predecessor agencies from 1952 to 1992 at INTEC.
- This Draft CSSF 3116 Basis Document addresses residual calcine and CSSF structures and components that have had contact with calcine. These structures and components include: (1) the waste storage and transfer equipment that comprises the bins, distributor lines, cyclones, and transport lines, (2) off-gas system, access risers, and rod-out lines, and (3) components contained within the bins such as the thermowells and corrosion coupons. CSSF structures, components, and residual waste included in this Draft CSSF 3116 Basis

Document will be referred to hereinafter as “CSSF bins (including integral equipment), transport lines, and any residual calcine therein.”

- DOE will remove calcine and other nonradioactive material (primarily startup bed nonradioactive material) from the CSSF using pneumatic retrieval technologies. Retrieval operations using pneumatic technologies have been proven effective by past and ongoing retrieval demonstrations. DOE anticipates, and has demonstrated (with retrieval demonstrations), that approximately 99% or more of the calcine (by volume) and approximately 99% of the radioactivity attributable to highly radioactive radionuclides (HRRs) will be removed from the CSSF.
- Following removal of calcine from the CSSF, DOE will stabilize the CSSF bins (including integral equipment), transport lines, and any residual calcine therein with grout.
- This Draft CSSF 3116 Basis Document does not include the calcination process, treatment or disposition of the retrieved calcine, or other facilities, systems, or wastes at INTEC or the INL Site.
- This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein will have had HRRs removed to the maximum extent practical at the time of closure. Removal of calcine containing HRRs will occur to the maximum extent practical using proven pneumatic retrieval technologies. As discussed in this Draft CSSF 3116 Basis Document, further removal of HRRs is not practical and would, among other things, increase the risk to workers and result in an insignificant reduction in the very low potential doses to a member of the public and the hypothetical human intruder.
- As demonstrated in this Draft CSSF 3116 Basis Document, the stabilized CSSF bins (including integral equipment), transport lines, and any residual waste therein at the time of closure are anticipated to meet concentration limits for Class C LLW, as set out in 10 *Code of Federal Regulations* (CFR) 61.55. DOE is consulting with the NRC on DOE’s disposal plans for the CSSF as described in this Draft CSSF 3116 Basis Document pursuant to NDAA Section 3116(a)(3)(B)(iii) to take full advantage of the consultation process established by Section 3116(a).
- This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein located at the CSSF at the time of closure will meet the 10 CFR 61, Subpart C, performance objectives so as to provide for the protection of public health and the environment. These performance objectives address protection of the general population from radioactivity releases, protection of individuals from inadvertent intrusion on the disposal site, protection of individuals during disposal facility operations, and stability of the disposal site after closure.
- Through the use of the performance assessment process, DOE has analyzed the possible doses to a future member of the public and inadvertent human intruder after CSSF closure. Results from the *Performance Assessment and Composite Analysis for the INTEC Calcined Solids Storage Facility at the INL Site* show that there is reasonable assurance that the peak annual all-pathways effective dose (ED) for a future hypothetical member of the public and a total ED for a hypothetical inadvertent human intruder will remain below 25 mrem and 500 mrem, respectively, in compliance with the performance objectives in 10 CFR 61.41 and 61.42.

- DOE has programs in place to ensure the protection of workers and the public during facility operations. As demonstrated in this Draft CSSF 3116 Basis Document, DOE requirements for occupational radiological protection and those for radiological protection of the public during operations are equivalent to the relevant requirements contained in the performance objective in 10 CFR 61.43.
- This Draft CSSF 3116 Basis Document demonstrates that the CSSF at the time of closure meets the performance objective in 10 CFR 61.44 concerning long-term site stability. DOE reviewed site characteristics, including demography, geography, meteorology, climatology, ecology, geology, seismology, and hydrogeology. As demonstrated in this Draft CSSF 3116 Basis Document, site conditions do not present hazards that impact CSSF stability. In addition, CSSF closure methods will result in a facility closure that does not require ongoing maintenance.
- CSSF structures will be removed from service (operationally closed) and stabilized pursuant to State-approved closure plans, consistent with the Idaho Hazardous Waste Management Act/Resource Conservation and Recovery Act partial permit for the CSSF (see Appendix B) and Section 3116(a) criteria.
- Furthermore, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not raise any unique considerations that, notwithstanding the demonstration that all other NDAA Section 3116(a) criteria have been met, require permanent isolation in a deep geologic repository.
- This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure meet the criteria in NDAA Section 3116(a). DOE is consulting with NRC and making this Draft CSSF 3116 Basis Document available to states, Tribal Nations, stakeholders, and the public for comment. After careful consideration of NRC advice and comments received from states, Tribal Nations, stakeholders, and the public, DOE will perform any necessary revisions of analyses and technical documents and will prepare a Final CSSF 3116 Basis Document. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in consultation with the NRC, may potentially determine in the future whether the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are not HLW and may be disposed of (closed) in place at the INL Site as LLW.

1.1 Overview

In accordance with the National Defense Authorization Act (NDAA)[5] Section 3116(a), certain waste from the reprocessing of spent nuclear fuel (SNF) in Idaho or South Carolina is not high-level radioactive waste (HLW)[6] that the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission (NRC), determines meets the provisions in NDAA Section 3116(a).

The U.S. Department of Energy (DOE) is issuing this Draft CSSF 3116 Basis Document to demonstrate that the criteria in NDAA Section 3116(a) will be met. DOE will remove calcine and other nonradioactive material from the CSSF and then stabilize (in grout) and close the CSSF. DOE will prepare a Final CSSF Basis Document, and the Secretary of Energy, in consultation with NRC, may potentially determine whether the CSSF meets the criteria in Section 3116(a) and may be disposed of (closed) in place as low-level radioactive waste (LLW).[7]

Those NDAA Section 3116(a) criteria are, in relevant part:

> "(1) [the waste] does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;

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5. Public Law 108-375

6. "High-level radioactive waste" is defined in Section 2(12) of the "Nuclear Waste Policy Act of 1982," as amended (42 USC 10101 et seq.), as: "(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation." Section 11.dd of the "Atomic Energy Act of 1954," as amended (42 USC 2011 et seq.), and Section 2(10) of the "Waste Isolation Pilot Plant Land Withdrawal Act," as amended (Public Law 102-579), incorporate the above definition. A similar definition is found in Attachment 2 of DOE M 435.1-1 Chg 3 (LtdChg), "Radioactive Waste Management Manual."

   Although the term "reprocessing" is not defined statutorily, DOE M 435.1-1 Chg 3 defines reprocessing as: "Actions necessary to separate fissile elements (U-235, Pu-239, U-233, and Pu-241) and/or transuranium elements (e.g., Np, Pu, Am, Cm, Bk) from other materials (e.g., fission products, activated metals, cladding) contained in spent nuclear fuel for the purposes of recovering desired materials. Separation processes include aqueous separation processes, e.g., the Redox and the Purex processes, and nonaqueous processes, e.g., pyrometallurgical and pyrochemical processes. Wastes that are produced upstream of these separations processes, from processes such as chemical or mechanical decladding, cladding separations, conditioning, or accountability measuring, are not high-level waste. Such wastes are considered processing wastes and should be managed in accordance with the appropriate Chapters of DOE M 435.1-1, as either transuranic, mixed low-level, or low-level waste. Likewise, wastes that are produced downstream of these separations processes, from such processes as decontamination, rinsing, washing, treating, vitrifying, or solidifying, are also not high-level waste and should be managed accordingly. Upstream and downstream wastes are not high-level waste because they do not result from reprocessing."

   The term "spent nuclear fuel" is defined in Section 2(23) of the "Nuclear Waste Policy Act of 1982," as amended (42 USC 10101, et seq.), as "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing." Section 11.dd of the "Atomic Energy Act of 1954," as amended (42 USC 2011 et seq.), and Section 2(15) of the "Waste Isolation Pilot Plant Land Withdrawal Act of 1992," as amended (Public Law 102-579), incorporate the above definition. The term "spent nuclear fuel" is defined in Attachment 2 of DOE M 435.1-1 Chg 3 in relevant part as: "Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Test specimens of fissionable material irradiated for research and development only, and not production of power or plutonium, may be classified as waste, and managed in accordance with the requirements of this Order when it is technically infeasible, cost prohibitive, or would increase worker exposure to separate the remaining test specimens from other contaminated material. [Adapted from: DOE 5820.2A]." NRC regulations include a similar definition. See 10 CFR 71.4.

7. LLW is essentially defined in relevant part in Section 2(9) of the "Low-Level Radioactive Waste Policy Amendments Act of 1985," as amended (42 USC 2021b et seq.) as "radioactive material … that is not high-level radioactive waste, spent nuclear fuel, or byproduct material (as defined in the Atomic Energy Act of 1954)." Section 2(16) of the "Nuclear Waste Policy Act of 1982," as amended (42 USC 10101, et seq.) and DOE M 435.1-1, Chg 3 similarly define LLW in relevant part as radioactive waste that "is not high-level radioactive waste, spent nuclear fuel, transuranic waste, [or] byproduct material[.]"

(2) [the waste] has had highly radioactive radionuclides removed to the maximum extent practical; and

(3) (A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will be disposed of –

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and

(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or

(B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of –

(i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;

(ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and

(iii) pursuant to plans developed by the Secretary in consultation with the Commission."

NDAA Section 3116(a) is provided in its entirety in Section 3 of this Draft CSSF 3116 Basis Document.

This Draft CSSF 3116 Basis Document concerns the CSSF located at the Idaho Nuclear Technology and Engineering Center (INTEC), which is on the Idaho National Laboratory (INL) Site near Arco, Idaho. The CSSF[8] stores calcine and other nonradioactive material (primarily startup bed nonradioactive material) in stainless-steel bins in a series of six discrete reinforced-concrete vaults, each containing three to 12 stainless-steel storage bins. Calcine was generated from 1963 to 2000 by converting liquid HLW and non-reprocessing waste into a granular solid.[9] The liquid HLW resulted from the reprocessing of SNF by the DOE and its predecessor agencies from 1952 to 1992 at INTEC.

This Draft CSSF 3116 Basis Document includes residual calcine[10] and CSSF structures and components that have had contact with calcine. These structures and components, described in more detail in Subsection 2.11.1, include: (1) the waste storage and transfer equipment that comprises the bins, distributor lines, cyclones, and transport lines,[11] (2) off-gas system, access risers, and rod-out lines, and

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8. CSSF refers to the seven calcine storage units identified as CSSFs 1 through 7. The facility is commonly referred to as the "bin sets" because of multiple bins housed in each of the seven reinforced-concrete storage vaults at CSSFs 1 through 7. Each CSSF comprises the waste and transfer equipment (bins, distributor lines, cyclone, and transport lines), facility structures (storage vault, cyclone vault, instrument and equipment building), and other equipment such as the off-gas system, sump and cooling systems, safety support systems, and utilities. Subsection 2.11.1 and Figures 2-27, 2-28, 2-32, 2-34, 2-36, 2-38, and 2-40 provide additional descriptions for CSSFs 1 through 6. CSSF 7 is excluded because it never received or stored calcine.

9. The calcination process is described in Subsection 2.11.2 of this Draft CSSF 3116 Basis Document. The CSSF currently stores approximately 4,400 m^3 (155,300 ft^3) of calcined solids.

10. For purposes of this Draft CSSF 3116 Basis Document, "residual calcine," "residual waste," or "residuals" means the relatively small amount of waste remaining in the CSSF after removal of radioactive nuclides to the maximum extent practical. Stabilization of these residuals will be carried out by filling the CSSF bins (including integral equipment), and remaining structures with grout after completion of waste retrieval activities.

11. Most of the CSSF waste storage and transfer equipment is contained within the concrete storage vaults, except for the cyclone and transport lines. The cyclone for each CSSF is housed in a cyclone vault on top of the storage vault, and the transport lines travel between the cyclone vaults and the calcining facilities. The CSSF storage vaults, which house the bins and distributor lines, do not contain residual calcine waste; however, it is expected the storage vaults will be grouted for structural stability. Subsection 2.11.1 describes in more detail the configuration of each CSSF.

(3) components contained within the bins such as the thermowells and corrosion coupons. CSSF structures, components, and residual waste included in this Draft CSSF 3116 Basis Document will be referred to hereafter as "CSSF bins (including integral equipment), transport lines, and any residual calcine therein."

DOE will remove calcine from the CSSF and stabilize the CSSF bins (including integral equipment), transport lines, and any residual calcine therein in place. DOE will remove calcine from each CSSF using pneumatic retrieval technology.[12] Retrieval operations using pneumatic technologies has been proven effective by past and ongoing retrieval demonstrations. DOE anticipates, and has demonstrated (in retrieval demonstrations), that approximately 99% or more of the calcine (by volume) and approximately 99% of the radioactivity attributable to highly radioactive radionuclides (HRRs)[13] will be removed from the CSSF. Given the risks inherent in exhuming the bins and other structures and equipment, DOE plans to close the CSSF in place, stabilizing the CSSF bins (including integral equipment), transport lines, and any residual calcine therein with grout[14] to reduce risks to workers, the public, and the environment. Subsection 2.11.4 provides additional information on closure of the CSSF.

This Draft CSSF 3116 Basis Document shows that the criteria in NDAA Section 3116(a) will be satisfied for the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at closure of the CSSF.

This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein will have had HRRs removed to the maximum extent practical at the time of closure. Removal of calcine containing HRRs to the maximum extent practical at the CSSF will occur using proven pneumatic waste retrieval technology. As discussed in Subsections 5.2 and 5.3 of this Draft CSSF 3116 Basis Document, further removal of HRRs is not practical and would, among other things, increase risk to workers and result in an insignificant reduction in the very low potential doses to the public and the hypothetical human intruder.

As demonstrated in this Draft CSSF 3116 Basis Document, stabilized CSSF residual wastes at closure are anticipated to meet concentration limits for Class C LLW as set out in 10 CFR 61.55. DOE nevertheless will consult with the NRC on DOE's disposal plans for the CSSF as described in this Draft CSSF 3116 Basis document and supporting references pursuant to the consultation process in NDAA Section 3116(a)(3)(B)(iii), to take full advantage of the consultation process established by NDAA Section 3116(a).

This Draft CSSF 3116 Basis Document demonstrates that stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet 10 CFR 61, Subpart C, performance objectives so as to provide for the protection of the public health and the environment. These performance objectives address protection of the general population from radioactive

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12. Under current plans, CSSF 1 calcine will be transferred to CSSF 6 in the first phase of closure of the CSSF. See Subsection 2.11.3.3 of this Draft CSSF 3116 Basis Document.

    For additional information, DOE will transfer the retrieved calcine from CSSF 6 and the remaining CSSFs that contain calcine to appropriate treatment and packaging facilities based on future decisions. See Subsection 1.3 of this Draft CSSF 3116 Basis Document.

13. HRRs are those radionuclides that contribute most significantly to radiological dose to workers, the public, and the environment, as well as radionuclides listed in 10 CFR 61.55 for low-level radioactive waste.

14. Grout will be made from materials such as cement, fly ash, fine aggregate, and water to create a free-flowing material that will be used to fill the bins after waste retrieval is completed. The grout will harden in the bins and vault structures to stabilize the residual waste and provide structural stability for closure of the CSSF. DOE will tailor and finalize the specific formulation of the grout in the future before it is added to the bins and vaults.

releases, protection of individuals from inadvertent intrusion into the disposal site, protection of workers and the public during disposal facility operations, and the stability of the disposal site after closure.

Through use of the performance assessment (PA) process, DOE has analyzed the potential doses to a future member of the public and hypothetical inadvertent intruder from the CSSF bins (including integral equipment), transport lines, and any residual calcine therein after closure. Results from the *Performance Assessment and Composite Analysis for the INTEC Calcined Solids Storage Facility at the INL Site* (CSSF PA/composite analysis [CA]) (DOE-ID 2022a)[15] show that there is reasonable assurance that the peak annual all-pathways effective dose (ED) for a future hypothetical member of the public and a total ED for a hypothetical inadvertent intruder will remain below 25 mrem and 500 mrem, respectively, in compliance with the performance objectives in 10 CFR 61.41 and 10 CFR 61.42.

DOE has programs in place to ensure the protection of workers and the public during facility operations. As demonstrated in this Draft CSSF 3116 Basis Document, DOE requirements for occupational radiological protection and those for radiological protection of the public and the environment are comparable to the relevant requirements contained in the performance objective in 10 CFR 61.43.

This Draft CSSF 3116 Basis Document demonstrates that the CSSF at the time of closure will meet the performance objective in 10 CFR 61.44 concerning long-term site stability. DOE reviewed site characteristics, including demography, geography, meteorology, climatology, ecology, geology, seismology, and hydrogeology. As demonstrated in this Draft CSSF 3116 Basis Document, site conditions do not present hazards that impact CSSF stability. In addition, CSSF closure methods will result in a facility closure that minimizes or does not require ongoing maintenance (see Subsection 7.4 and Appendix B for additional details).

CSSF structures will be removed from service (operationally closed) and stabilized pursuant to State-approved closure plans, consistent with the *Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL* (PER-114) under the Idaho Hazardous Waste Management (HWMA) (Idaho Code 39-4401 et seq.)/Resource Conservation and Recovery Act (RCRA) (42 USC 6901 et seq.) and the criteria of Section 3116(a). The information provided in Appendix B demonstrates that DOE's approach for closure of the CSSF ensures HWMA/RCRA closure standards and requirements will be met and closed in accordance with PER-114.

Furthermore, stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not raise any unique considerations that, notwithstanding the demonstration that all other NDAA Section 3116(a) criteria have been met, require permanent isolation in a deep geologic repository.

This Draft CSSF 3116 Basis Document demonstrates that stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure meet the criteria in NDAA Section 3116(a). DOE is consulting with NRC and making this Draft CSSF 3116 Basis Document available for comments by states, Tribal Nations, stakeholders, and the public. After careful consideration of NRC comments and comments received from states, Tribal Nations, stakeholders, and the public, DOE will perform any necessary revisions of analyses and technical documents and will

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15. A PA and CA are required and maintained pursuant to DOE M 435.1-1 Chg 3. Generally, a PA is a multidisciplined assessment (e.g., geochemistry, hydrology, materials science, and health physics) that uses a variety of computational modeling codes to evaluate groundwater concentrations and doses at various points of assessment over time. In doing this assessment, DOE evaluates the impact of natural features (e.g., hydrology, soil properties, groundwater infiltration) and engineered barriers (e.g., closure cap, fill grout, bin design) on the release of radionuclides to estimate, among other things, the potential dose to a hypothetical member of the public and a hypothetical inadvertent intruder. The results of the CSSF PA/CA, as reported here, should not be considered limits or thresholds. As required by DOE M 435.1-1 Chg 3, maintenance of the CSSF PA/CA will include future PA revisions or special analyses to incorporate new information, update model codes, and reflect analyses of actual residual inventories.

prepare a Final CSSF 3116 Basis Document. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in consultation with the NRC, may potentially determine in the future whether the CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are not HLW and, accordingly, may be closed in place (disposed of) at the INL Site as LLW.

The information summarized in this section is discussed in detail in subsequent sections of this Draft CSSF 3116 Basis Document and references cited therein.

1.2 Purpose and Scope

The purpose of this Draft CSSF 3116 Basis Document is to demonstrate and document that, after waste retrieval and stabilization activities (i.e., grouting in place) are complete, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure meet the criteria in NDAA Section 3116(a).

DOE is consulting with the NRC and providing the opportunity for comments by states, Tribal Nations, stakeholders, and the public, as discussed in Subsections 1.1 and 1.4. Thereafter, DOE plans to prepare a Final CSSF 3116 Basis Document considering NRC comments and comments from states, Tribal Nations, stakeholders, and the public, and performing any necessary revisions of analyses. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in consultation with the NRC, may potentially determine, in a future 3116 Secretarial Determination, whether the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at closure satisfy the criteria in NDAA Section 3116(a), are not HLW, and may be disposed of (closed) in place on the INL Site as LLW. The scope of this Draft CSSF 3116 Basis Document exclusively addresses the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein after stabilization and at the time of CSSF closure. This Draft CSSF 3116 Basis Document does not include the calcination process, treatment or disposition of the retrieved calcine, or other facilities, systems, or wastes.

This Draft CSSF 3116 Basis Document is premised on the facts, assumptions, and analyses contained or referenced herein. The Final CSSF 3116 Basis Document will also be premised on facts, assumptions, and analyses contained or referenced therein. Accordingly, an NDAA Section 3116(a) Secretarial Determination made in reliance on a Final CSSF 3116 Basis Document will only cover situations consistent with those facts, assumptions, and analyses.

1.3 Additional Background

A 1995 Settlement Agreement (DOE 1995) (also referred to as the Batt Agreement) was signed by DOE, the State of Idaho, and the Department of Navy to settle certain litigation related to the "National Environmental Policy Act of 1969" (NEPA) (42 USC 4321 et seq.) and incorporated into a Consent Order entered by the U.S. District Court for the District of Idaho.[16] The Settlement Agreement/Consent Order provides that: "DOE shall treat all high-level waste currently at INEL so that it is ready to be moved out of Idaho for disposal by a target date of 2035." The Settlement Agreement/Consent Order also provides that a Record of Decision (ROD) be issued not later than December 31, 2009, pursuant to an environmental impact statement (EIS) that analyzes alternatives for treatment of calcined waste. The *Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement* (Final EIS) (DOE 2002) was issued in 2002, and the *Record of Decision for the Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement* (2005 ROD) (DOE 2005), "Amended Record of Decision: Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement" (2006 ROD Amendment) (71 FR 68811), and "Amended Record of Decision: Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement Revised by State 12/21/09" (2010 ROD Amendment) (75 FR 137) were issued in 2005, 2006, and 2010, respectively.

The 2002 Final EIS (DOE 2002) analyzed alternatives for managing HLW calcine, mixed transuranic waste/sodium-bearing waste, newly generated liquid waste at the INL Site in solid and liquid forms, and the final disposition of INTEC HLW management facilities. DOE and the State of Idaho (which participated as a cooperating agency) identified separate waste processing alternatives but preferred the same alternative for facilities disposition, which is to use performance-based closure methods for existing facilities and to design new facilities consistent with clean closure methods. The 2005 ROD (DOE 2005) adopted a phased decision-making strategy to issue a series of amended RODs that would address future waste processing and closure of HLW facilities. The 2005 ROD (DOE 2005) addressed sodium-bearing waste treatment, facilities disposition (excluding the INTEC Tank Farm Facility [TFF]), bin sets closure, and DOE's strategy for HLW calcine.) The 2006 ROD Amendment (71 FR 68811) addressed performance-based closure of the TFF. The 2010 ROD Amendment (75 FR 137) identified hot isostatic pressing (HIP) as the preferred technology to treat calcine. However, DOE has determined that HIP treatment may not be the most effective path forward, as described in a letter from Nicole Hernandez, with the DOE Idaho Operations Office, to Natalie Creed, with the Idaho Department of Environmental Quality (DEQ), on September 30, 2019, titled "Notification for Site Treatment Plan (STP) Table 5-1 Milestones for Calcine Disposition Project" (Hernandez 2019). Additionally, in 2021, DOE published the *Independent Analysis of Alternatives for Disposition of the Idaho Calcined High-Level Waste Inventory – Final Report* (AoA Final Report) (DOE-EM 2021), which evaluated potential treatment technologies and any newly available disposal pathways for calcine. The AoA Final Report (DOE-E53 2021) concluded that vitrification provides the best processing option and key aspects of vitrification, including waste form development and testing, should be initiated to mature the technology for calcine treatment.

Regardless of the treatment approach that ultimately will be used, all options require removal of calcine from the CSSF. Treatment of calcine is not important to this Draft CSSF 3116 Basis Document, except that the capability to treat the waste is needed to complete retrieval operations from the CSSF to achieve closure of all the bin sets.[17]

The 2005 ROD phased strategy provides that DOE will develop calcine retrieval demonstration processes. DOE has tested a full-scale retrieval system to demonstrate DOE's ability to safely retrieve calcine from

16. The Consent Order was entered by the U.S. District Court for the District of Idaho on October 17, 1995, to settle *Public Service Co. of Colorado v. Batt*, No. CV-91-0035-S-EJL (D. Idaho) and *United States v. Batt*, No. CV-91-0065-S-EJL (D. Idaho). Since 1995, the State of Idaho has entered into five agreements that provide waivers to certain requirements and milestones of the 1995 Settlement Agreement.

17. For additional information, DOE will transfer the retrieved calcine to appropriate facilities based on future decisions.

the CSSF (see Subsection 5.2 for details). Testing objectives were to eliminate risks, optimize final design configurations, and determine the efficacy of calcine removal using a pneumatic retrieval system. Retrieval testing demonstrated that calcine can be removed to the maximum extent practical. Testing has also demonstrated that RCRA (42 USC 6901 et seq.) closure performance standards, under authority of the Idaho HWMA (Idaho Code 39-4401 et seq.), and Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (42 USC 9601 et seq.) remedial action objectives, under the Federal Facility Agreement and Consent Order (FFA/CO) (DOE-ID 1991), can be met. See Subsection 2.11.4 for additional details on the CSSF closure approach and State-approved closure plans.

The retrieval demonstration is being performed as a subproject under the Calcine Disposition Project, which is part of the overall Idaho Cleanup Project (ICP) mission (see Subsection 5.2 for details). In 2015, DOE published the *Independent Analysis of Alternatives for Disposition of the Idaho Calcined High-Level Waste Inventory, Volume 1 – Summary Report* (AoA Summary Report) (DOE-EM 2016) for the treatment and disposal of calcine. DOE concluded that, regardless of treatment and disposal uncertainties, all treatment and disposal options require removal of calcine from the CSSF. Thus, per recommendations in the AoA Summary Report, and confirmed in the AoA Final Report (DOE-EM 2021), DOE divided the Calcine Disposition Project into two subprojects: calcine retrieval and calcine processing, with a near-term project priority on calcine retrieval activities. The Calcine Retrieval Project was initiated in 2016, and scope was added to the ICP contract to develop and test a full-scale retrieval system to demonstrate DOE's ability to safely retrieve calcine from CSSF 1 and transfer it to CSSF 6.[18] Results from Calcine Retrieval Project testing and previous retrieval demonstrations have shown pneumatic retrieval has removal efficiencies of approximately 99% or more of the calcine (by volume) and approximately 99% of the radioactivity attributed to HRRs. Though the Calcine Retrieval Project demonstration focused on the retrieval and transfer of calcine from CSSF 1 to CSSF 6, the pneumatic retrieval processes are applicable to each CSSF and calcine can be transferred to a different location, such as a treatment facility. Additional information on the closure approach and calcine retrieval technologies is presented in Subsections 2.11.4 and 5.2, respectively.

The *Idaho National Laboratory Site Treatment Plan* (INL STP) (DOE-ID 2022b), first issued and approved by the Idaho DEQ in 1995, establishes an enforceable framework and schedule for DOE to achieve compliance with mixed waste land disposal restrictions. The INL STP was developed pursuant to RCRA, as amended by the "Federal Facilities Compliance Act of 1992" (Public Law 102-386), and the Idaho HWMA. Calcine is identified in the INL STP as mixed waste to be treated in a manner that will be suitable for long-term storage outside of Idaho.

DOE is implementing a strategy to test pneumatic retrieval technologies that have proven the effectiveness of these technologies in removing the waste. The retrieval technology demonstration is consistent with the 2005 ROD (DOE 2005).

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18. The AoA Summary Report (DOE-EM 2016) recommended a full-scale radioactive demonstration of retrieval and transport systems that includes transferring the 220 m^3 of calcine in CSSF 1 to CSSF 6. CSSF 6 was not filled to capacity due to calcine operations ending in 2000 and has approximately 793 m^3 of usable capacity to receive CSSF 1 calcine. Continuing with the retrieval demonstration was recommended in the AoA Final Report (DOE-EM 2021).

1.4 Opportunity for Comment

DOE is consulting with the NRC on this Draft CSSF 3116 Basis Document and making it available for comment by states, Tribal Nations, stakeholders, and the public.[19] After careful consideration of comments from the NRC and comments from states, Tribal Nations, stakeholders, and the public, DOE will perform any necessary revisions of analyses and technical documents and will prepare a Final CSSF 3116 Basis Document. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in consultation with the NRC, may potentially determine in the future whether the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure satisfy the criteria in NDAA Section 3116(a), are not HLW, and may be disposed of (closed) in place as LLW.

1.5 Document Organization

The remainder of this document is organized as follows:

- Section 2—Provides general background information regarding the INL Site and INTEC, and describes the CSSF and its operational history. This section also presents the composition of the calcine and calculations of the radionuclide inventory for the CSSF.
- Section 3—Provides NDAA Section 3116(a) criteria.
- Section 4—Describes how the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein after stabilization (grouting) at the time of closure meet Criterion (1) of NDAA Section 3116(a) and explains that the waste contained therein does not require permanent isolation in a deep geologic repository for HLW.
- Section 5—Describes which radionuclides are HRRs and how HRRs will be removed from the CSSF to the maximum extent practical, thereby meeting Criterion (2) of NDAA Section 3116(a).
- Section 6—Demonstrates that radionuclide concentrations in the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of CSSF closure are less than Class C concentration limits for LLW in 10 CFR 61.55, “Waste Classification,” thereby meeting Criterion (3)(A) of NDAA Section 3116(a).
- Section 7—Discusses how the NRC performance objectives in 10 CFR 61, Subpart C, “Performance Objectives,” will be achieved, thereby meeting Criterion (3)(A)(i) of NDAA Section 3116(a).
- Section 8—Summarizes closure of the CSSF in compliance with State-approved closure plans, thereby meeting Criterion 3(A)(ii) of NDAA Section 3116(a).
- Section 9—Presents conclusions from this Draft CSSF 3116 Basis Document.
- Section 10—Provides references used in this document.
- Appendix A—Provides a comparison of the DOE and NRC LLW disposal requirements
- Appendix B—Provides DOE’s strategy for regulatory closure of the CSSF.

19. Though not required by NDAA Section 3116(a), DOE is providing states, Tribal Nations, stakeholders, and the public with the opportunity to comment. The DOE and NRC consultative process will also be open for the public to observe, including access to formal documentation related to the consultation process. DOE will consider all comments received during a 45-day comment period.

2. BACKGROUND

Section 2 Purpose

This section provides background information on the INL Site necessary to understand the environment in which the CSSF is located. Additionally, this section provides background on the CSSF and the CSSF inventory both before and after retrieval operations.

Section 2 Contents

This section provides general background information regarding the INL Site, including the local land use, meteorology, ecology, geology, seismology, volcanology, and hydrology. It also describes the design of the CSSF, the calcine waste, the radionuclide inventory at the CSSF both before and after retrieval operations, and the closure approach at CSSF.

Section 2 Key Points

- This section describes INL Site land use, physical characteristics, and natural resources.
- The INL Site is remotely located from most developed areas, and access to the INL Site is restricted. DOE expects to retain ownership and control of the INL Site until at least 2095 and will continue to manage portions that cannot be released for unrestricted land use beyond 2095. Adjacent areas are not likely to experience residential or commercial development. No residential development will be allowed to occur within the INL Site boundaries while the land is under DOE control.
- Meteorological data have been collected at the INL Site from a network of observation towers since 1949 to support design engineering, facility operations, and operations safety. The Eastern Snake River Plain (ESRP) region is classified as an arid climate with overall light annual rainfall, mostly mild winds, warm summers, and cold winters.
- The INL Site is underlain by a sequence of volcanic rocks and sedimentary interbeds that are more than 3,408 m (10,000 ft) thick. The volcanic rocks consist mainly of basalt flows in the upper part of the sequence and rhyolitic ash-flow tuffs in the lower part. At least 178 basalt-flow groups and 103 sedimentary interbeds underlie the INL Site above the effective base of the Snake River Plain Aquifer (SRPA).
- The INL Site is surrounded by the seismically active Intermountain and Centennial Tectonic seismic belts, located in the mountain ranges east and west of the INL Site, respectively. However, long-term seismic monitoring indicates that the ESRP is relatively seismically inactive, and ongoing activity is likely associated with nearby volcanic processes.
- The SRPA is one of the largest and most productive groundwater resources in the United States and underlies the INL Site. The Class I aquifer is the primary source of water for domestic, municipal, and industrial use in southeastern Idaho. The U.S. Geological Survey and DOE maintain a groundwater monitoring network at the INL Site to collect water levels and water quality data.
- The INL Site represents the largest remnant of undeveloped, ungrazed sagebrush-steppe ecosystem in the Intermountain West. Portions of the INL Site are dedicated as a Sagebrush-Steppe Ecosystem Reserve and Sage-grouse Conservation Area. The natural plant life is limited by soil type, meager rainfall, and extended drought periods and consists mainly of sagebrush and various grasses. The INL Site supports a variety of wildlife, including small mammals, birds, reptiles, and a few large mammals.
- This section describes INTEC and CSSF historical SNF reprocessing operations. SNF reprocessing began in 1952 and was phased out in 1992. INTEC's current mission is to receive and temporarily store SNF and other radioactive waste and perform remedial actions to address legacy waste.

- Calcine production began in 1963 at the Waste Calcining Facility (WCF), filling CSSFs 1 through 3. Operations then switched to the New Waste Calcining Facility (NWCF) in 1982, which filled CSSFs 4 through 6. CSSF 7 was built to receive calcine from the NWCF, but it was never placed in service. CSSFs 1 through 5 are filled to or near capacity. CSSF 6 was partially filled before calcining operations ended in 2000. Approximately 4,440 m^3 of calcine is stored in CSSFs 1 through 6.
- Although designed and constructed over a span of 30 years, each CSSF shares common design features with the others, including several stainless-steel storage bins (the basis for the term "bin set") housed in a reinforced-concrete vault that is below or partially below grade, a high-efficiency particulate air (HEPA)-filtered storage vault, and cooling and monitoring systems.
- The transport lines used to transfer calcine from the WCF and NWCF to the CSSF vary in size, length, and depth. Generally, however, the transport lines share some common features: Each CSSF has a set of two 3-in. stainless-steel lines that travel in a containment pipe fabricated from either carbon- or stainless-steel pipe. The transport lines and containment pipe are encased in reinforced-concrete shielding.
- To calculate the calcine radionuclide residual inventory expected after waste removal from the CSSF, data from the Historical Processing Model (HPM) was used. HPM data provide accurate detailed data on calcine volume, mass, and composition (chemical and radioactivity). The HPM uses multiple sources (i.e., historical knowledge and waste samples) and verified databases and calculation methods to calculate calcine activities.
- The residual radionuclide inventory was calculated based on the assumption that a 5.1-cm (2-in.) depth of residual calcine will remain in each bin after waste retrieval operations. Retrieval operations using pneumatic technologies have been proven effective by historic and ongoing retrieval demonstrations. DOE anticipates and has demonstrated that at approximately 99% or more of the calcine (by volume) and approximately 99% of the radioactivity attributable to HRRs will be removed from the CSSF.
- The CSSF bins (including integral equipment), transport lines, and any residual calcine therein equipment that cannot be removed after waste retrieval operations are complete will be stabilized with grout and remain in place at closure. It is anticipated that the CSSF will potentially be covered with an engineered cap, which will be designed at a later date. This Draft CSSF 3116 Basis Document and its supporting references do not assume or take credit for a potential closure cap.

2.1 Site Characteristics

Originally established in 1949 as the National Reactor Testing Station, the 2,305-km^2 (890-mi^2) INL Site is a DOE-managed reservation located in a relatively remote, lightly populated portion of the Eastern Snake River Plain (ESRP) in southeastern Idaho (see Figure 2-1). Historically, the INL Site also has borne the names Idaho National Engineering Laboratory and Idaho National Engineering and Environmental Laboratory. In 2003, DOE defined two business units for the INL Site:

- INL research and development missions under DOE's Office of Nuclear Energy. Battelle Energy Alliance, LLC (BEA), is the INL management and operating contractor for these missions. INL research and development missions focus on nuclear energy research, sustainable energy systems, and national and homeland security.
- ICP cleanup mission is under DOE's Office of Environmental Management (DOE-EM). The Idaho Environmental Coalition, LLC, is the contractor responsible for the environmental cleanup mission at the INL Site and will be the responsible contractor from 2022 through 2031. The cleanup mission focuses on legacy wastes generated from World War II-era conventional weapons testing, government-owned research and defense reactors, SNF reprocessing, and nuclear energy research.

Current DOE operations take place at several different facilities at the INL Site. Most of the facilities are centrally located in the southern portion of the Site (see Figure 2-1). INTEC, formerly known as the Idaho Chemical Processing Plant, was established in 1952, and its primary mission as stated in the 2002 Final EIS (DOE 2002) was to reprocess (recover) uranium from SNF. INTEC includes approximately 80 ha (200 acres) and is located in the south-central portion of the INL Site (see Figure 2-1). In addition to reprocessing SNF, INTEC's original missions included storage of SNF, as well as nuclear research (DOE 2002). DOE phased out the reprocessing operations in 1992 (DOE 1992) and redirected the plant's mission to:

- Receive and temporarily store SNF and other radioactive waste for future disposition
- Manage current and past (legacy) wastes
- Perform remedial actions.

A more in-depth description of INTEC SNF operations and the CSSF is provided in Subsection 2.11. Subsections that follow describe the INL Site geography, land use, physical characteristics, hydrology, ecology, and natural resources. Cited references provide additional details.

2.1.1 Geography and Demographics

The INL Site is located in southeastern Idaho, on the north-central part of the ESRP (see Figure 2-2). Included in its 2,305-km^2 (890-mi^2) area are portions of five Idaho counties (Bingham, Bonneville, Butte, Clark, and Jefferson). The nearest INL Site boundaries are 51 km (32 mi) west of Idaho Falls, 37 km (23 mi) northwest of Blackfoot, 71 km (44 mi) northwest of Pocatello, and 11 km (7 mi) east of Arco, Idaho. The INL Site is approximately equidistant from the three other metropolitan areas of Salt Lake City, Utah, 339 km (211 mi); Boise, Idaho, 413 km (257 mi); and Butte, Montana, 344 km (214 mi).

The surface of the INL Site is a relatively flat, semiarid, sagebrush desert. Predominant relief is manifested either as volcanic buttes jutting up from the desert floor or as unevenly surfaced basalt flows or flow vents and fissures (Irving 1993). Elevations on the INL Site range from 1,460 m (4,790 ft) in the south to 1,802 m (5,913 ft) in the northeast, with an average elevation of 1,524 m (5,000 ft) above sea level (Irving 1993). Mountain ranges bordering the INL Site on the north and west are the Lost River Range, the Lemhi Range, and the Beaverhead Mountains of the Bitterroot Range (Irving 1993). The Snake River Plain Aquifer (SRPA) underlies approximately 25,900 km^2 (10,000 mi^2) of the ESRP. The location of the INL Site relative to the SRPA is illustrated in Figure 2-3. The SRPA, which consists of saturated basalt and sediments, is one of the largest aquifers in the United States (Irving 1993) and, in 1991, was classified as a sole-source aquifer by the U.S. Environmental Protection Agency (EPA) (56 FR 50634). Generally, groundwater flows in the SRPA from the northeast to the southwest, and the aquifer discharges at Thousand Springs, near Twin Falls, Idaho, 241 km (150 mi) from the INL Site border (DOE-ID 2022a).

INTEC occupies approximately 80 ha (200 acres) in the south-central portion of the INL Site and consists of more than 100 buildings, trailers, and support facilities (see Figure 2-4). Primary facilities at INTEC include space for storage and treatment of SNF, mixed HLW, and mixed transuranic waste (DOE-ID 2022a). Located outside the INTEC perimeter fence are parking areas, a helicopter landing pad, the wastewater treatment lagoon, various pits, and evaporation ponds. These areas occupy approximately 22 ha (55 acres) (DOE-ID 2022a).

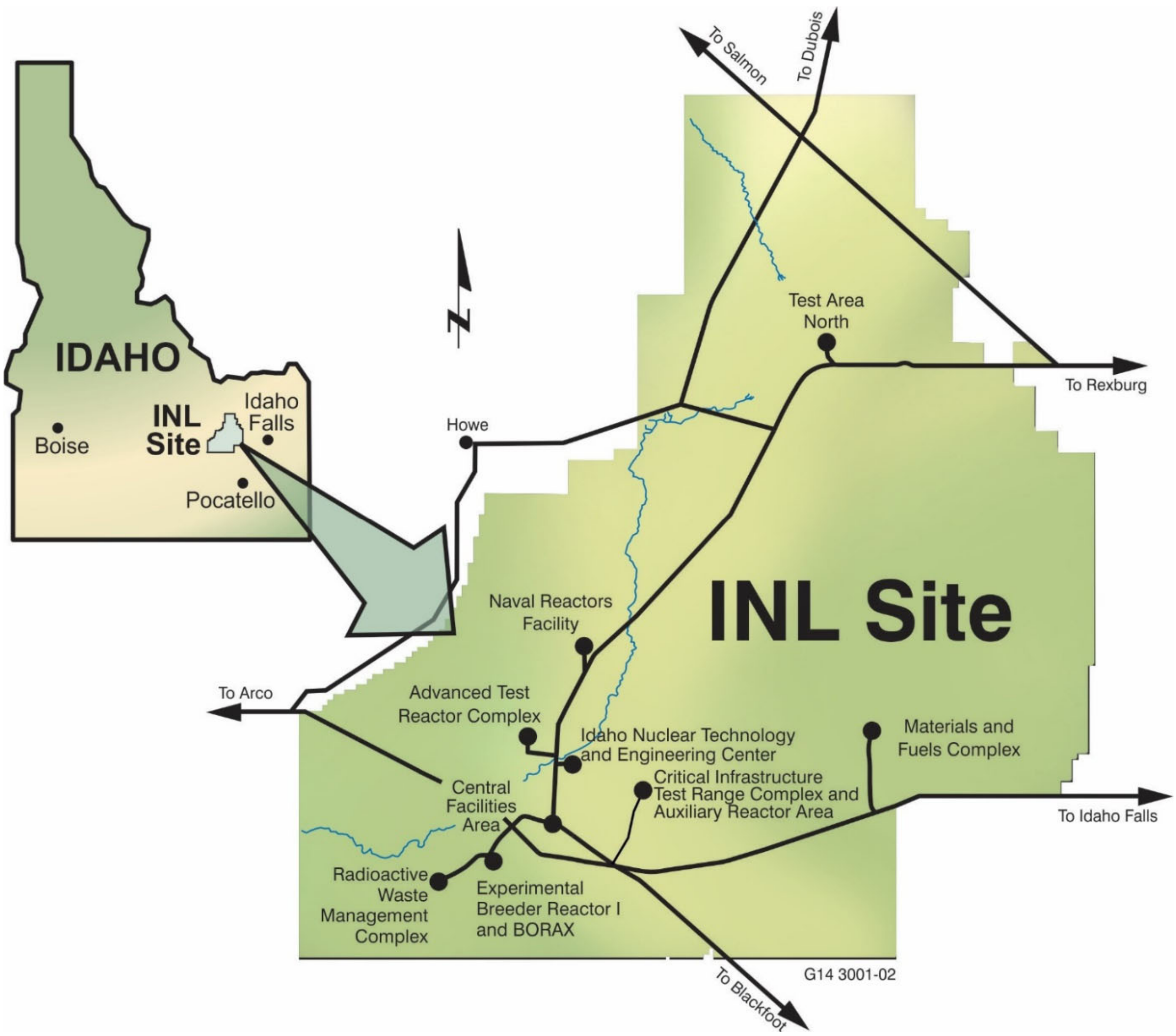

Figure 2-1. Idaho National Laboratory Site (DOE-ID 2022a).

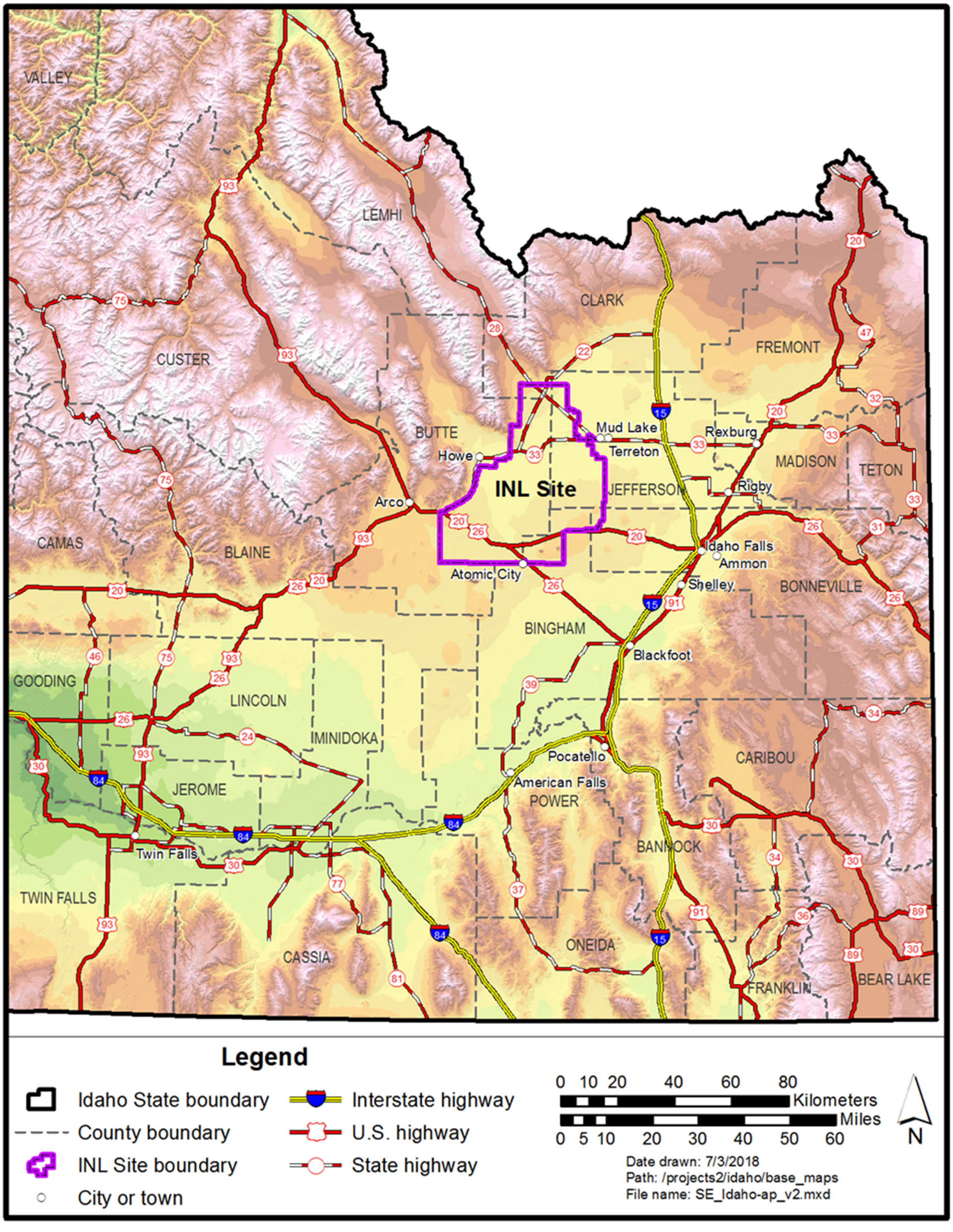

Figure 2-2. Location of the Idaho National Laboratory Site in southeastern Idaho (DOE-ID 2022a).

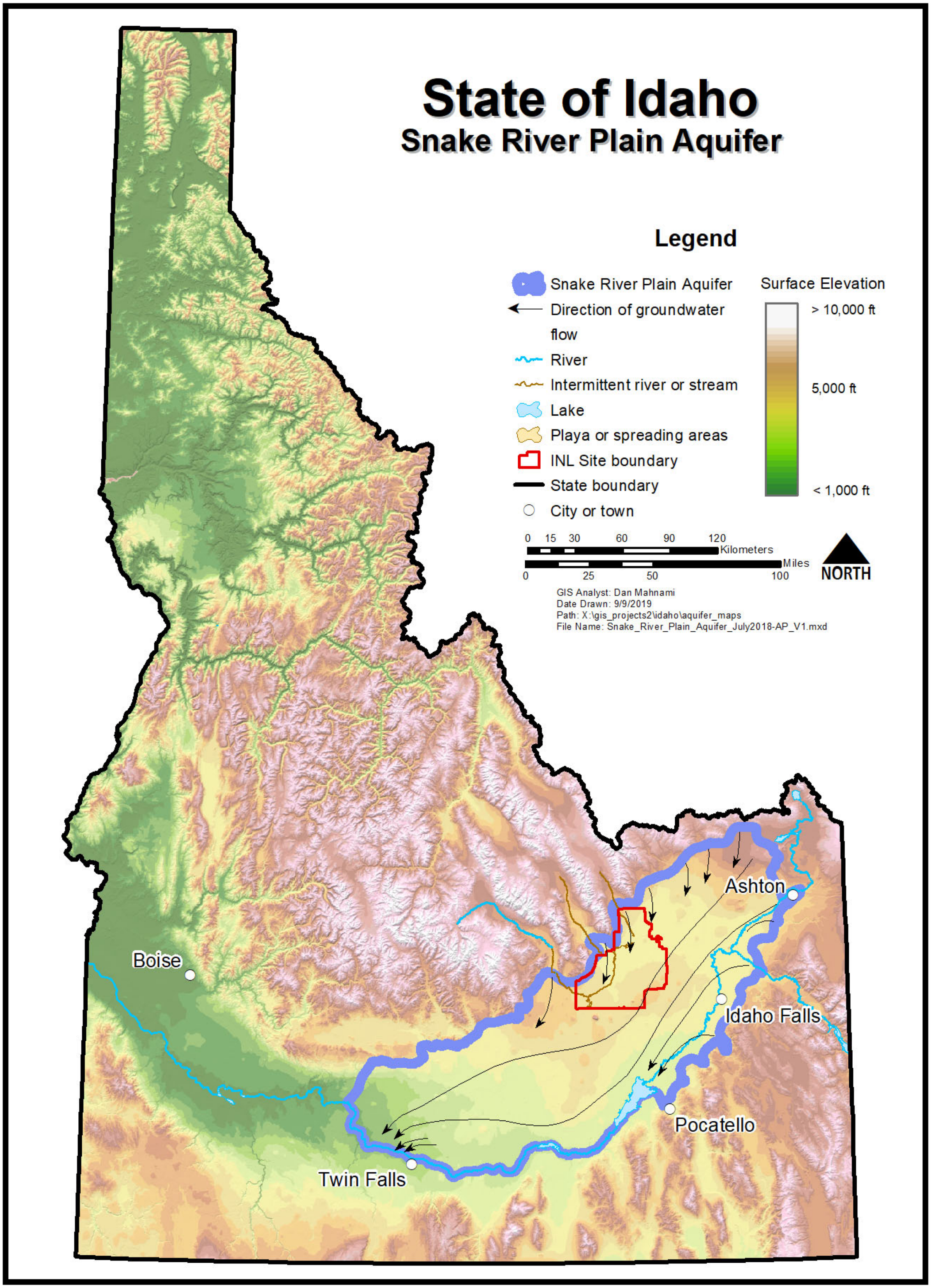

Figure 2-3. Surrounding terrain and aquifer flow directions beneath the Idaho National Laboratory Site in southeastern Idaho (DOE-ID 2022a).

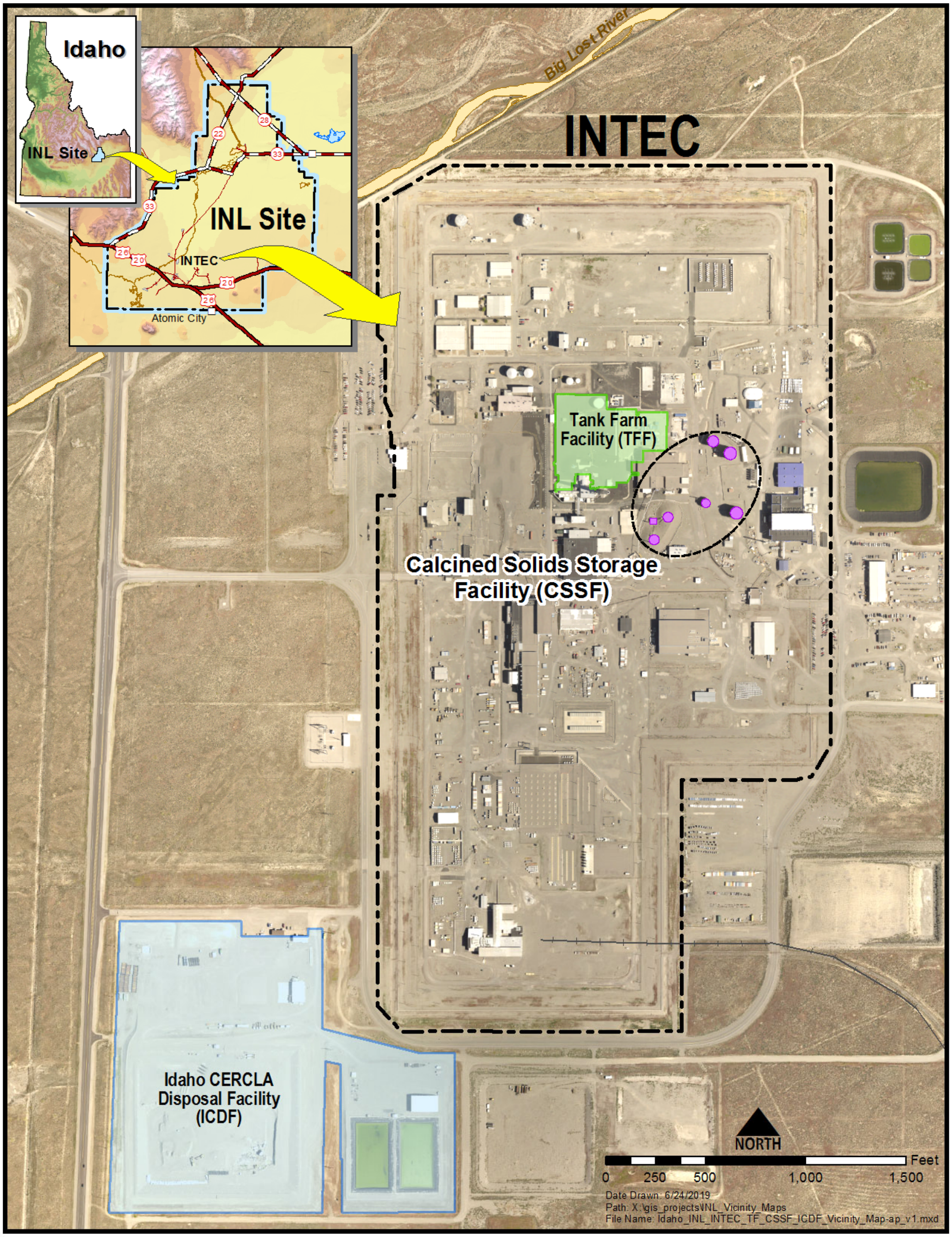

Figure 2-4. Location of the Idaho Nuclear Technology and Engineering Center at the Idaho National Laboratory Site (DOE-ID 2022a).

2.1.1.1 Population Distribution

Populations potentially affected by INL Site activities include INL Site employees, ranchers who graze livestock in areas on or near the INL Site, hunters on or near the INL Site, residential populations in neighboring communities, travelers along U.S. Highway 20/26, and visitors at the Experimental Breeder Reactor I National Historic Landmark (DOE-ID 2022a).

2.1.1.2 On-Site Populations

Eight separate facilities[20] at the INL Site comprise a total of more than 580 buildings and trailers and more than 400 other support facilities. As of October 2018, the INL and ICP contractors employed more than 6,400 personnel (DOE-ID 2022a). Approximately 41% of the total INL and ICP workforce works in Idaho Falls, Idaho, and 59% works at the INL Site (DOE-ID 2022a). The total INL and ICP workforce includes approximately 3,842 employees at INL Site locations (774 at INTEC, 520 at the Central Facilities Area [CFA], 1,015 at the Materials and Fuels Complex [MFC], 530 at the Advanced Test Reactor Complex, 296 at the Test Area North [TAN], 700 at the Radioactive Waste Management Complex [RWMC], and seven at unspecified areas) and 2,619 employees in Idaho Falls occupying numerous offices, research laboratories, and support facilities (DOE-ID 2022a). Authorized groups and visitors occasionally are escorted at the INL Site. Subcontracted employees and personnel from Idaho DEQ and EPA oversight programs also visit the area (DOE-ID 2022a).

2.1.1.3 Off-Site Populations

The INL Site is bordered by five Idaho counties: Bingham, Bonneville, Butte, Clark, and Jefferson (Irving 1993) (see Figure 2-2). Major communities include Blackfoot and Shelley in Bingham County, Idaho Falls and Ammon in Bonneville County, and Rigby in Jefferson County (see Figure 2-5). Population estimates from the U.S. Census Bureau (2023) for the counties surrounding the INL Site and the largest population centers in these counties are shown in Table 2-1. No residents are within 17 km (10 mi) of INTEC in the general southern direction of groundwater flow (DOE-ID 2022a). The community nearest to the INL Site is Atomic City, Idaho (population 29 [U.S. Census Bureau 2023]), located south of the INL Site boundary on U.S. Highway 20/26. Other population centers near the INL Site include Arco, 11 km (7 mi) west of the Site; Howe, west of the Site on U.S. Highway 22/33; and Mud Lake and Terreton on the northeast border of the INL Site. The INL Site supports no permanent residents. Figure 2-5 shows communities near the INL Site with populations greater than 2,000.

2.1.1.4 Shoshone-Bannock Tribes

The Shoshone-Bannock Tribes of the Fort Hall Indian Reservation are a federally recognized Tribal Nation and a sovereign government. The Fort Bridger Treaty of July 3, 1868, Stat. 673, secured the Fort Hall Reservation (see Figure 2-5) as the permanent homeland of the Shoshone-Bannock people (15 Stat. 673). The 1868 treaty (15 Stat. 673) also reserved aboriginal rights to these peoples that extend to areas of unoccupied land in Idaho and surrounding states, allowing access for cultural, political, and economic activities essential to the Tribes' survival. Though the INL Site is occupied land, DOE protects cultural resources and allows tribal members access to areas of cultural and religious significance at the INL Site (DOE-ID 2022c). The agreement-in-principle (DOE-ID 2022c) with the Tribes ensures that activities being conducted at the INL Site protect the health, safety, environment, and cultural resources of the Tribes and address tribal interests in DOE-administered programs. From its inception, the agreement-in-principle (DOE-ID 2022c) has been updated periodically to reflect the working relationship between the Tribes and DOE.

20. The Naval Reactors Facility is one of the eight facilities at the INL Site. The facility is operated by the DOE Naval Reactors Idaho Branch Office, separate from the DOE Idaho Operations Office, which oversees the INL and ICP missions.

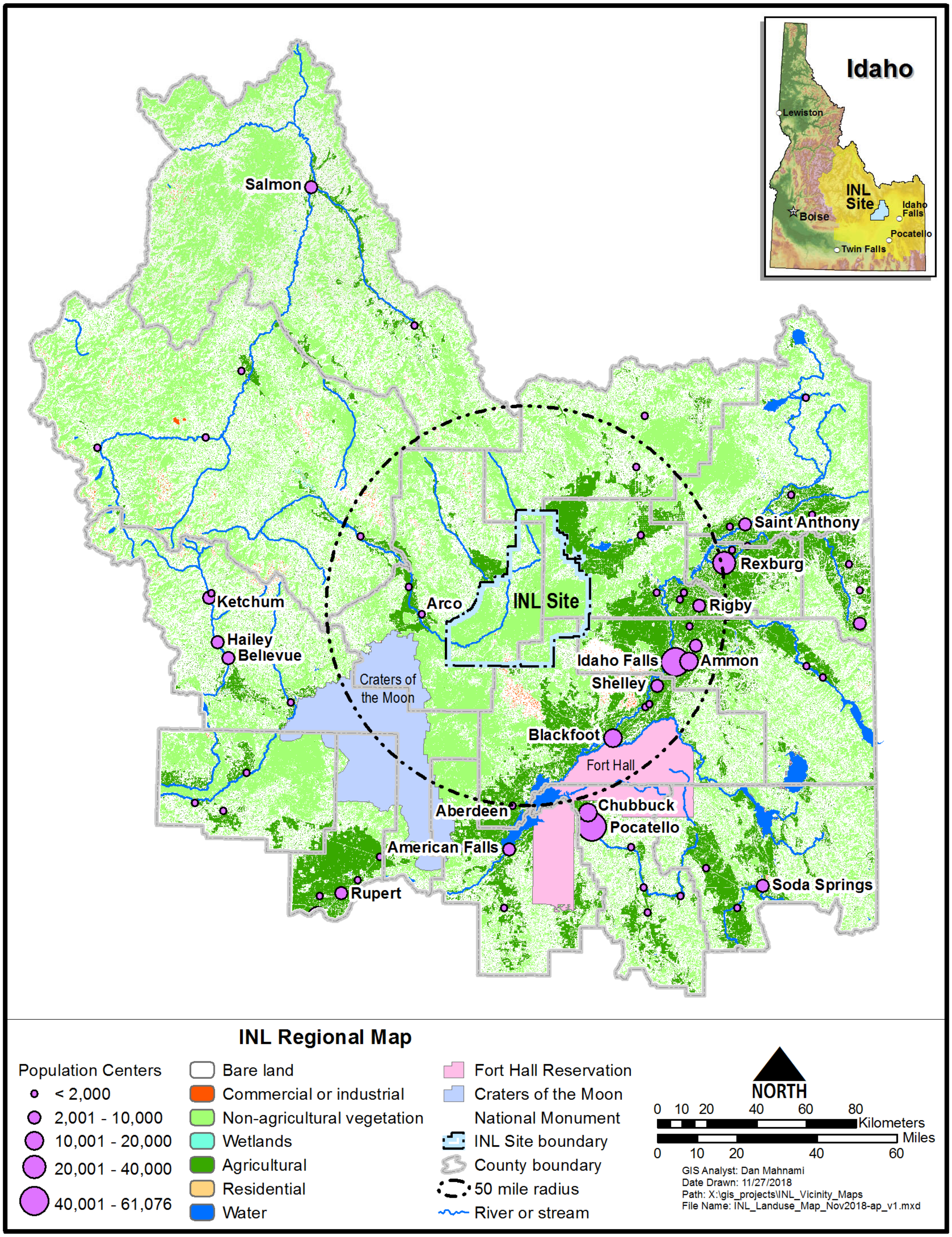

Figure 2-5. Population centers greater than 2,000 near the Idaho National Laboratory Site.

Table 2-1. Population estimates for counties and selected communities surrounding the Idaho National Laboratory Site (U.S. Census Bureau 2023).

County[a]	Major Community	Census 2010 Population	
Bingham			45,607
	Blackfoot	11,899	
	Shelley	4,409	
Bonneville			104,234
	Ammon	13,816	
	Idaho Falls	56,813	
Butte			2,891
	Arco	995	
Clark			982
	Dubois	677	
Jefferson			26,140
	Rigby	3,945	

a. All counties listed are in Idaho.

2.1.2 Land and Resource Use

The current primary use of the INL Site is to support DOE's facility and program operations, with central portions of the INL Site reserved for ICP and INL facilities and supporting infrastructure. Because the entire INL Site is a designated National Environmental Research Park, remaining land within the site boundary, which is largely undeveloped, is used for environmental research, ecological preservation, and sociocultural preservation (Irving 1993). Public highways and the Experimental Breeder Reactor I National Historic Landmark are the only portions of the INL Site with unrestricted access (DOE-ID 2022a). Additional areas of the INL Site (see Figure 2-6) are managed as follows:

- Sagebrush-Steppe Ecosystem Reserve—In 1999, DOE, the U.S. Fish and Wildlife Service, U.S. Bureau of Land Management (BLM), and Idaho Department of Fish and Game designated approximately 30,000 ha (74,130 acres) of the INL Site to conserve native ecosystem components, cultural resources, and Native American tribal values, while allowing scientific investigation of those resources present on the INL Site (BLM 2004).
- Sage-grouse Conservation Area—In 2014, DOE and the U.S. Fish and Wildlife Service (U.S. Department of Interior) signed an agreement (DOE-ID and USFWS 2014) that protects lands near sage-grouse breeding grounds and establishes a large portion of the INL Site as a conservation area.
- Grazing—The BLM administers permits for cattle and sheep grazing leases on up to 137,360 ha (340,000 acres) within the INL Site perimeter, and the U.S. Sheep Experiment Station uses 364 ha (900 acres) at the junction of U.S. Highways 28 and 33 (INL 2016).
- Hunting—DOE collaborates with the Idaho Department of Fish and Game via a memorandum of agreement (DOE-ID and IDFG 2017) to permit controlled hunts for elk and pronghorn within designated areas along northeast and northwest portions of the INL Site boundary.

INL Site operations are performed within eight primary facility areas: TAN, the Advanced Test Reactor Complex, INTEC, CFA, MFC, the Naval Reactors Facility, the Critical Infrastructure Test Range Complex, and the RWMC (DOE-ID 2022a). DOE manages each facility except for the Naval Reactors Facility, which is operated by the DOE Naval Reactors Idaho Branch Office. A 140,000-ha (340,000-acre) security and safety buffer zone, consisting of BLM-administered grazing land, is located around the centralized development area (Figure 2-6).

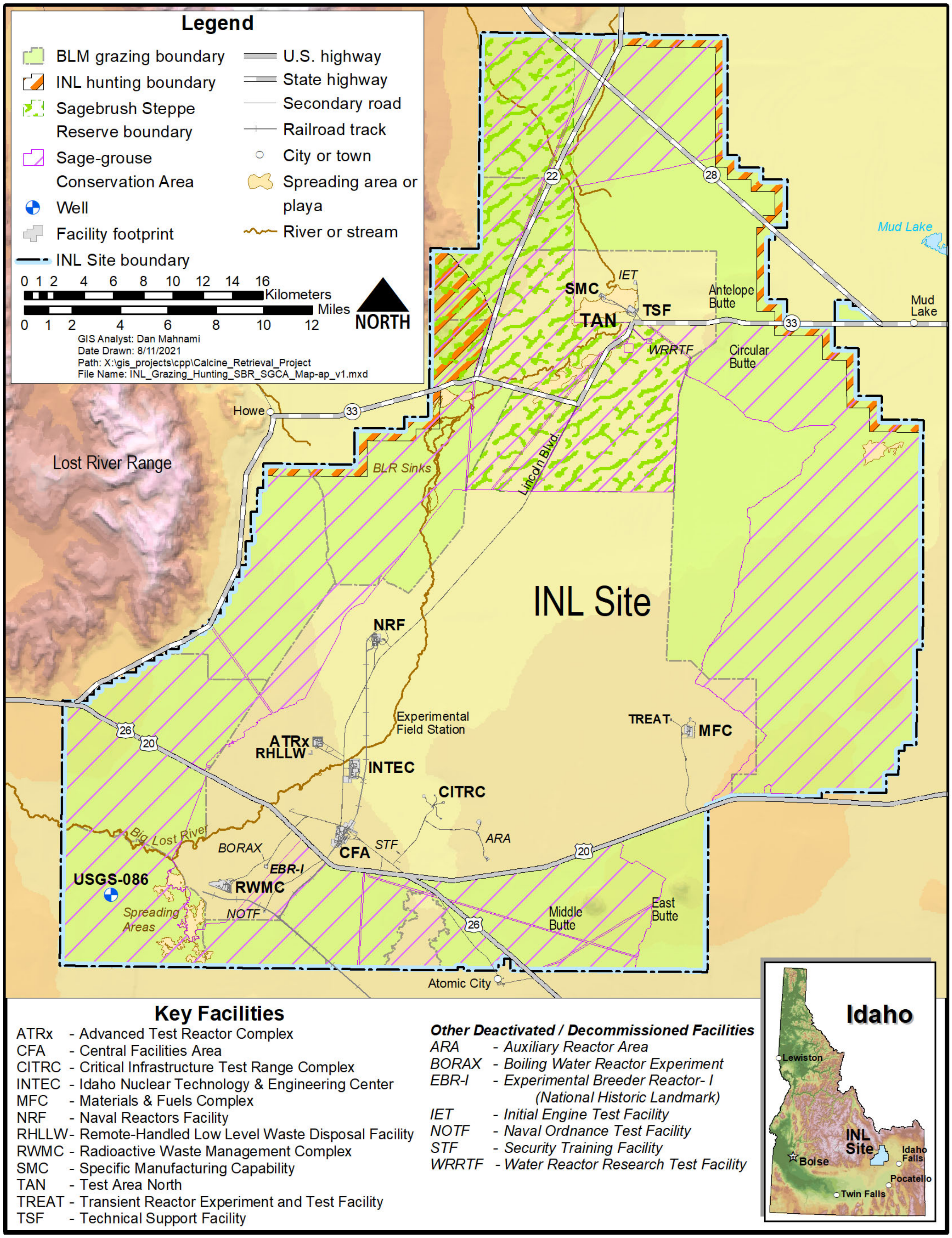

Figure 2-6. Land uses on the Idaho National Laboratory Site (DOE-ID 2022a).

2.2 Future Land and Resource Use

Future land use likely will be similar to current uses, with research facilities within INL Site boundaries and agricultural and open land surrounding the INL Site. DOE expects to retain ownership and control of the INL Site until at least 2095 and will continue to manage portions that cannot be released for unrestricted land use beyond 2095 (INL 2016). As part of the efforts related to the end state vision of the INL Site, planning assumptions for land use within, and adjacent to, the INL Site have moved toward the assumption that key areas of the INL Site, including INTEC, will remain under government control until at least 2095 and portions of the INL Site will remain under government control in perpetuity (INL 2016; DOE-ID 2022d). No new major, private developments (residential or nonresidential) are expected in areas adjacent to the INL Site. Future land use at the CSSF and INTEC for at least a 200-year period (and more likely for perpetuity) is expected to remain essentially the same as the current use: a research facility or controlled access within INL Site boundaries, especially in the major working areas.

Similarly, DOE manages INL Site groundwater resources and expects to retain management until at least 2095. Residential use of groundwater is not allowed. Agricultural use of groundwater within the boundaries of the INL Site is limited to water for livestock drawn from one well, USGS-086, located within a grazing allotment on the southwest part of the INL Site (Figure 2-6). The SRPA provides all water for industrial uses and drinking water (e.g., potable water, process water, and fire water) at the individual facilities.

2.3 Adjacent Land Uses

Adjacent land outside the boundaries of the INL Site is a combination of public and private land. The BLM controls approximately 75% of the land adjacent to the INL Site. This federally managed land provides wildlife habitat and is used for mineral and energy production, grazing, and recreation. The State of Idaho owns approximately 1% of the adjacent land and uses it for the same purposes. The remaining 24% of the land adjacent to the INL Site is privately owned and is used primarily for grazing and crop production (INL 2016). Livestock produced on land surrounding the INL Site includes sheep, beef and dairy cattle, hogs, and poultry. Major crops produced on the surrounding lands include wheat, alfalfa, barley, potatoes, oats, corn, and sugar beets (see Table 2-2). Land ownership surrounding the INL Site is illustrated in Figure 2-7.

Irrigated agriculture provides the economic base for the people of southern Idaho, and the SRPA makes possible a significant percentage of that base. It is a significant water resource to the area, and pumping for irrigation during a typical year averages 1.6 million acre-ft. Springs from the SRPA provide clean, safe water at just the right temperature for raising more than 25,000,000 lb of rainbow trout, approximately 75% of the entire United States annual production. It is estimated that more than 300,000 people depend on the SRPA for domestic and municipal water needs (IDEQ 2005). Additional hydrological information on the SRPA is provided in Subsection 2.7.5.

Table 2-2. Acreage (by county) of major crops harvested on land surrounding the Idaho National Laboratory Site (USDA 2018).

County	Alfalfa	Barley	Oats	Potatoes	Silage Corn	Sugar Beets	Wheat
Bingham	49,000 (17)	19,990 (17)	600 (14)	60,900 (16)	4,488 (12)	22,800 (16)	145,820 (12)
Bonneville	33,600 (17)	60,400 (17)	13 (12)	23,000 (14)	1,996 (12)	—	50,313 (12)
Butte	40,000 (17)	8,700 (17)	876 (12)	595 (12)	—	—	6,089 (12)
Clark	23,800 (17)	4,800 (13)	180 (12)	—	—	—	8,713 (12)
Jefferson	86,000 (17)	51,300 (16)	70 (12)	21,426 (12)	6,210 (12)	—	39,846 (12)

Note: Number in parentheses () following acreage indicates year for data.

— Indicates data unavailable

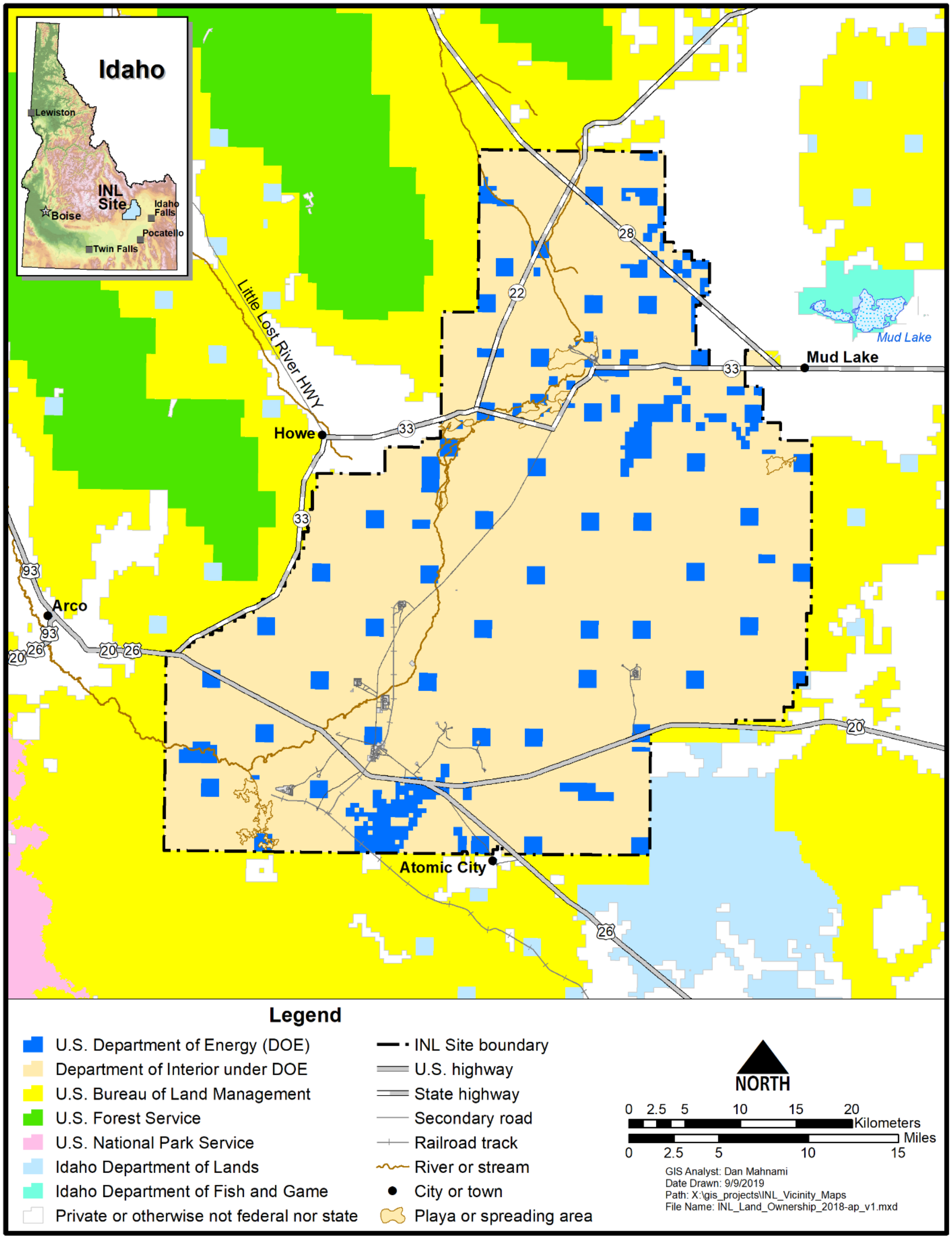

Figure 2-7. Land ownership distribution near and at the Idaho National Laboratory Site.

2.4 Meteorology and Climatology

Meteorological data have been collected periodically at several locations on and near the INL Site since 1949. Data collected from the network of observation towers support design engineering, facility operations, and operations safety at the INL Site (Clawson et al. 2018). The ESRP region is classified as an arid climate, and three distinct local-climate zones—influenced by topography and geological features—compose the INL Site. Three primary observation stations are located within each of three general climatic zones to document characteristics specific to each area: Station GRI located near INTEC (southwest climatic zone), Station MFC at MFC (southeast climatic zone), and Station SMC located at TAN (northwest climatic zone). These primary stations are more densely instrumented and equipped to take measurements at multiple levels. Station GRI at INTEC is in the "southwest climatic zone." This zone is commonly influenced by shallow down-valley winds associated with the Big Lost River channel from CFA to INTEC, as well as strong pre-frontal southwesterly winds and frequent afternoon winds (also from the southwest) generated by the diurnal heating cycle (Clawson et al. 2018). Station GRI is located approximately 1.6 km (1 mi) north of INTEC on the east side of Lincoln Boulevard. It is representative of the southwest climatic zone and, for this reason, replaces the longest-operating station at CFA as the source of tall tower data for this portion of the INL Site.

The National Oceanic and Atmospheric Administration Field Research Division of the Air Resources Laboratories conducts most of the meteorological monitoring within 80 km (50 mi) of the INL Site. The following subsections summarize climate data from *Climatography of the Idaho National Laboratory* (Clawson et al. 2018), which is a compilation of climate data collected at the INL Site from 1950 to 2015.

2.4.1 Wind

Prevailing winds at INTEC and at most locations on the INL Site are southwesterly. Station GRI is located close to the Big Lost River channel, where the station easily captures data from the influencing wind patterns of the southwest climatic zone. Data are collected at the 10-m (33-ft) and 61-m (200-ft) levels. Average wind speeds at the 10-m (33-ft) level range from 11.4 km/hour (7.1 mph) in December to 17.5 km/hour (10.9 mph) in May. At the 61-m (200-ft) level, average wind speeds range from 15.1 km/hour (9.4 mph) in January to 23.5 km/hour (14.6 mph) in May. Generally, the southwesterly wind patterns at the INL Site are less than 16.1 km/hour (10 mph). However, strong driving winds (24.1–48.3 km/hour [15–30 mph]) are most likely to occur during the summer season mid-afternoons. Thunderstorms at the INL Site also may be accompanied by microbursts, i.e., strong localized, gusty winds.

2.4.2 Air Temperature

INL Site surface air temperatures are best characterized by stations located at CFA and TAN. The average monthly air temperature for the 65-year period of record at CFA ranges from a low of -8.7°C (16.4°F) in January to a high of 20.3°C (68.6°F) during July. INL Site air temperatures may be highly variable from place to place for short periods of time. Simultaneous winter observations have occasionally shown temperature differences between CFA and TAN in excess of 25°F; spatial variation in the summer is not typically as large. Extreme temperatures have been recorded as high as 40.6°C (105°F) in July and -43.9°C (-47°F) in December.

2.4.3 Precipitation

Most daily precipitation measurements are taken from the CFA station (operating since 1950). Newer stations have been collecting data since 1993; thus, most precipitation statistics are from CFA. The average annual precipitation at CFA is 21.3 cm (8.38 in.). The highest recorded average annual amount of precipitation was 36.6 cm (14.4 in.), and the lowest amount was 7.72 cm (3.04 in.). The pronounced precipitation peak occurs in May and June, with an average precipitation for these 2 months of approximately 3 cm (1.2 in.). The greatest daily precipitation value during the period of record is 4 cm (1.64 in.). The greatest precipitation event recorded was in June 1995 when 4.65 cm (1.83 in.) accumulated in a 29-hour period. The longest precipitation event was 56.3 hours in May 1995 when 3.84 cm (1.15 in.) of precipitation was recorded. Averages for the minimum and maximum annual snowfall range from 17.3 to 151.6 cm/year (6.8 to 59.7 in./year), respectively, with an annual average of 65.8 cm (25.5 in.). The maximum average monthly snowfall is 15.5 cm (6.1 in.), occurring in January.

2.4.4 Relative Humidity

The highest relative humidity is observed in December and January, when the average mid-day relative humidity is approximately 94%. The lowest relative humidity is observed in July and August, when the average mid-day relative humidity is about 14%. For a 5-minute average of extreme values, an absolute maximum relative humidity value of 100% was observed in every month of the year during the 21-year time period from January 1994 through December 2015. The lowest 5-minute average of extreme values for relative humidity observed was 4% in July. This is indicative of the very dry summers generally experienced across the entire ESRP and at the INL Site, in particular.

2.4.5 Evaporation

Potential annual evaporation from a saturated ground surface at the INL Site is approximately 84.18 cm (33.14 in.), with 80% of the evaporation occurring between May and October. During July, the warmest month of the year, the monthly potential evaporation rate is approximately 14.40 cm (5.67 in.). Evaporation occurring during the remainder of the year is small. Actual evaporation rates are much lower than potential rates because the ground surface is rarely saturated. The average annual evapotranspiration rate by native sagebrush of the Snake River Plain is estimated at 47.80 cm (18.82 in.). From late winter to spring, precipitation is most likely to infiltrate into the ground because of the low evapotranspiration rates, except when frozen conditions preclude infiltration (Mundorff, Crosthwaite, and Kilbum 1964).

2.4.6 Special Phenomena

Several other types of meteorological phenomena occur at the INL Site, such as thunderstorms, blowing snow, and tornadoes. The INL Site may experience an average of 2 or 3 thunderstorm days during each of the summer months from June through August with considerable year-to-year variation. Thunderstorms over the INL Site are usually much less severe than what is normally experienced in the mountains surrounding the ESRP or east of the Rocky Mountains. Hence, the precipitation from many thunderstorms evaporates before reaching the ground (i.e., virga). Small hail has been observed to occasionally occur in conjunction with thunderstorms. Hail size is usually smaller than 6.35 mm (1/4 in.) in diameter; however, on very rare occasions, the diameter may range up to 19.05 mm (3/4 in.). No hail damage has ever been reported at the INL Site.

The INL Site, Upper Snake River Valley, and Idaho in general are in an area where weather patterns are not conducive to large, severe tornadoes. Weather patterns that produce severe tornadoes in the midwestern and southwestern United States (rapidly moving cold fronts or squall lines overrunning warm, moist air) are rare in Idaho. A tornado is defined as a violent local vortex in the atmosphere that reaches the ground. If the vortex does not reach the ground, it is classified as a funnel cloud. Tornadoes and funnel clouds only occur in association with thunderstorms, especially those that produce hail. Since 1949, many confirmed funnel clouds and tornadoes have been documented for the Upper Snake River Valley, with multiple sightings occurring within the boundaries of the INL Site. The wind hazard curve for the INL Site indicates that the probability of exceeding the threshold tornado wind speed (193 km/hour [120 mph]) for any given year is less than 1 in 10,000 per year (SAR-100-1).

2.5 Ecology

The INL Site represents the largest remnant of undeveloped, ungrazed sagebrush-steppe ecosystem in the Intermountain West. This ecosystem has been listed as critically endangered with less than 2% of its original coverage remaining (Noss, LaRoe, and Scott 1995; Saab and Rich 1997). In 1975, the INL Site was dedicated as one of five DOE National Environmental Research Parks. It is an outdoor laboratory used to study ecological relationships and the effects of human activities on natural systems. In addition, it provides a unique setting for scientific investigation because the public has been excluded from much of the area for the past 40 years. Ecological data collected from the Idaho National Environmental Research Park provide a basis for analyzing environmental changes over time and assessing the effect of human influence on the environment.

The following subsections describe flora, fauna, and potentially threatened, endangered, and sensitive species that may reside on the INL Site. INL Site flora and fauna research has largely been conducted by, or in conjunction with, the DOE Environmental Surveillance, Education, and Research Program. INL Site contractors comply with ecological protection programs, such as the Migratory Bird Treaty Act (16 USC 703 et seq.) and permits and agreements with regulatory agencies through company policies and procedures.

2.5.1 Flora

Long-term vegetation transects were established on the INL Site in 1950 for assessing impacts of nuclear energy research and production on surrounding ecosystems (Forman and Hafla 2018). Since the transects were established, extensive surveys of INL Site vegetation have been carried out, and the data generated compose one of the oldest and most comprehensive vegetation data sets for sagebrush-steppe ecosystems in North America.

Generally, vegetation and habitat on the INL Site can be grouped into several broad community types: juniper woodlands; native grasslands; sagebrush-steppe; low shrubs on lava; sagebrush-rabbitbrush; sagebrush-winterfat; salt desert shrub; wetlands; playas, bare ground, and disturbed areas; and lava (Anderson et al. 1996a). Vegetation is dominated by big sagebrush (*Artemisia tridentata*) with an understory of grasses and forbs. Most vegetation communities within the INL Site boundaries are dominated by various species or subspecies of sagebrush, although some communities that are dominated by saltbush (*Atriplex spp.*), juniper (*Juniperus spp.*), crested wheatgrass (*Agropyron cristatum*), and Indian ricegrass (*Achnatherum hymenoides*) are present and distributed throughout the INL Site.

The most common shrub on the INL Site is the big sagebrush (*Artemisia tridentata)*, although basin big sagebrush (*Artemisia tridentata* ssp. *tridentata*) may dominate or be codominant with Wyoming big sagebrush on sites having deep soils or sand accumulations (Shumar and Anderson 1986). Big sagebrush communities occupy most of the central portions of the INL Site. Green rabbitbrush (*Chrysothamnus viscidiflorus)* is the next most abundant shrub, and other common shrubs include winterfat (*Krascheninnikovia lanata*), spiny hopsage (*Grayia spinosa*), gray rabbitbrush (*Ericameria nauseosa*), broom snakeweed (*Gutierrezia sarothrae*), and horsebrush (*Tetradymia DC*). Communities dominated by Utah juniper (*Juniperus osteosperma*) and three-tipped sagebrush (*Artemisia tripartita*), black sagebrush (*Artemisia nova*), or both are limited to areas along the INL Site periphery, specifically on the slope of the buttes and on the foothills of adjacent mountain ranges to the northwest. Salt-desert shrub communities may be found on the sediment in the sinks and playas associated with the Big Lost River and Birch Creek. These communities are dominated by shadescale saltbrush (*Atriplex confertifolia*), Gardner's saltbush (*Atriplex gardneri*), or winterfat (*Krascheninnikovia lanata)*.

The understory grasses include natives such as thick-spiked wheatgrass (*Elymus lanceolatus*), bottlebrush squirreltail (*Elymus elymoides*), Indian ricegrass (*Achnatherum hymenoides*), needle-and-thread grass (*Hesperostipa comata*), and Sandberg's bluegrass (*Poa secunda*). Creeping wild rye (*Leymus triticoides*) and western wheatgrass (*Pascopyrum smithii*) may be locally abundant. Communities dominated by basin wild rye (*Leymus cinereus*) are common in depressions between lava ridges and in other areas having deep soils. Bluebunch wheatgrass (*Pseudoroegneria spicata*) is common at slightly higher elevations southwest and east of the INL Site.

Vegetation communities within the INL Site boundaries contain an unusually high diversity of forbs, largely due to the exclusion of livestock grazing common throughout the sagebrush-steppe region. Forb species are numerous but not abundant in many areas. Common forbs include tapertip hawksbeard (*Crepis acuminata*), Hood's phlox (*Phlox hoodia*), hoary false yarrow (*Chaenactics douglasii*), globe-mallow (*Sphaeralcea munroana*), evening primrose (*Oenothera caespitosa*), bastard toadflax (*Comandra umbellata*), and various paintbrushes (*Castilleja spp.*), buckwheats (*Eriogonum spp.*), lupines (*Lupinus spp.*), milkvetches (*Astragalus spp.*), and mustards (*Brassicaceae*) (INL 2016).

Two large-scale vegetation projects were completed during the 5 years between 2006 and 2011. The first project completed was a new plant community classification and vegetation map of the INL Site; the second addressed patterns of non-native species invasion and factors driving plant biodiversity. The plant community classification and vegetation map was completed in 2011 and represents the most thorough statistical classification of plant communities and detailed map of vegetation classes ever prepared for the INL Site (Forman, Hafla, and Blew 2013). The updated vegetation map resulted in 26 vegetation classes for the INL Site (Shive et al. 2011). A poster-size map detailing INL Site vegetation can be viewed in Appendix F of Shive et al. (2011).

Several studies of plant rooting depths have been conducted at the INL Site. Studies of plant uptake of radionuclides at the INL Site have focused primarily on (1) determining if deep-rooted plants are a mechanism for waste pit intrusion and subsequent uptake of radionuclides and (2) analyzing inventories of radionuclides in aerial portions of plants. Aerial portions of plants are important because they can potentially transport subsurface contaminants through dispersal of leaves, consumption by herbivores, use by birds as nesting materials, and wildfire.

One study comparing radionuclide uptake by crested wheatgrass (rooting depth 75 cm [29.5 in.]) with that by Russian thistle (rooting depth 100 to 500 cm [39 to 197 in.]) showed higher radionuclide concentrations in the deeper-rooted species (Arthur 1982). Examples of other deep-rooting species are rabbitbrush and sagebrush. General examples of shallow-rooting plant types are grasses and annual forbs.

Reynolds and Fraley (1989) found that the roots of big sagebrush extended to a depth of 225 cm (89 in.), green rabbitbrush to a depth of 190 cm (75 in.), and Great Basin wild rye to a depth up to 200 cm (79 in.). Maximum lateral spread of the roots of both big sagebrush and Great Basin wild rye was 100 cm (39 in.) and occurred at a depth of 40 cm (16 in.). In addition, studies indicate root penetration for sodar (cultivated variety of streambank wheatgrass) and crested wheatgrass at the INL Site ranged from 40 cm (16 in.) to 160 cm (63 in.) (Markham 1987).

Revegetation rates of big sagebrush on disturbed sites have been studied near the INL Site, as well as in other areas with similar environments. While seeding establishment may begin immediately following a fire or other disturbance, recovery of big sagebrush was found to take several decades before mature sagebrush stands were able to return and dominate a site (Harniss and Murray 1973; Wambolt, Walhof, and Frisina 2001; Welch and Criddle 2003). Baker (2006) suggests that full recovery of Wyoming big sagebrush, a common subspecies of big sagebrush on the INL Site, generally requires 50 to 120 years. Environmental factors such as soil conditions may have an impact on the rate of reestablishment (Chambers 2000).

2.5.2 Fauna

The INL Site supports a variety of wildlife, including small mammals, birds, reptiles, and a few large mammals. Fish species reported on the INL Site are limited to the Big Lost River during years when water flow is sufficient. However, periods of drought and upstream water diversion for agricultural and flood-prevention purposes has severely restricted the flow of the Big Lost River on the INL Site, thereby restricting the presence of native fish species. Similarly, the Great Basin spadefoot toad (*Spea intermontane*), the INL Site's only reported resident amphibian, is limited by water flow in the Big Lost River. Reptiles include five species of snake, three species of lizards, and the western skink (*Eumeces skiltonianus*) (INL 2016). Fish species observed in the Big Lost River on the INL Site include rainbow trout (*Oncorhynchus mykiss*), mountain whitefish (*Prosopium williamsoni*), brook trout (*Salvelinus fontinalis*), and the shorthead sculpin (*Cottus confusus*) (Overton, Grove, and Johnson 1976).

A total of 219 vertebrate species have been recorded on the INL Site. Vertebrate species include six species of fish, one amphibian, nine reptiles, 164 birds, and 39 mammals. An additional nine fish, five amphibian, five reptile, 19 bird, and 14 mammal species are considered as possibly occurring at the INL Site because portions of their range overlap the INL Site area or they have been reported within 30 km (18 mi) of the INL Site. However, no verified observations of these species have been reported on the INL Site (INL 2016). Several vertebrate species present on the INL Site are considered sagebrush-obligate species, meaning that they rely upon sagebrush for survival. These species include the sagebrush sparrow *(Amphispiza nevadensis*), Brewer's sparrow *(Spizella breweri*), northern sagebrush lizard *(Sceloporus graciosu*s), sage grouse *(Centrocercus urophasianu*s), and pygmy rabbit *(Brachylagus idahoensi*s) (Reynolds et al. 1986).

Studies have been performed on burrowing characteristics of small mammals such as ground squirrels, deer mice, and voles (Arthur, Grant, and Markham 1983; Markham 1987; Reynolds and Laundre 1988). Results of the studies indicate that burrows are no deeper than 1.4 m (4.6 ft) at the INL Site.

2.5.3 Threatened, Endangered, and Sensitive Species

The DOE Environmental Surveillance, Education, and Research Program conducts ecological research, field surveys, and NEPA evaluations regarding ecological resources on the INL Site. Particular emphasis is given to threatened and endangered species and species of special concern identified by the U.S. Fish and Wildlife Services and Idaho Department of Fish and Game (DOE-ID 2018a).

One species that occurs or may occur on the INL Site, the yellow-billed cuckoo (*Coccyzus americanus*), has been categorized as threatened under the "Endangered Species Act of 1973" (16 USC 35 et seq.). The yellow-billed cuckoo is a riparian-obligate species and is primarily associated with willow-cottonwood riparian forest. One occurrence near the INL Site boundary has been documented; however, the observed yellow-billed cuckoo was most likely utilizing the area as a stop-over habitat during an exploratory trek or migration (Hughes 2017; DOE-ID 2018a). Several species have been removed from the INL Site list of designated species based on the limited likelihood they would occur on the INL Site, or they were removed from the endangered list. For example, the peregrine falcon (*Falco peregrinus*), gray wolf (*Cani lupus*), and bald eagle (*Haliaeetus leucocephalus*) were removed from endangered list in 1999, 2011, and 2007, respectively. Though the bald eagle was officially removed from the endangered list, federal protection is maintained under the "Bald and Golden Eagles Protection Act" (16 USC 668). The State of Idaho continues protection for the peregrine falcon (IDAPA 13.01.06), and wolf populations are monitored to ensure long-term survival (IDFG 2002).

Several INL Site species are classified by the U.S. Forest Service Intermountain Region (Region 4) and BLM as sensitive species and special status species. The State of Idaho has protections in place for multiple bird, mammal, and reptile species. The yellow-billed cuckoo (*Coccyzus americanus*) and greater sage-grouse (*Centrocercus urophasianus*) are classified as Tier 1 "species of greatest conservation need," which represents species with the most critical conservation needs (IDFG 2017).

Greater sage-grouse populations have decreased significantly in the past 100 years due to degraded rangelands, encroachment of urban and agricultural development on grouse-preferred habitat, and invasion of non-native species (e.g., cheat grass [*Bromus tectorum*]) (INL 2016). Thus, the INL Site is a refuge for the greater sage-grouse because of its relatively undisturbed sagebrush habitat. In 2014, DOE voluntarily entered into a Candidate Conservation Agreement with the U.S. Fish and Wildlife Service (DOE-ID and USFWS 2014) and committed to implement conservation measures and objectives to avoid or minimize threats to sage-grouse and its habitats. This was the first such agreement signed in Idaho for sage-grouse. The Candidate Conservation Agreement establishes population and habitat triggers that, if tripped, will initiate a prescribed response by DOE and the U.S. Fish and Wildlife Service.

Among the many INL Site wildlife species, bats are recognized as an important resource. Fourteen confirmed bat species reside in the State of Idaho; nine of those species are documented to occupy the INL Site during some part of the year (Whiting and Bybee 2011). Because of recent emerging threats (i.e., white-nose syndrome and wind-energy development) that have caused bat population to decline drastically, several bat species detected at the INL Site are considered for different levels of protection by the U.S. Fish and Wildlife Service, BLM, Western Bat Working Group, and other conservation organizations. Though there is currently no regulatory driver, DOE is voluntarily implementing a bat protection plan (DOE-ID 2018b) to monitor and assess bat populations and habitats on the INL Site. Data collected under the plan will be used to ensure that bat populations at the INL Site and adjacent areas (e.g., Craters of the Moon National Monument) are protected and that DOE makes informed land management decisions.

2.6 Geology, Seismology, and Volcanology

The following subsections present geology, seismology, and volcanology at the INL Site, INTEC, and CSSF area.

2.6.1 Geology

The INL Site is located on the west-central part of the ESRP, which is commonly divided into two regions: a northwest-trending depositional basin in the western region of the ESRP and a northeastern-trending volcanic plain in the eastern region. These two regions mark the path of the Yellowstone hotspot across southern Idaho, from its beginning near the Nevada-Oregon-Idaho border to the current location under Yellowstone National Park. The ESRP is the product of plains-style volcanism due to low-viscosity magma that flowed laterally from vents. Overlapping flows from one or more vents produced shield formations across the plain, followed by minor fissure-fed flows into low areas between shields. Underlying the western region of the ESRP is a sequence of Tertiary and Quaternary volcanic rocks and sedimentary interbeds that extend beyond the depth of 3,048 m (10,000 ft). The uppermost part of the volcanic rocks consists mainly of basalt flows with rhyolitic ash-flow tuffs composing the lowermost part (Anderson, Liszewski, and Cecil 1997).

At least 121 basalt-flow groups and 102 sedimentary interbeds underlie the INL Site above the effective base of the SRPA and range in age from approximately 200,000 to 4 million years before present (Anderson et al. 1996b; Anderson 1991). Basalt flow groups comprise one or more distinct basalt flows deposited from the same magma source during a single eruptive event that produces a geological fingerprint, such as paleomagnetic properties, potassium contents, and natural-gamma emissions (Anderson, Liszewski, and Cecil 1997).

Most of the basalt flows under the INL Site consist of vesicular to dense olivine basalt. Individual flows generally range from 3 to 15 m (10 to 50 ft) thick and are locally interbedded with scoria and thin layers of sediment. Significant changes in a flow's thickness are often related to changes in the lithology of the flow or the change in a flow's margin, in which the flow appears as a lobe of basalt. The lithologic changes that can influence a flow's thickness are the existence of pyroclastic deposits on or within the flow, making it more susceptible to the effects of erosion. During periods of volcanic quiescence, sedimentary interbeds accumulated on the ancestral land surface. Across the INL Site, sedimentary interbeds can be as thick as 15 m (50 ft) and consist of clay, silt, sand, and gravel, with the occasional scoria and basalt rubble zones (Anderson, Liszewski, and Cecil 1997).

2.6.1.1 INTEC Lithology Description

The stratigraphy underlying INTEC and the surrounding area is based on work presented by Anderson (1991) and studies to support remedial actions at INTEC under the Final Record of Decision Idaho Nuclear Technology and Engineering Center, Operable Unit 3-13, Idaho National Engineering and Environmental Laboratory (CERCLA OU 3-13 ROD) (DOE-ID 1999a) and the Record of Decision for Tank Farm Soil and INTEC Groundwater, Operable Unit 3-14 (CERCLA OU 3-14 ROD) (DOE-ID 2007a). In the early 1990s, research evaluating the stratigraphy underlying INTEC and the surrounding area established that the subsurface, below the surficial alluvium, contained up to 23 basalt flows and 15 to 20 sedimentary interbeds (Anderson 1991). A subsequent monitoring well and tracer study, Phase I Monitoring Well and Tracer Study Report for OU 3-13, Group 4, Perched Water (DOE-ID 2003a), conducted from 2000 through 2002, further defined INTEC lithology and perched water in the INTEC subsurface.

The tracer study well data set includes core textural features, geophysical logs, paleomagnetic measurements, basalt geochemistry, and potassium-argon age dating data that were collected from perched water monitoring wells and select SRPA wells (DOE-ID 2003a). The study identified six lithologic marker units below the surficial alluvium that are commonly seen throughout the INTEC subsurface: surficial alluvium, 34-m (110-ft) interbed, high K_2O basalt flow, 43-m (140-ft) interbed, middle massive basalt flow, and 116-m (380-ft) interbed. A plan view and detailed north-south cross section from the data set are shown in Figures 2-8 and 2-9.

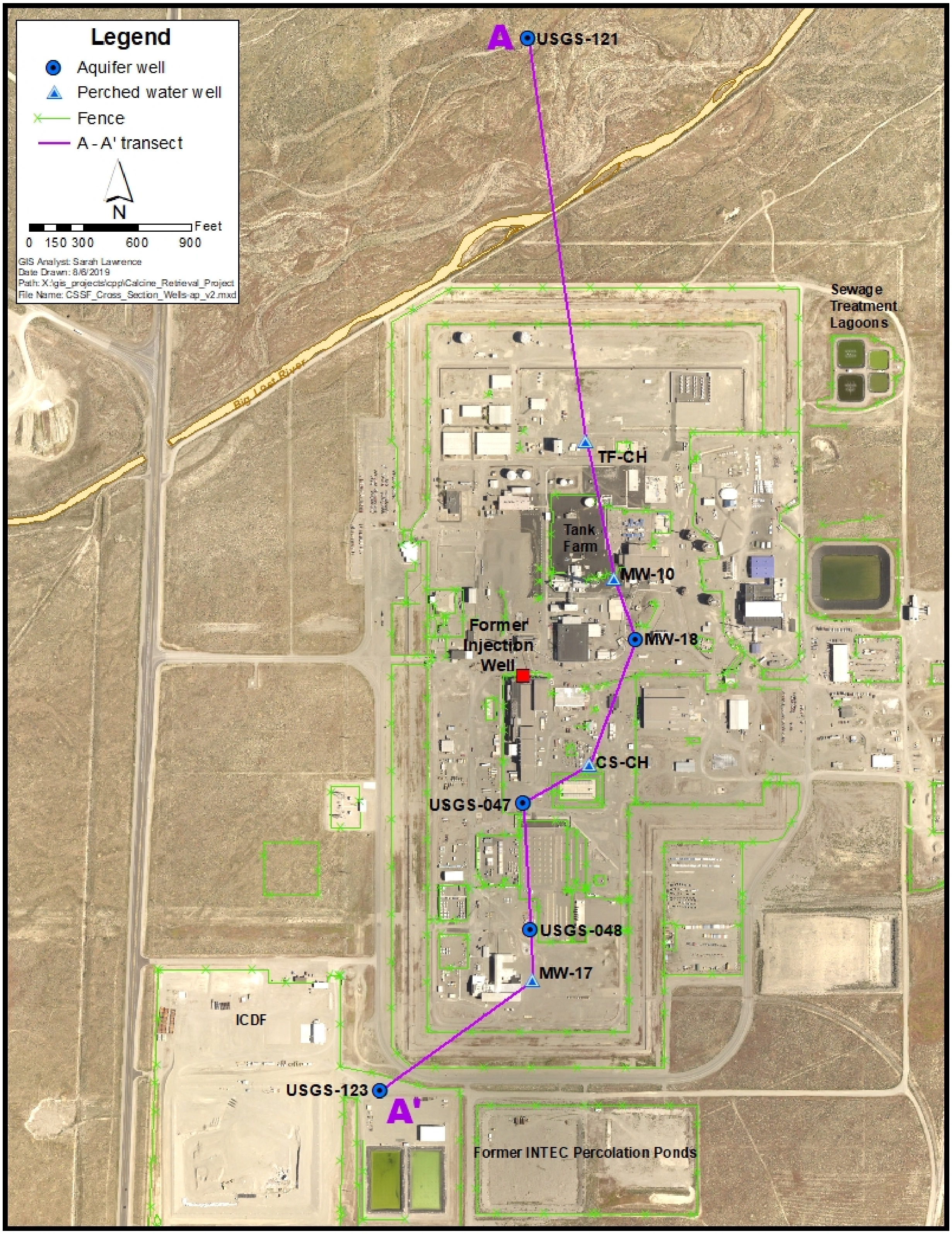

Figure 2-8. Plan view of north-south geologic cross section at the Idaho Nuclear Technology and Engineering Center.

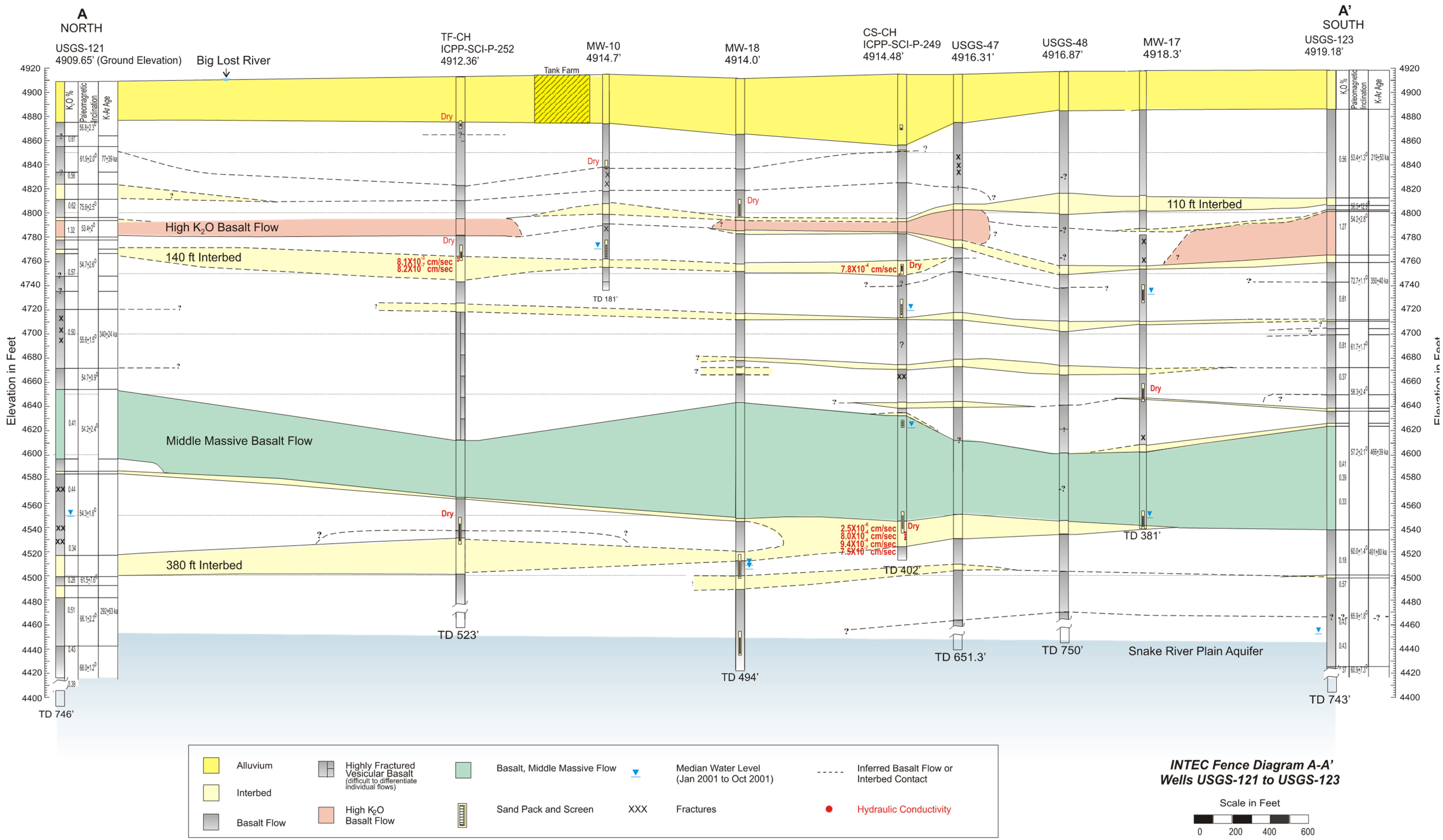

Figure 2-9. North-south geological cross section at the Idaho Nuclear Technology and Engineering Center (DOE-ID 2003a).

The six INTEC lithologic marker units, as described in the 2003 tracer study (DOE-ID 2003a), are summarized as follows:

- ***Surficial alluvium*—**Surficial sediments are composed of alluvial, fluvial, and eolian silt, sand, and gravel deposited on top of the uppermost basalt flow. Surficial sediment thickness ranges from approximately 6.7 to 18.6 m (22 ft to 61 ft).
- ***110-ft interbed*—**The closest sedimentary unit to the surface is an interbed found between 30.5 and 30.6 m (100 and 120 ft) below land surface (bls) with a thickness ranging from 1 to 7.6 m (3 to 25 ft). Consisting primarily of sand, silt, and clay and having a high content of potassium-containing minerals, the interbed at this layer is distinguished on natural-gamma logs. Due to various recharge sources, downward migration of infiltrating water is impeded by this layer, consequently forming bodies of perched water throughout INTEC. Within the CSSF area, this interbed has an average thickness of 1.2 m (4 ft) to 2.4 m (8 ft).
- ***High K_2O basalt flow*—**The high K_2O basalt flow is identified by its higher natural-gamma signature found on geophysical logs. This flow is typically encountered between 33.5 and 39.6 m (110 and 130 ft) bls and is described as a very hard, non-porous flow with an influence on perched water movement. This flow is typically only seen in wells on the westerly side of INTEC and contains approximately two to three times higher K_2O content than the average of other flows.
- ***140-ft interbed*—**The second most distinguished interbed from the surface is typically found between 43 and 46 m (140 and 150 ft) bls. The continuity of the 43-m (140-ft) interbed is less well defined than the 34-m (110-ft) interbed because most shallow perched water monitoring wells target the 34-m (110-ft) interbed for completion and were not drilled deep enough to encounter the 140-ft interbed. However, like the 34-m (110-ft) interbed, perched water is associated with this layer throughout INTEC. Within the CSSF area, this interbed has an average thickness that ranges from 1.8 m (6 ft) to 2.1 m (7 ft).
- ***Middle massive basalt flow*—**The middle massive basalt flow is one of the thickest and most massive flows in the INTEC unsaturated (vadose) zone. Typical thickness of the unit is approximately 30.5 m (100 ft). In addition to the massive structure of the basalt, the low-moisture content distinguishes the layer on neutron logs.
- ***380-ft interbed*—**The deepest interbed in the unsaturated zone is found between 98 and 128 m (320 and 420 ft) bls. It ranges in thickness from 1.8 and 8.2 m (6 and 27 ft). The thickness thins to the south of INTEC and consists of sand, silt, and clay layers, with a small amount of gravel. Perched water has been associated with this interbed, although not as regularly as the 34-m (110-ft) interbed.

2.6.1.2 CSSF Lithology Description

Lawrence and Jolley (2018) compiled geophysical logs for the CSSF PA/CA (DOE-ID 2022a) to construct a general lithologic cross section and lithological column of INTEC, as well as to provide a description of the lithology near the CSSF. Results from the 2003 tracer study (DOE-ID 2003a) optimized the placement of four additional CERCLA monitoring wells: ICPP-2018, ICPP-2019, ICPP-2020, and ICPP-2021. These wells, along with four historically installed wells (55-06, MW-10, MW-18, and MW-20), provide details for inference of thickness and location of the sedimentary interbeds in the unsaturated zone near the CSSF (see Figure 2-10). The CSSF vicinity cross section (see Figure 2-11) was developed using unit marker depths and thicknesses identified in each monitoring well log (natural gamma and driller logs). However, due to variable periods of active volcanism or dormancy within the ESRP, subsurface lava flow extents and thickness, as well as interbed thickness and uniformity, are inconsistent; thus, the interbed connections between wells in the cross section are inferred. The extent of the unsaturated zone is represented on the cross section at 143.2 m (470 ft) bls, based on the average 2018 water level measurements from the three SRPA monitoring wells (MW-18, ICPP-2020, and ICPP-2021).

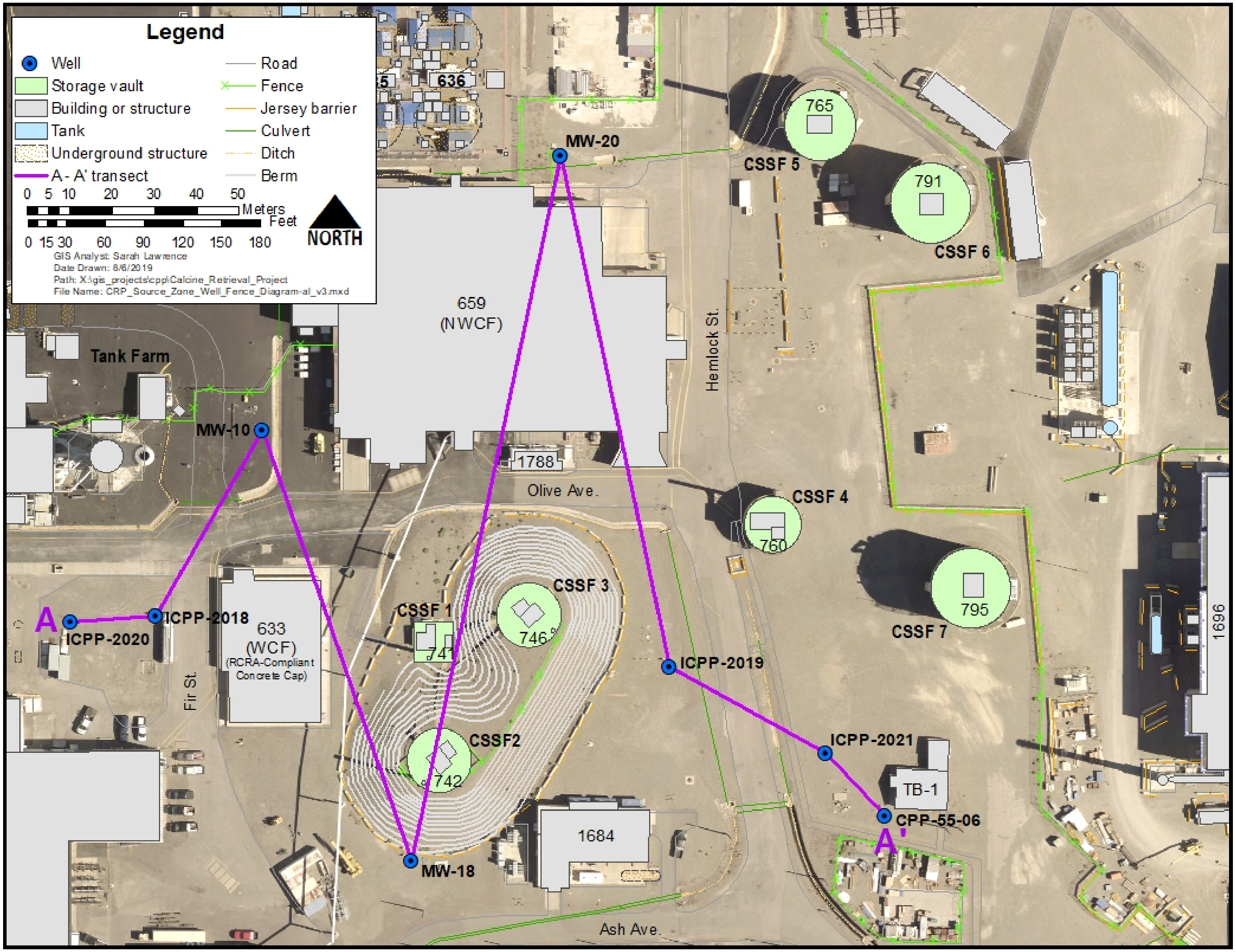

Figure 2-10. Calcined Solids Storage Facility plan view of Transect A-A′.

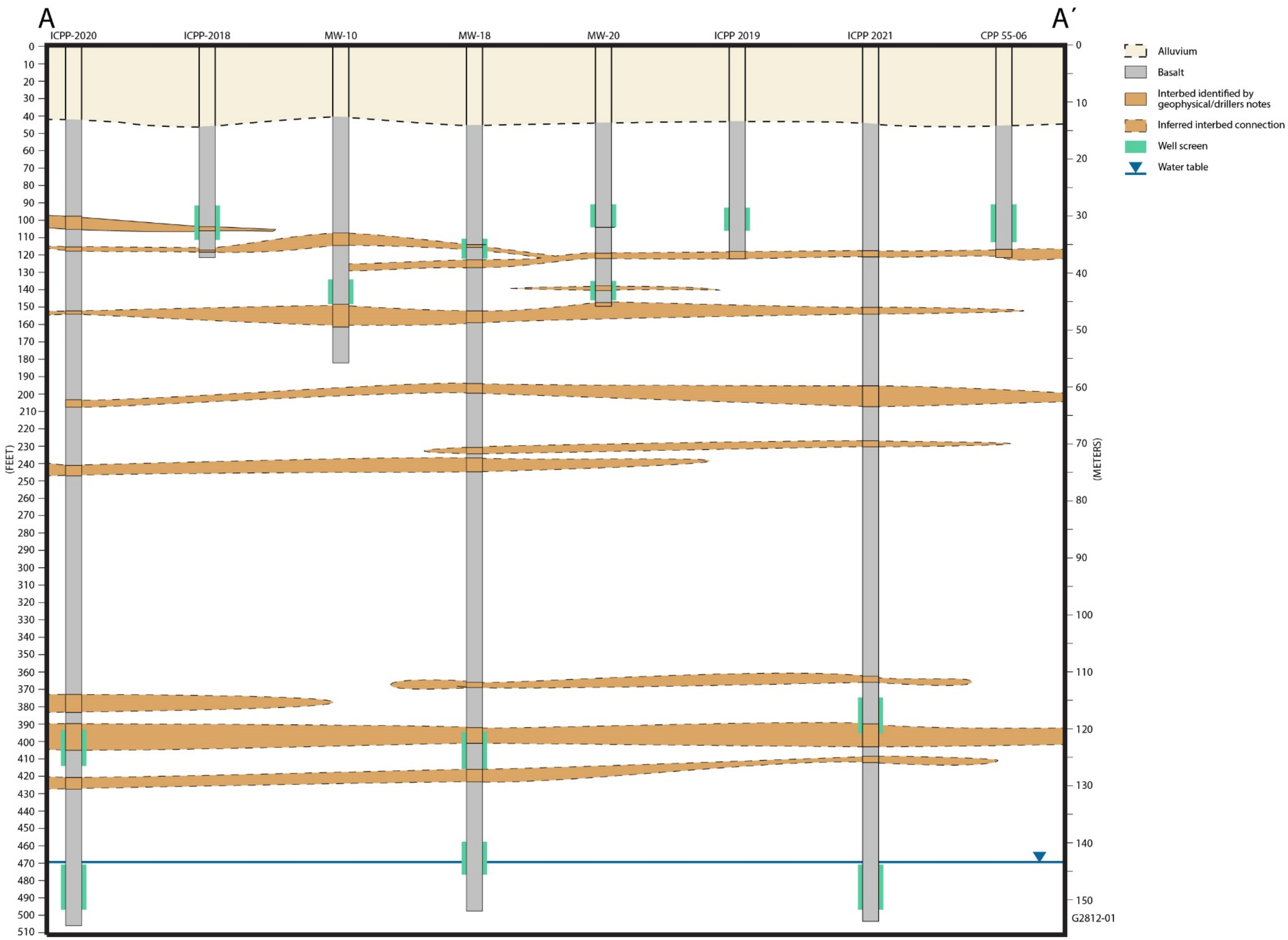

Figure 2-11. East-to-west Calcined Solids Storage Facility cross section of Transect A-A′.
Note: Cross section and distance between wells are not to scale.

The CSSF vicinity generalized lithologic column was constructed from land surface to 154.5 m (507 ft) bls by averaging the eight selected monitoring wells' unit marker thickness and depth. The CSSF area is characterized by approximately 14 m (46 ft) of surficial sediments deposited over nine basalt flows separated by eight discontinuous sedimentary interbeds. Basalt is typically massive with vertical fracturing common at the upper and lower surfaces of the flow units. Interbeds range in thickness from approximately 0.9 to 3.6 m (3 to 12 ft), with an average thickness of approximately 1.8 m (6 ft). Basalt flows are typically 7.6 to 10.6 m (25 to 35 ft) thick; however, the middle massive basalt flow is approximately 38 m (125 ft) thick. The aquifer is encountered approximately 143.2 m (470 ft) bls (see Figure 2-12). Additional information regarding the CSSF vicinity cross section and general lithologic column can be found in Lawrence and Jolley (2018).

2.6.2 Seismology

The seismically active Intermountain seismic and Centennial tectonic seismic belts surround the ESRP (SAR-100-1). A historical catalog has been compiled from regional seismic networks for earthquakes within a 322-km (200-mi) radius of the INL Site that have magnitudes 2.5 and greater and occurred from 1884 to 2018 (DOE-ID 2022a) (see Figure 2-13). Seismic activity in eastern Idaho is concentrated along the Intermountain seismic belt, which extends more than 1,287 km (800 mi) from southern Arizona through eastern Idaho to western Montana (Irving 1993). The Centennial tectonic seismic belt extends from central Idaho into southwestern Montana (Irving 1993). This distribution of epicenters indicates that the Snake River Plain is devoid of earthquakes relative to the active areas surrounding it.

The largest recorded seismic event in the Intermountain seismic belt occurred on August 17, 1959, and had a magnitude of 7.5 (SAR-100-1). It was located near Hebgen Lake in southwestern Montana, approximately 160 km (100 mi) from the INL Site (Figure 2-13). Numerous aftershocks, including one as large as a magnitude of 6.3, shook the region for several years. Although this earthquake was felt at the INL Site, no significant damage occurred. The largest recorded earthquake in the Centennial tectonic belt occurred on October 28, 1983, and had a magnitude of 7.3. The earthquake resulted from slippage along a normal range-front fault, with relative movement down to the west. The epicenter for this event was along the western flank of Borah Peak in the Lost River Range, approximately 80 to 115 km (50 to 70 mi) northwest of INL Site facilities. No significant damage occurred at the INL Site. The only large earthquake reported within the ESRP was located at Shoshone, Idaho, in 1905. However, after extensive review of historical records, this Magnitude 6 earthquake is believed to have occurred within the Basin and Range province near the Idaho/Utah border (WCFS 1996).

Under requirements in DOE O 420.1C Chg 3, "Facility Safety," the INL Site has a seismic network for monitoring earthquake activity on and around the ESRP to support DOE operations. The INL Seismic Monitoring Program (managed by BEA, the DOE's Office of Nuclear Energy subcontractor) operates 32 seismic stations and 30 strong-motion accelerograph sites on and near the INL Site (Payne et al. 2016). The seismic stations are used to determine the time, location, and size of earthquakes occurring near the INL Site. Strong-motion accelerographs are used to record strong ground motions from local moderate or major earthquakes. The seismic network has compiled earthquake epicenters within a 161-km (100-mi) radius of the INL Site occurring from 1972 to 2013. Figure 2-14 shows earthquake epicenters with magnitudes greater than 2.5 within a 161-km (100-mi) radius of the INL Site from 1897 to 2018, using data from the regional network of seismic monitors. Monitoring indicates that the ESRP is relatively seismically inactive when compared to surrounding Basin and Range regions (Payne et al. 2016). Recent ongoing activity is likely associated with nearby volcanic processes (Payne et al. 2016).

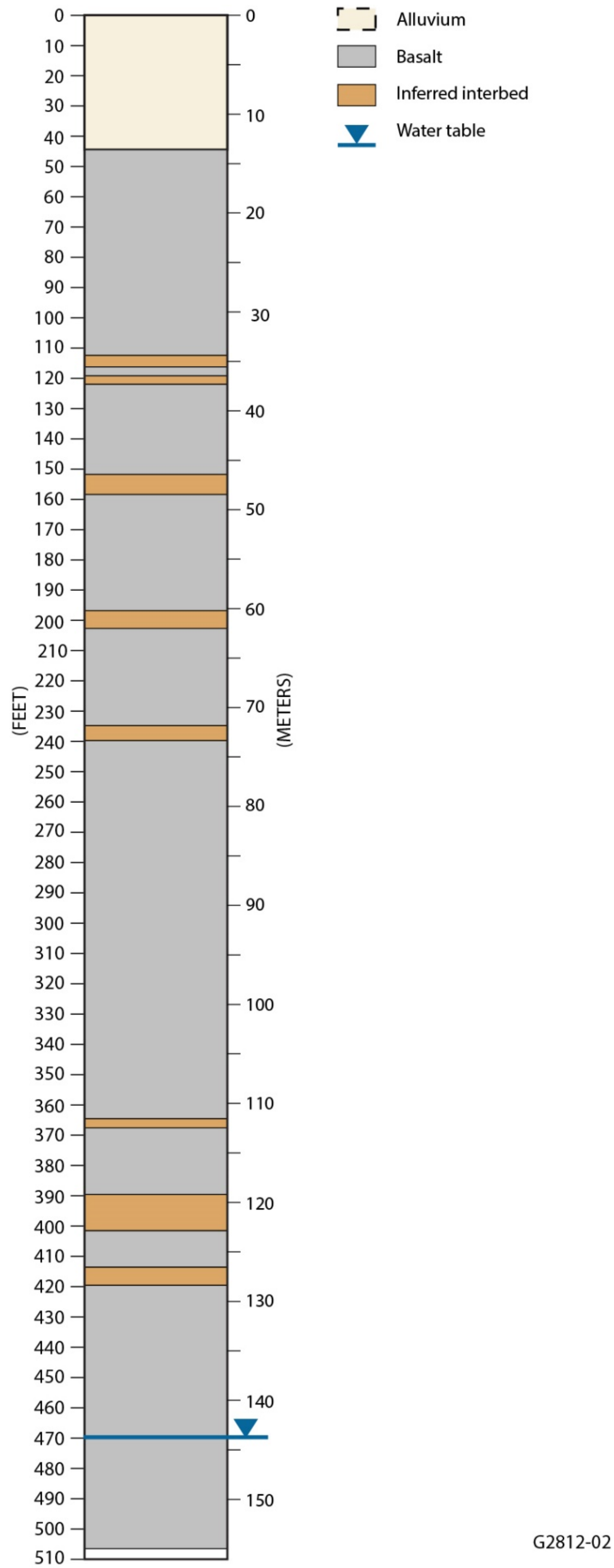

Figure 2-12. General lithology column near the Calcined Solids Storage Facility inferred from the east-west cross-section stratigraphy (Lawrence and Jolly 2018).

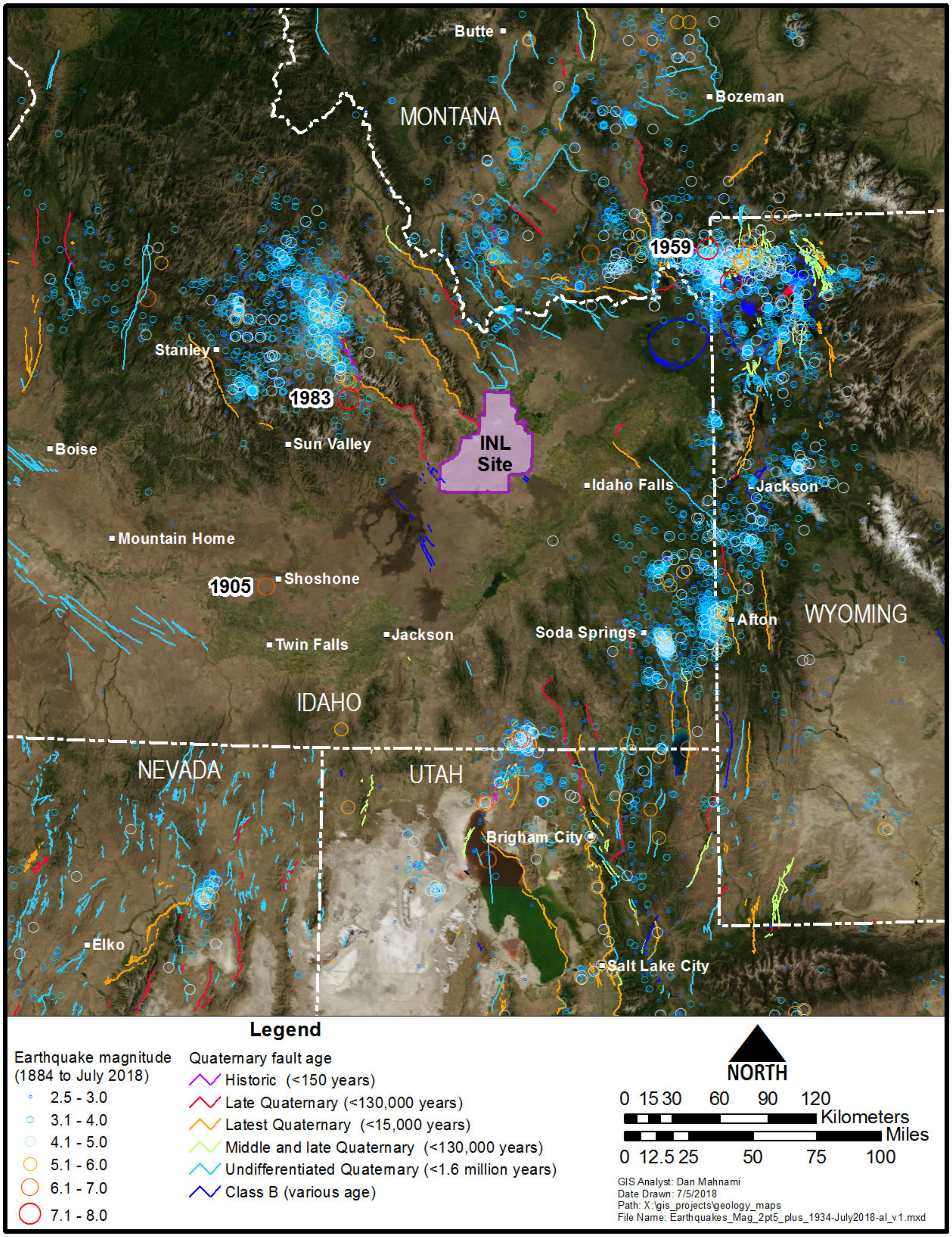

Figure 2-13. Map showing earthquake epicenters with magnitudes greater than 2.5 within a 322-km (200-mi) radius of the Idaho National Laboratory Site from 1884 to 2018.

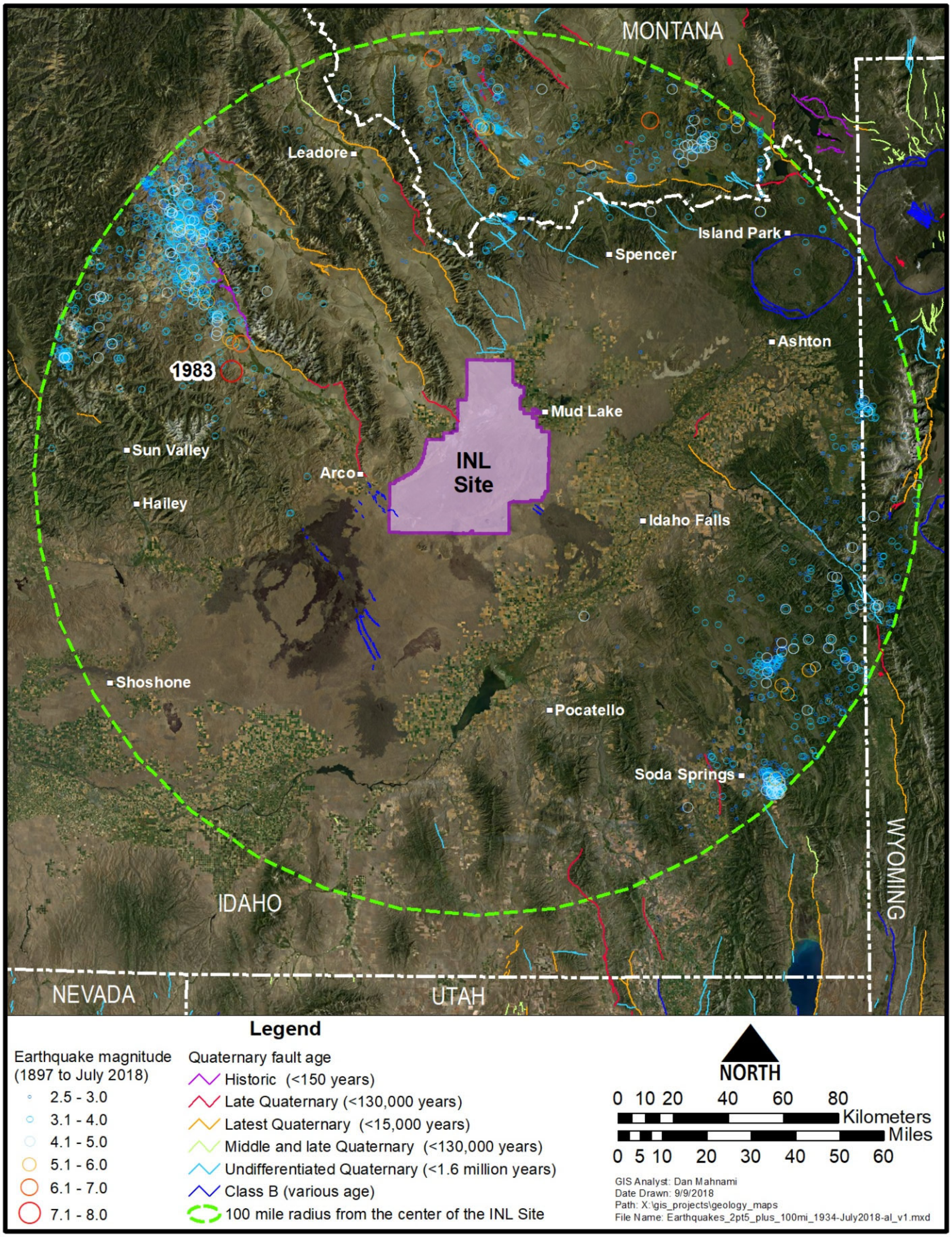

Figure 2-14. Map showing earthquake epicenters with magnitudes greater than 2.5 within a 161-km (100-mi) radius of the Idaho National Laboratory Site from 1897 to 2018.

Because the seismically active Intermountain and Centennial seismic belts surround the ESRP and several Quaternary faults are located near the western boundary of the INL Site, seismic hazard assessments were completed for all facility areas at the INL Site (WCFS 1996; INEEL 2000; Payne et al. 2002). Seismic hazard evaluations were conducted using a probabilistic methodology that incorporates the most up-to-date region- and site-specific geologic, seismologic, and geotechnical information for the INL Site. These assessments quantitatively estimated peak ground motions that INL Site facilities may experience from nearby large-magnitude earthquakes. The site-specific geological, seismological, and geotechnical data used in INL Site ground motion evaluations were provided to the U.S. Geological Survey (USGS), which also used these data to develop the national seismic hazard maps. The seismic design levels in the form of peak ground accelerations are obtained from the USGS seismic hazard maps (USGS 2018) for the International Building Code (ICC 2015), which is currently used for nonnuclear facilities at INTEC.

Design criteria for the CSSF are based on design codes, standards, regulations, and DOE orders existing at the time of construction; each CSSF was constructed at various times between 1959 and 1981. The CSSF Safety Analysis Report (SAR-105) states that the CSSF storage vault structures must meet Performance Category-3 design criteria for natural phenomena hazards. A 2003 assessment concluded that the concrete vaults meet criteria of the 2000 International Building Code (SAR-105), and thus, the CSSF storage vaults, as constructed, are sufficiently robust to expect no radiological release in the event of a Performance Category-3 earthquake.

2.6.3 Volcanology

The INL Site is in a region of Pleistocene and Holocene volcanic activity typically characterized by nonviolent, effusive basalt lava flows (Hackett and Smith 1992). Explosive rhyolite volcanism occurred beneath the INL Site 4 million to 7 million years ago, forming calderas now buried beneath basalt lava flows. In the region immediately surrounding the INL Site, the youngest lava flow erupted approximately 4,100 years ago from Hell's Half Acre lava flow southeast of the INL Site. Within INL Site boundaries, the most recent lava flow—the Cerro Grande flow—occurred 13,000 years ago, near the southern boundary (Hackett, Pelton, and Brockway 1986).

Renewed explosive rhyolite volcanism at the INL Site is very unlikely (INEL 1990). Geological and geochronological data indicate an eastward progression of silicic volcanism. The mantle plume or hotspot assumed responsible for the volcanism now lies beneath Yellowstone National Park. Past patterns of volcanism suggest that future volcanism at the INL Site within the next 1,000–10,000 years is very improbable (INEL 1990), and the two most likely sources of future basalt flows on the INL Site are the Arco-Big Southern Butte and Lava Ridge-Hell's Half Acre rift zones (see Figure 2-15).

Most of the INL Site is underlain by a 0- to 1-km-thick (0- to 0.6-mi-thick) sequence of Tertiary and Quaternary basalt lava flows and interbedded sediments. Based on drill-hole information, regional mapping along the margins of the ESRP, and geophysical information, the basalt and sediment sequence is underlain by an older section (up to several kilometers thick) of late Tertiary rhyolitic volcanic rock. These two volcanic sequences are a consequence of the passage of the Yellowstone mantle plume (hotspot) through the INL Site area of the ESRP in the late Tertiary period. The Tertiary rhyolitic volcanic rocks were erupted 6.5 million to 4.3 million years ago when the hotspot resided beneath the INL Site area (Pierce and Morgan 1992). These volcanic rocks are composed mostly of ash-flow tuffs erupted during large, violent explosive episodes and large rhyolitic lava flows. These rocks are analogous to the ash flow tuffs and lava flows that erupted from calderas in the Yellowstone Plateau from 2.0 million to 0.6 million years ago.

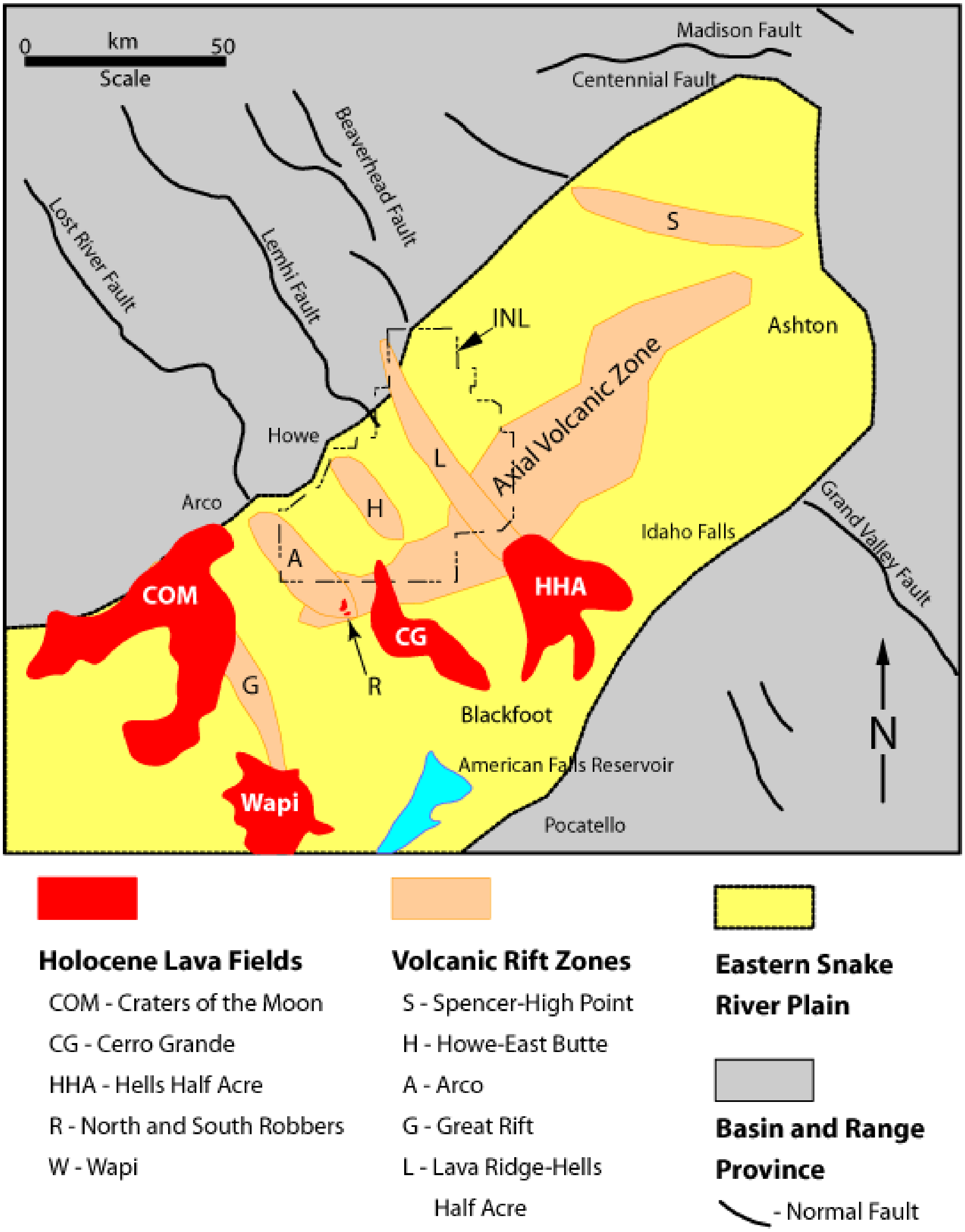

Figure 2-15. Map showing the locations of volcanic rift zones and Holocene basalt lava fields in southeastern Idaho (modified from SAR-100-1, Figure 1-11).

These types of large-scale explosive eruptions can occur only directly over the position of the mantle hotspot because large inputs of heat into the lower and middle crust are required to generate such large volumes of rhyolitic magma. Because the hotspot is now situated beneath Yellowstone National Park, there is no possibility for recurrence of this type of volcanic activity in the INL Site area (DOE-ID 2022a). Regional extension of the crust and residual heat in the upper mantle after passage of the hotspot has resulted in basaltic magmas that have risen to the surface and erupted onto the subsiding ESRP. Basaltic eruptions in the INL Site area began at about 4 Ma, soon after passage of the hotspot, and have occurred as recently as 2,100 years ago along the Great Rift.

Basalt vents on the ESRP include broad, low-relief shield volcanoes, small spatter cones, and spatter ramparts along eruptive fissures. Lava fields related to single vents range in surface area from 2 to 400 km^2 (0.7 to 154 mi^2) and in volume from 0.05 to 7 km^3 (0.01 to 1.7 mi^3) (Kuntz, Covington, and Schorr 1992). Volcanic vents are not randomly distributed on the ESRP; they are concentrated in northwest-trending linear zones known as volcanic rift zones (see Figure 2-15).

In addition, vents are concentrated in a northeast-trending zone, known as the Axial Volcanic Zone, along the central axis of the ESRP (Hackett and Smith 1992). The Axial Volcanic Zone is a constructional highland caused by more voluminous magma output along the axis of the ESRP.

Based on radiometric age determinations of basalt lava flows, the Arco Volcanic Rift Zone north of Big Southern Butte was active between 200 and 700 ka (Kuntz, Covington, and Schorr 1992). The Cerro Grande and North and South Robbers flows (10,500 to 12,000 ka) near Big Southern Butte occur at the intersection of the Arco Volcanic Rift Zone and the Axial Volcanic Zone. Except for volcanism along the Great Rift, all of the Holocene volcanic fields within the ESRP occur along the Axial Volcanic Zone (see Figure 2-15). Recurrence of volcanism in the ESRP is more likely along the Great Rift or the Axial Volcanic Zone. The estimated volcanic-recurrence interval near INTEC is 45 ka and the corresponding annual eruption probability is 6E-05 per year (Hackett, Smith, and Khericha 2002).

Available geologic map data and geochronometry of basalt lava flows at the INL Site suggest the minimum volcanic-recurrence intervals of 1E-04 to 1E-05/year for the Axial Volcanic Zone and the Arco and Lava Ridge-Hell's Half Acre volcanic rift zones (DOE-ID 2022a). Therefore, probabilistic risk of basalt lava inundation or intrusion-related ground disturbance is estimated to be less than 1E-05/year (i.e., 1 chance in 100,000/year) for the southern INL Site. The probability of significant impact from volcanic phenomena (e.g., growth of new rhyolite domes on the ESRP or tephra falls thicker than 8 cm [3 in.] from non-Snake River Plain vents) is estimated to be less than E-05/year because of the combined effects of great distance, infrequency, low volume, and topographic or atmospheric barriers to dispersal of tephra on the INL Site (DOE-ID 2022a).

INTEC is unlikely to be impacted by tephra-, gas-, or dike-induced ground deformation because it is not within a mapped volcanic rift zone. The chief volcanic hazard at INTEC is inundation by lava flows from source vents outside INTEC boundaries, as suggested by Hackett, Smith, and Khericha (2002). The Volcanism Working Group (INEL 1990) estimated the probability of inundation of INTEC by basalt flows to be E-06 (i.e., 1 chance in 1,000,000) per year.

2.6.4 Future Changes and Site Stability

Holocene surficial geology and archaeology suggest that fluvial and eolian deposition and tectonic subsidence in the INL Site area have been in approximate net balance for at least the past 10,000 years (DOE-ID 2022a). A reversal of the long-term, regional pattern of ESRP subsidence, sedimentation, and volcanism into an erosional rather than a depositional regime would require major changes from the Holocene tectonic or climatic configuration of the ESRP. Worldwide geologic evidence indicates that the Quaternary epoch (approximately the past 2 million years) has been a time of major climatic fluctuations.

During colder and wetter periods, glaciers occupied high-elevation areas. Lowland areas such as the ESRP received thick, widespread loess blankets. If the future ESRP climate were to become warmer and more arid, the probable consequences would be decreased vegetation and increased eolian transport of fine-grained sediment, mainly as longitudinal dunes of fine sand.

Future climatic fluctuations on the ESRP, to either colder/wetter or warmer/drier conditions, are not expected to erode the INTEC land surface (DOE-ID 2022a). Quaternary geologic and Holocene archaeological data suggest the INL Site area will probably continue its long-term history of regional subsidence and net accumulation of sedimentary and volcanic materials, although sedimentation patterns on the ESRP will change in response to future climate fluctuations (DOE-ID 2022a).

Surface soil erosion at INTEC could occur because of faulting and uplift, but this erosion would involve a major change in the Quaternary tectonic configuration of the ESRP. Therefore, this scenario is improbable within the next 10,000 years, considering:

- The regional seismicity and tectonic history of the INL Site area
- The absence of Quaternary tectonic faults on the ESRP near INTEC
- The long response time for significant erosion to occur because of protracted faulting and uplift.

In summary:

- During the past 4 million years, the ESRP and INTEC area have undergone regional subsidence, basaltic volcanism, and fluvial and eolian sedimentation. Erosion has not been a significant process on the ESRP.
- Surficial and subsurface geologic data indicate that the INTEC area has both subsided and accumulated basalt lava flows and sediments at an average rate of 0.03 cm (0.01 in.)/year. Significant uplift or erosion has not interrupted this long-term trend.
- Lava inundation or magma intrusion associated with volcanism from the nearby Arco Volcanic Rift Zone is improbable, considering the volcanic history of the area. Lava inundation or magma intrusion would not likely result in the release of radionuclides to the environment.

2.7 Hydrology

The following subsections discuss surface water, infiltration, and groundwater at the CSSF and vicinity.

2.7.1 Surface Water

Surface water sources at INTEC and near the CSSF include (1) the Big Lost River (when flowing), (2) ponded rain and snowmelt, (3) the CERCLA storm water evaporation pond (construction completed October 2003), (4) the Idaho CERCLA Disposal Facility (ICDF) evaporation ponds (operations began September 2003), (5) the INTEC Sewage Treatment Plant, and (6) the former INTEC percolation ponds. The CERCLA storm water evaporation pond, ICDF evaporation ponds, and INTEC Sewage Treatment Plant are lined ponds managed by DOE and, as such, are not considered a likely source of infiltration; thus, they are not discussed further in this section. The former INTEC percolation ponds also are not considered a likely source of infiltration; they were relocated 3.2 km (2 mi) west of INTEC. Further discussion on the former INTEC percolation ponds is provided in Subsection 2.7.4.

The Big Lost River is the major surface water feature on the INL Site (Cahn et al. 2006). At its closest point, the channel of the Big Lost River lies within 30 m (100 ft) of the northwest corner of INTEC (Cahn et al. 2006). The Big Lost River is an intermittent stream that flows north through the INL Site to its terminus at the Big Lost River sinks, where the water either infiltrates into the ground or evaporates (Cahn et al. 2006). The stretch of the Big Lost River on the INL Site is ephemeral (INL 2010) with no recreational or consumptive uses of the water (e.g., irrigation, manufacturing, or drinking).

The Big Lost River flows are regulated at Mackay Reservoir, which is located approximately 64.4 km (40 mi) northwest of the INL Site (DOE-ID 2022a). Flows that reach the INL Site may be diverted at the INL Site diversion dam to the flood control “spreading areas” (INL 2010) located southwest of RWMC (see Figure 2-16). Water that is not diverted to the spreading areas continues to flow northeastward across the INL Site in a shallow channel to the Big Lost River sinks (Ostenaa and O’Connell 2005).

When it is flowing, the Big Lost River constitutes a source of recharge to perched water and the SRPA (Cahn et al. 2006). However, this recharge is limited to the immediate vicinity of the Big Lost River and is not a significant source of recharge near the CSSF (DOE-ID 2022a). Flow in the Big Lost River depends on winter snowpack conditions and whether controlled releases are occurring from Mackay Reservoir (Cahn et al. 2006). Figure 2-17 shows the mean daily discharge (flow rate) for the Big Lost River at the Lincoln Boulevard gaging station at INTEC during the period 1984–2018. The river flowed for extended periods during the wet years of 1984 to 1987 and again during 1995–2000 (Cahn et al. 2006). During periods of below-normal precipitation/snowpack, the Big Lost River may remain dry for several years in a row. Most recently, the Big Lost River was dry from May 2000 through May 2005 as the result of a 5-year drought. However, due to above-normal snowpack during 2004–2006, brief periods of river flow occurred at INTEC during the spring and early summer of 2005, 2006, 2009, 2010, 2011, and 2012. The river did not flow past INTEC during 2013–2016, but because of above-normal snowpack, large flows in the river occurred during 2017 and 2018. When flow does occur, peak flows are typically in June and July due to snowmelt, and there is often no flow in the river during the winter months.

The U.S. Bureau of Reclamation’s *Big Lost River Flood Hazard Study Idaho National Laboratory, Idaho* (Ostenaa and O’Connell 2005) delineated the approximate 100-year historical discharge records of the Big Lost River and augmented the data with a paleoflow analyses; the study modeled precipitation-derived flows onto the INL Site and the potential to reach proposed facility locations. One objective of the Ostenaa and O’Connell (2005) study was to construct a soil chronosequence of the Big Lost River on the INL Site to define the paleohydrologic and geomorphic setting of the river. The study determined the geomorphic setting to be more than 10,000 years, thus allowing a direct interpretation of the potential risk of flooding in unregulated or unmodified channel systems. The refined paleohydrologic bounds reduced uncertainty and calibrated the study’s models for unregulated flood frequency analyses and prediction of inundation extent (INL 2010). The 100-year flood peak flow is estimated to be 3,072 cfs (87 cms) with the 1,000-year flood peak flow estimated to be 4,626 cfs (131 cms) at the INL Site diversion dam (Ostenaa and O’Connell 2005). Based on historical maximum channel discharges between the INL Site diversion dam and near INTEC at Lincoln Boulevard, the river’s course alluvium channel bed can contribute to a flow rate loss up to 15%, thus potentially reducing flow reaching INTEC (Ostenaa and O’Connell 2005).

INTEC flood inundation maps, with various scenarios of flow infiltration and Lincoln Boulevard culvert flow, indicate the north-northwest end of INTEC to be more susceptible to flooding. Figure 2-18 depicts the flooding extent of the river and water depth with a flow of 150 cms (5,295 cfs); the CSSF vicinity is projected to have a water depth up to 0.5 m (1.6 ft) from a 40-hour flow (Ostenaa and O’Connell 2005). The Ostenaa and O’Connell (2005) study constructed probabilistic mean water elevations for a 100-, 500-, 2,000-, and 10,000-year flood. With an average riverbank elevation northwest of INTEC of 1,497 m (4,912 ft), the 500- and 2,000-year flood mean water elevation for the Big Lost River is calculated to be 1,497.54 m (4,913.18 ft) and 1,497.67 m (4,913.62 ft), respectively (Cahn et al. 2006; Ostenaa and O’Connell 2005). The southeast corner of the NWCF, directly adjacent to the CSSF vicinity, is modeled to be dry through a 500-year flood and have a 2,000-year flood mean water elevation of 1,496.98 (4,911.35 ft) (Ostenaa and O’Connell 2005).

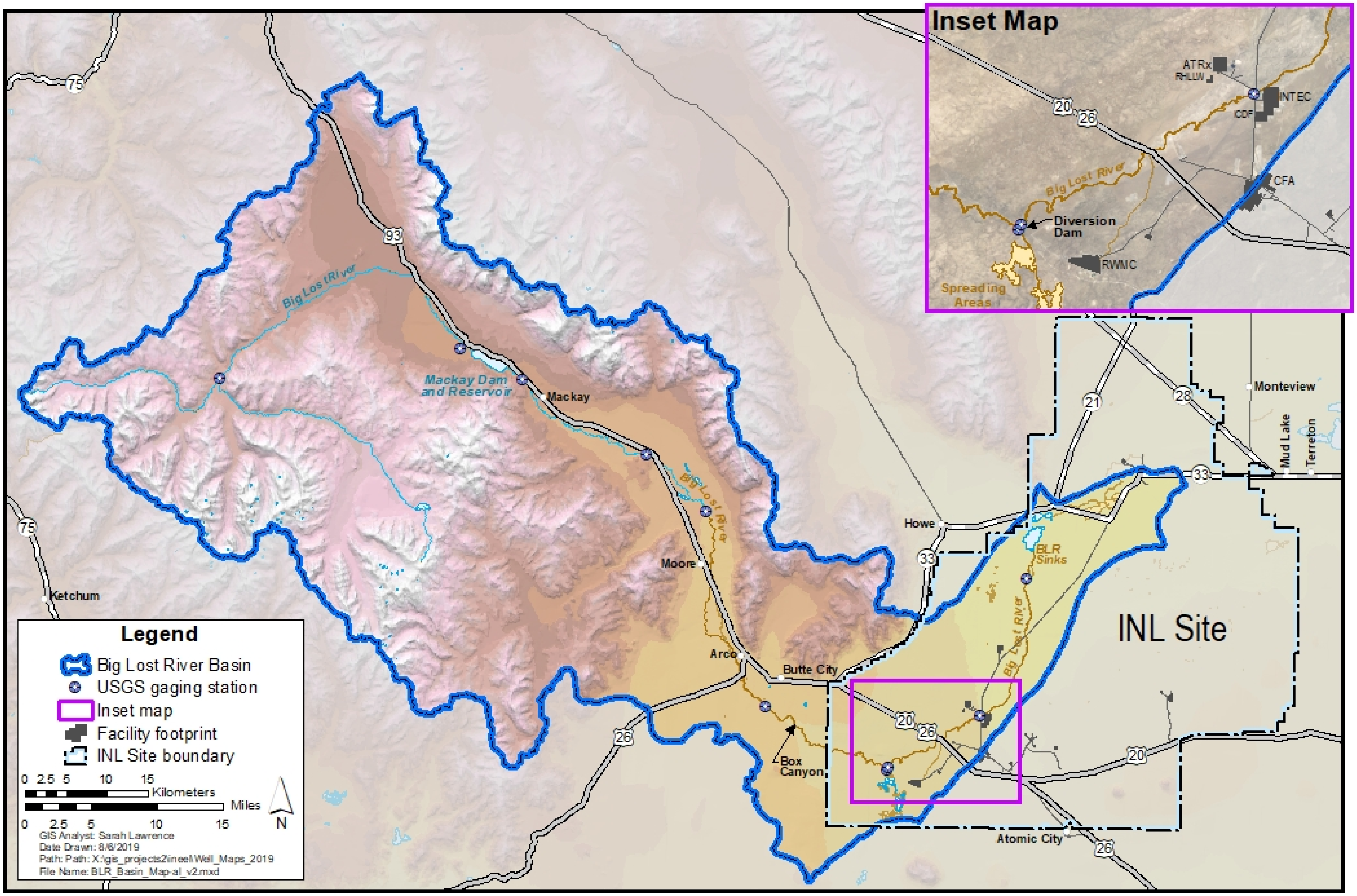

Figure 2-16. Location of Mackay Dam and the Idaho National Laboratory Site diversion dam on the Big Lost River.

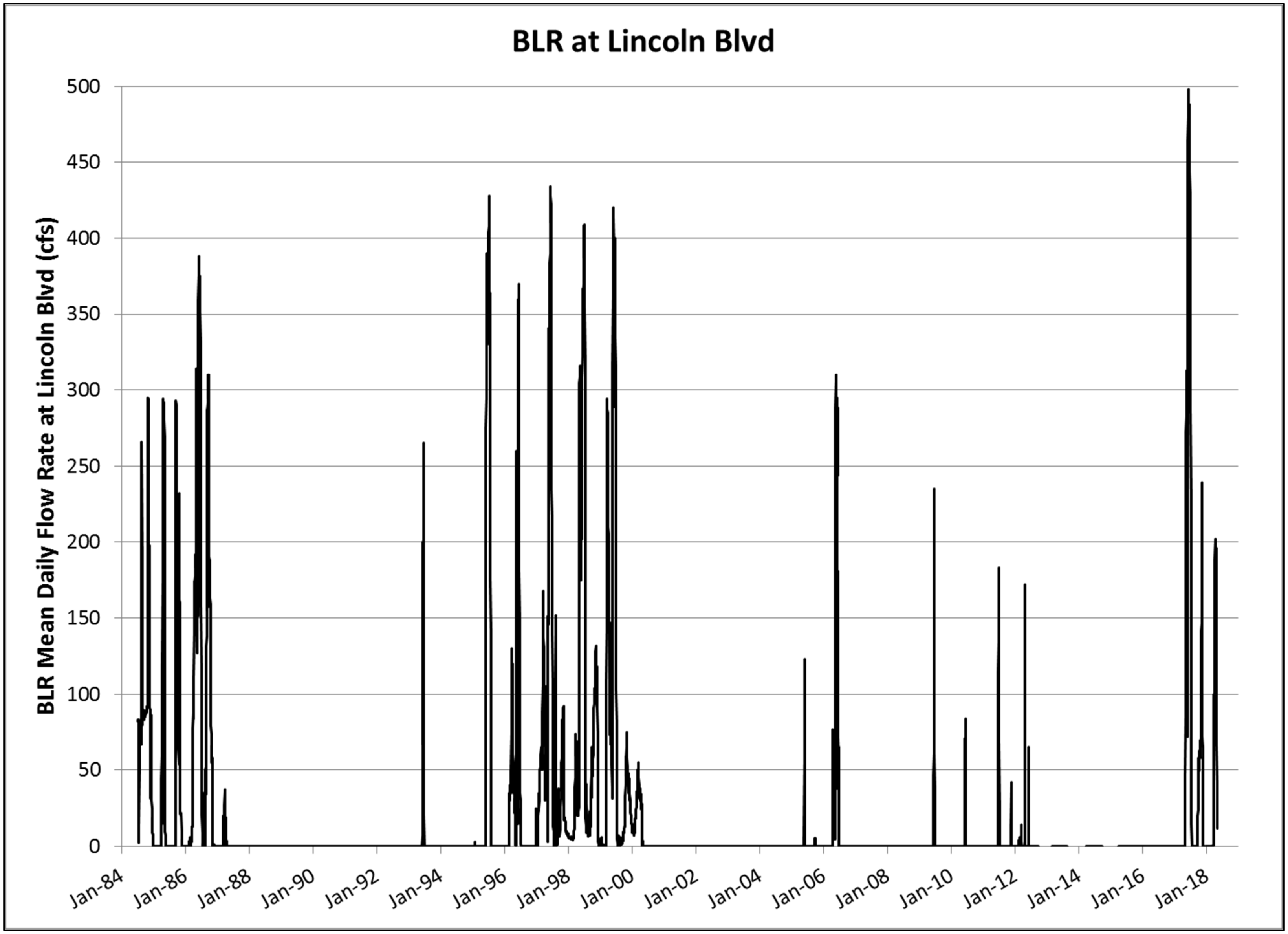

Figure 2-17. Big Lost River hydrograph for 1984–2018.

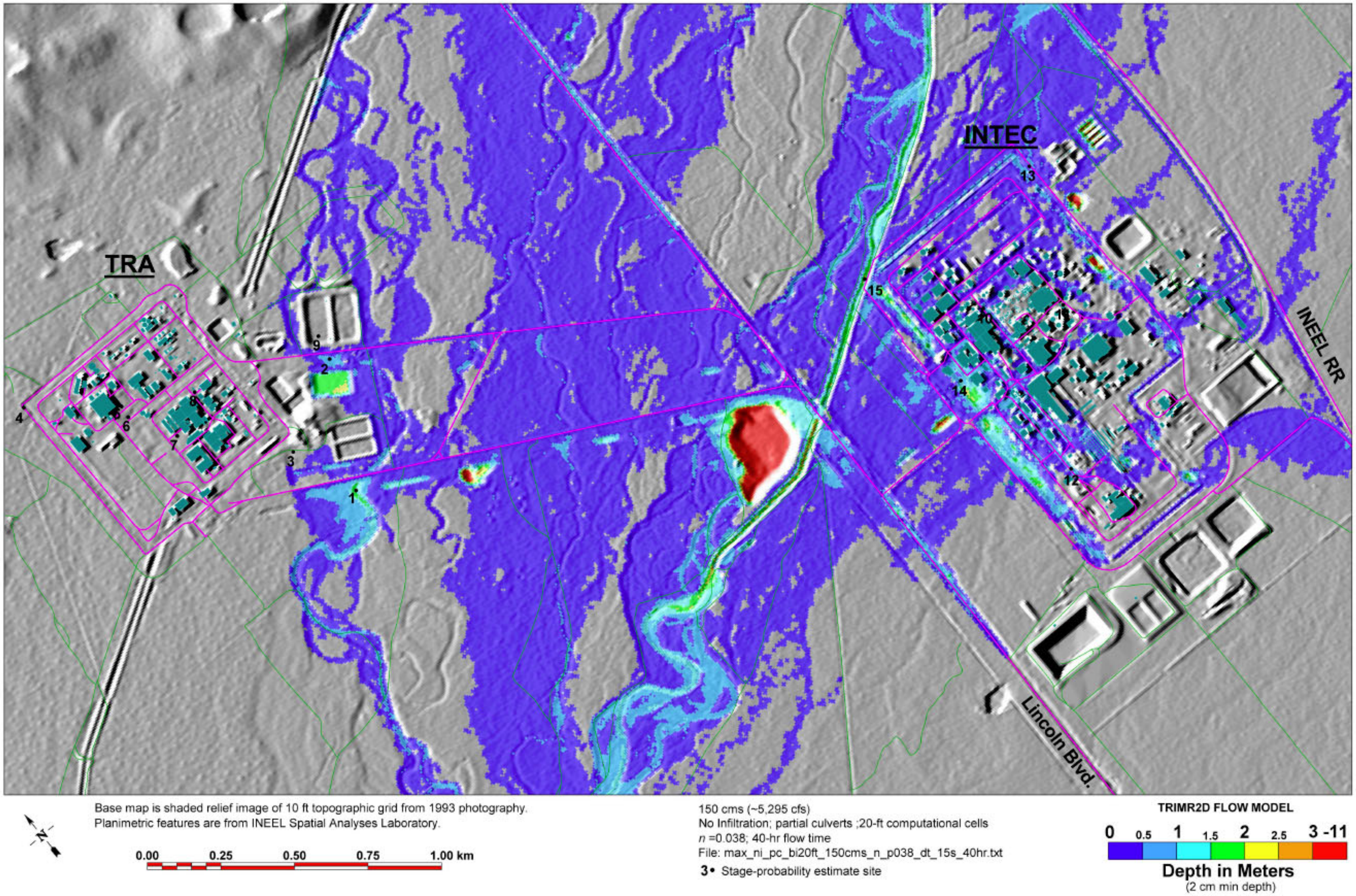

Figure 2-18. Modeled scenario for the Big Lost River, with no flow infiltration between the Idaho National Laboratory Site diversion dam and northeast of Idaho Nuclear Technology and Engineering Center and partial flow through Lincoln Boulevard culverts (from Ostenaa and O'Connell [2005], Appendix E).

The *Performance Assessment for the Tank Farm at the Idaho National Engineering and Environmental Laboratory* (DOE-ID 2003b) for closure of the TFF, located in the vicinity of CSSF, evaluated the impact of a Big Lost River flood on the TFF. The flood bounding scenario was an extreme precipitation event within the drainage basin and above the Mackay Dam, causing the overtopping failure of the dam. The evaluation concluded that impact from the extreme flooding conditions at INTEC is expected to be minimal. INTEC's elevation is near the highest elevation that floodwaters could potentially reach. Because INTEC would be near the edge of floodwaters, surface water flow velocities would have minor erosional effects. One to two meters (3.3 to 6.5 ft) of water could cover the facility, but this would occur only for a short duration. A small wetting front infiltrating into the unsaturated zone would occur. However, based on infiltration rates measured by Dunnivant et al. (1998) and the short duration that ponded water could occur at INTEC, the wetting front would only advance 4.9 to 10 m (16 to 33 ft) into the underlying alluvial soils (DOE-ID 2003b).

Perched water levels are monitored in the INTEC area, and the approximate lateral extent of the northern shallow perched water from the Big Lost River is also evaluated each year (Shanklin, Forbes, and Lawrence 2018). When the Big Lost River is flowing past INTEC, only one INTEC monitoring well—Well BLR-CH, which is the monitoring well located closest to the river and 152.4 m (500 ft) from the river channel—has consistently shown a significant water-level response to the river flow events. No other wells have shown any response to flow changes in the river (Shanklin, Forbes, and Lawrence 2018). These data support the assumption of neglecting any influence of recharge from the Big Lost River on unsaturated zone flow at the CSSF.

2.7.2 Precipitation Infiltration

Rain and snowmelt periodically infiltrate into the gravelly alluvium in and around INTEC and the CSSF. Though average annual precipitation (22.1 cm/year [8.7 in./year]) is much less than the pan evaporation rate (109 cm/year [42.9 in./year]) (Cahn et al. 2006), water from snowmelt or heavy rains can and does infiltrate into the ground to depths where it cannot evaporate. This water then continues to move downward until it recharges perched water and the SRPA.

The combination of coarse surficial sediments and lack of vegetation permits infiltration of a large fraction of the natural precipitation. Furthermore, many areas at INTEC are occupied by buildings or are paved with asphalt or concrete. Precipitation falling on building roofs is routed to downspouts. Water falling on paved surfaces tends to flow laterally to the pavement edge, where it may then infiltrate or flow into drainage ditches. The ditches are mostly unlined, and a significant fraction of infiltration is likely to occur along the ditches. Therefore, infiltration may be greater due to the impervious areas, which act to focus much of the surface run-off into gravelly areas or unlined drainage ditches.

The *Operable Unit 3-14 Tank Farm Soil and Groundwater Remedial Investigation/Baseline Risk Assessment* (Cahn et al. 2006) concluded that the recharge rate inside the INTEC security fence may be approximately 18 cm/year (7 in./year), which constitutes 85% of the average annual precipitation (22 cm/year [8.66 in./year]). Additional details regarding precipitation infiltration and recharge rates can be found in that report.

Ongoing remedial actions under the CERCLA OU 3-14 ROD (DOE-ID 2007a) likely have reduced infiltration and recharge rates within the northern portion of INTEC (DOE-ID 2018c). These remedial actions required recharge controls to reduce infiltration of precipitation and anthropogenic water (e.g., water leaks from underground pipelines) within the recharge control zone over the TFF area adjacent to the CSSF (see Figure 2-19). Required actions include capturing roof run-off from area buildings, eliminating landscape watering, eliminating steam condensate drip-leg discharges to the ground, lining drainage ditches with concrete or plastic, applying low-permeability asphalt pavement to reduce water infiltration, and directing water run-off toward lined ditches and a double-lined evaporation pond. The OU 3-14 Interim Remedial Action Report (DOE-ID 2018c) summarizes actions completed to date since the CERCLA OU 3-14 ROD was signed.

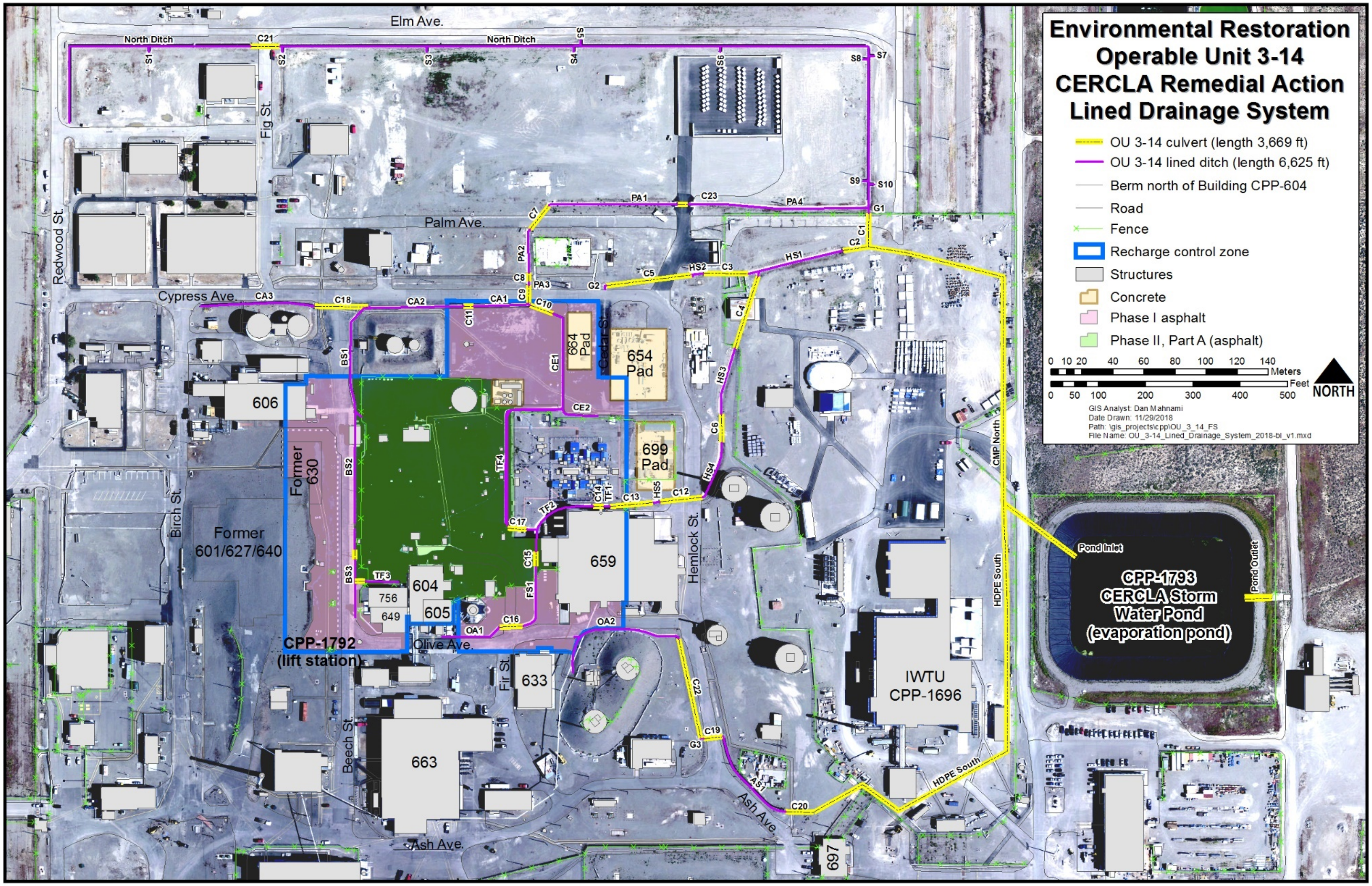

Figure 2-19. Operable Unit 3-14 recharge control drainage system at the Idaho Nuclear Technology and Engineering Center.

2.7.3 Anthropogenic Water Infiltration

Anthropogenic water includes intentional clean water discharges to the ground and accidental water leaks from underground water pipelines. It has been demonstrated that anthropogenic water infiltration at INTEC can be more important than precipitation infiltration with respect to perched water recharge (DOE-ID 2007b).

Over the past decade, numerous remedial actions have been performed to reduce anthropogenic recharge rates at and near INTEC. These actions include:

- Old percolation ponds were taken out of service (2002)
- Concrete-lined ditches were installed around tank farm (2003–2004)
- A lined evaporation pond was installed (2003)
- Sewage effluent was redirected to new percolation ponds (2004)
- Lawn irrigation was eliminated (2005)
- Subsurface injection of steam condensate was eliminated (2008)
- A plastic liner was installed in the North Ditch (2009)
- Asphalt and concrete were installed; sealcoating was applied to existing asphalt and concrete within the recharge control zone (2009–2017)
- Underground water line leaks were located/repaired (ongoing).

Of the actions listed above, elimination of the former percolation ponds in 2002 was the most significant in reducing anthropogenic recharge near the southern portion of INTEC (DOE-ID 2007b). Elimination of the infiltration trenches at the Sewage Treatment Plant in 2004 resulted in a significant reduction in water infiltration in the northern part of INTEC (DOE-ID 2007b).

2.7.4 Perched Water

Perched water zones have been observed at various depths within the 140-m-thick (460-ft-thick) unsaturated zone beneath INTEC as early as 1956 (Robertson, Schoen, and Barraclough 1974). Perched water has been observed at depths that coincide with the presence of low-permeability sedimentary interbeds within the thick sequence of basalt flows. Water moves downward through the alluvium, into fractures in the basalt, and continues vertically downward with minor lateral spreading until it encounters sedimentary interbeds, where vertical flow is impeded. Perched water has been observed in two distinct geographic areas: northern and southern INTEC. Based on the distribution and geochemistry of the perched water, it has been determined that the northern and southern shallow perched water systems were discontinuous, with separate recharge sources (Cahn et al. 2006).

Perched water was present in the southern portion of INTEC as early as 1963 (Robertson, Schoen, and Barraclough 1974), but larger volumes of perched water began to accumulate beneath INTEC starting in 1984, when the former percolation ponds began receiving service wastewater. Prior to 2002, which is the year the former percolation ponds were taken out of service, the principal zones of southern perched water observed were (1) intermittent perched water in the alluvium at the top of basalt at 9.1 m (30 ft) bls, (2) shallow perched water at the 34-m (110-ft) interbed, (3) intermediate perched water at a low-permeability basalt or interbed at 76 m (250 ft) bls, and (4) deep perched water at the 116-m (380-ft) sedimentary interbed.

Figure 2-20 shows hydrographs for shallow perched water monitoring wells located near CSSF, and Figure 2-21 shows the locations of these and other wells in the norther portion of INTEC. Perched water hydrographs (Figure 2-20) show a cyclical water-level oscillation, which is attributed to annual spring snowmelt and the resulting recharge of the perched water.

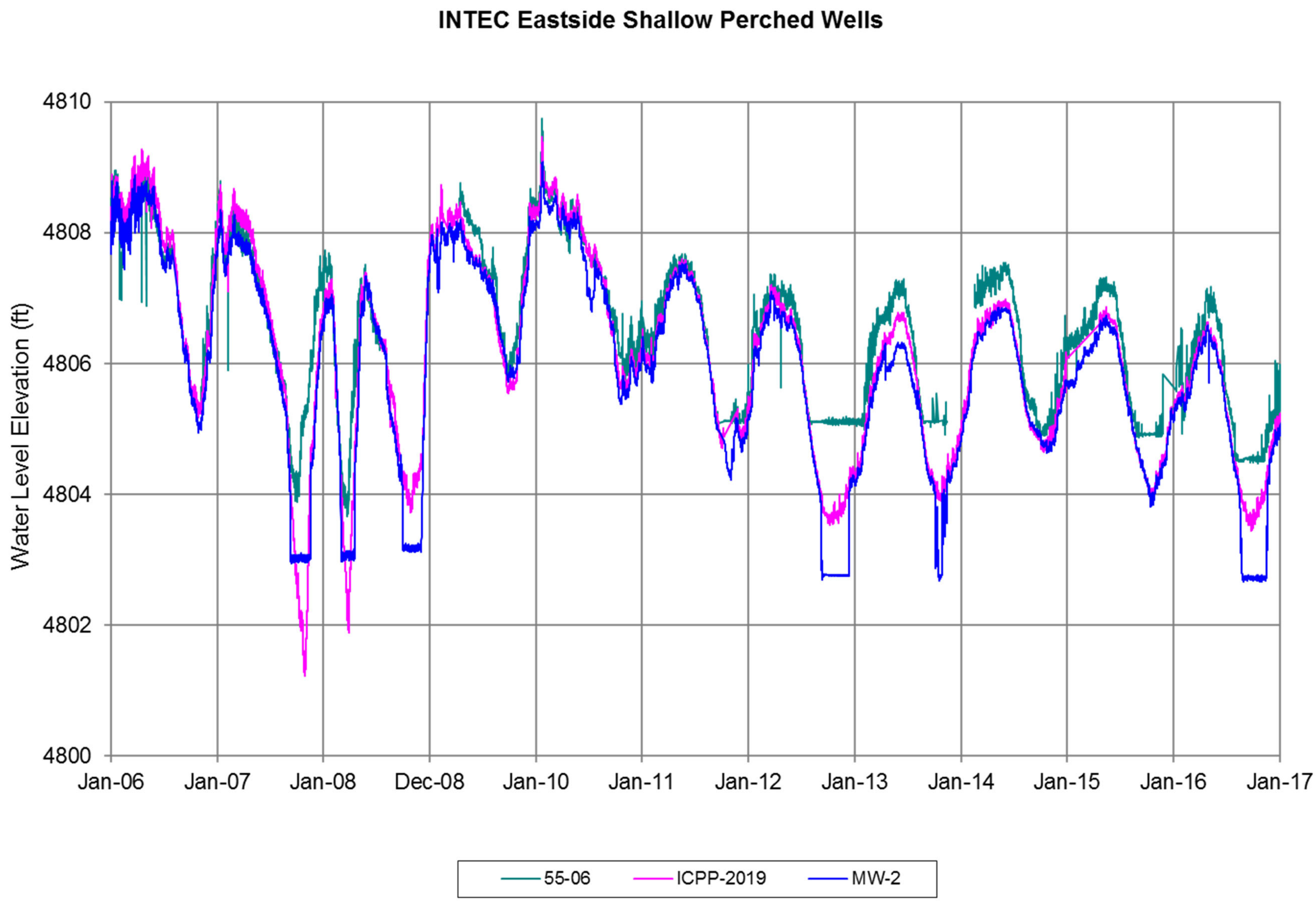

Figure 2-20. Hydrographs for selected perched water monitoring wells near the Calcined Solids Storage Facility (DOE-ID 2022a).

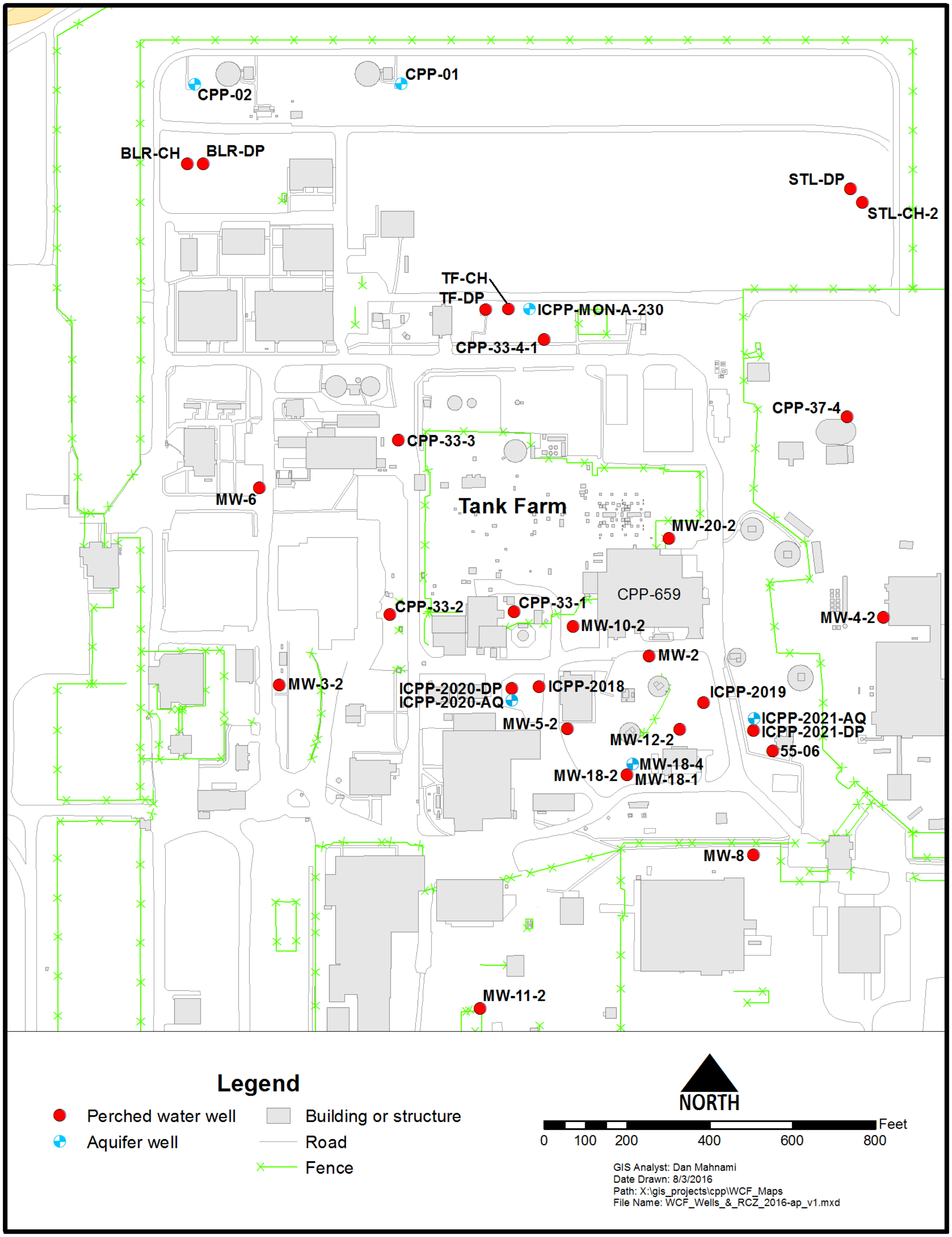

Figure 2-21. Monitoring well locations in the northern portion of the Idaho Nuclear Technology and Engineering Center.

The relocation of the INTEC percolation ponds in August 2002 to their current location 3.2 km (2 mi) west of INTEC caused a large reduction in anthropogenic recharge, resulting in rapid drain-out of perched water beneath the southern part of INTEC. By 2004, nearly all the perched water monitoring wells near the former percolation ponds had gone dry. More details regarding the accumulation and subsequent dissipation of perched water beneath the former percolation ponds can be found in Cahn and Ansley (2004) and Cahn et al. (2006).

2.7.5 Snake River Plain Aquifer

The SRPA, one of the largest and most productive groundwater resources in the United States, underlies the INL Site. The SRPA is listed as a Class I aquifer. It is the primary source of water for domestic, municipal, and industrial use in southeastern Idaho and provides large quantities of water for agricultural irrigation (along with surface water from the Snake River). The SRPA consists of a series of saturated basalt flows and interlayered pyroclastic and sedimentary materials that underlie the ESRP. The SRPA is approximately 322 km (200 mi) long, 64 to 97 km (40 to 60 mi) wide, and covers an area of 24,853 km^2 (9,600 mi^2). It extends from Bliss, Idaho, on the southwest margin of the ESRP to near Ashton, Idaho, northeast of the INL Site. Aquifer boundaries are formed by contacts with less permeable rocks at the margins of the ESRP (Mundorff, Crosthwaite, and Kilbum 1964).

Groundwater flow in the SRPA occurs predominantly through fractures (joints) in the basalt and along rubble zones at flow contacts (bedding planes) (DOE-ID 2022a). In the eastern SRPA, regional groundwater flow is to the southwest (DOE-ID 2022a). Recharge occurs primarily in mountain-front areas near the Yellowstone Plateau and Lost River Range. Lesser recharge sources include seepage into the bed of the Big Lost River (when flowing) and infiltration of irrigation water applied to agricultural lands near Howe and Mud Lake to the north and northeast of the INL Site, respectively (DOE-ID 2022a). The groundwater then flows southwest toward discharge areas at Thousand Springs near Hagerman, Idaho. On a local scale, groundwater flow directions may differ from regional flow paths because of fracture orientations (DOE-ID 2022a).

The USGS has maintained a groundwater-monitoring network at the INL Site to characterize the occurrence, movement, and quality of water and to delineate the movement of facility-related wastes in the SRPA since 1949. This network consists of a series of wells from which periodic water level and water quality data are obtained. In addition to the independent USGS groundwater monitoring, a groundwater monitoring program was implemented at INTEC in October 1991 to satisfy HWMA/RCRA and DOE O 231.1B Admin Chg 1, “Environment, Safety and Health Reporting,” groundwater monitoring requirements.

Hydraulic conductivities in the SRPA near INTEC commonly exceed 1,000 ft/day, with a maximum value of 8,800 ft/day at the former INTEC injection well (Anderson, Kuntz, and Davis 1999). Hydraulic conductivities beneath INTEC are among the highest anywhere at the INL Site. The very large hydraulic conductivities and fractured nature of the basalt aquifer matrix result in very rapid groundwater flow velocities, i.e., up to 5 ft/day near INTEC.

Depths to SRPA groundwater near INTEC range from 140.2 to 155.4 m (460 to 510 ft) bls. Figure 2-22 shows a groundwater elevation contour map for the area surrounding INTEC. The general direction of groundwater flow is south to southwest. The groundwater hydraulic gradient varies considerably across the map area but is relatively flat between INTEC and CFA, with less than 1.2 m (4 ft) of head difference over this 3.2-km (2-mi) distance because this is an area of very high hydraulic conductivity.

Groundwater levels declined during the 2000–2005 period because of drought during this time. Figure 2-23 shows groundwater hydrographs for several SRPA wells. The hydrographs show that groundwater levels declined more than 3 m (10 ft) in many SRPA wells across the southern INL Site during the most recent drought cycle (2000–2005). Groundwater levels have remained relatively constant from 2005 through 2015 because of near-normal precipitation but increased in response to the excessive snow pack in the winter of 2016–2017.

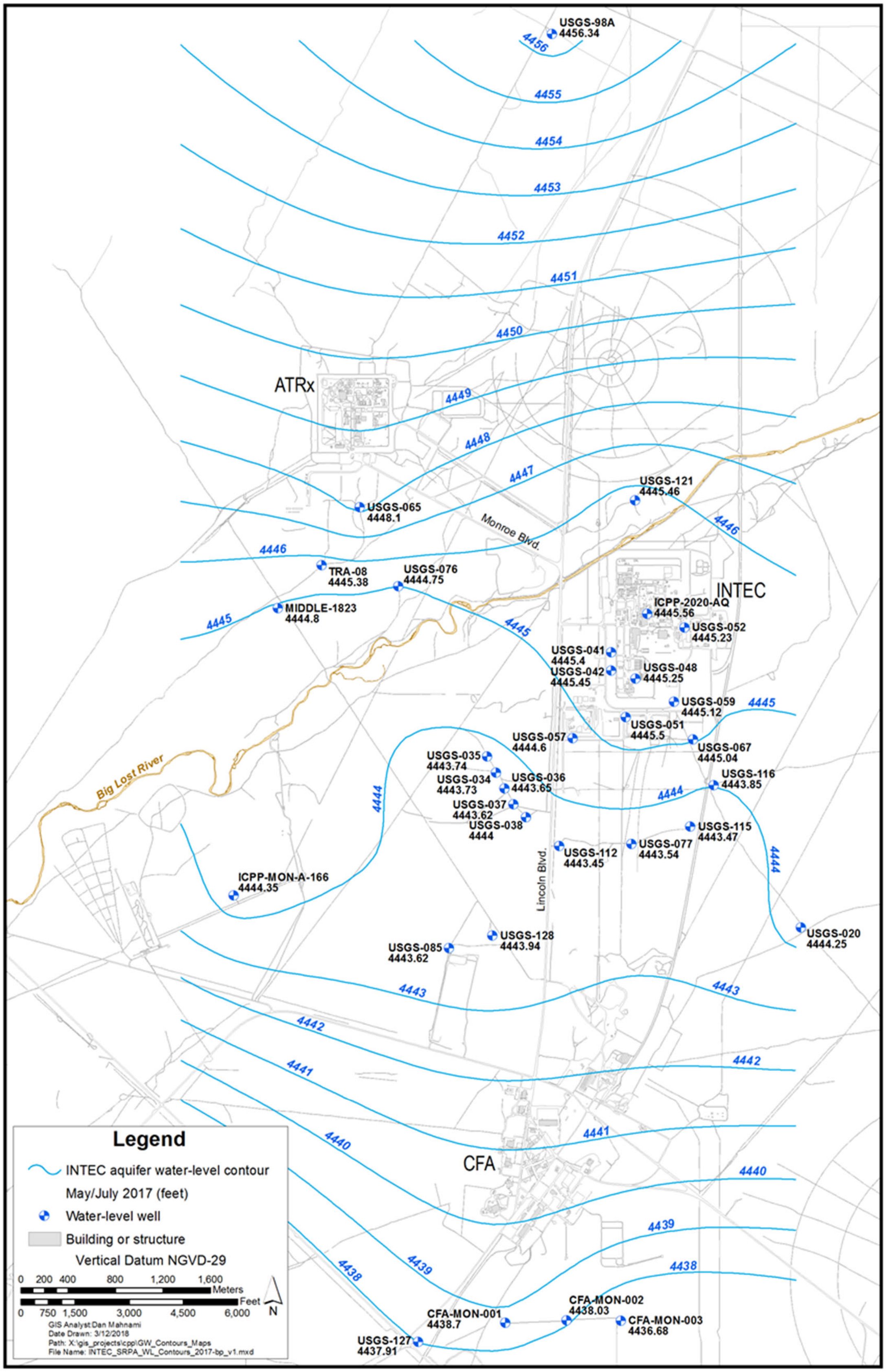

Figure 2-22. Snake River Plain Aquifer groundwater elevation contour map, May/July 2017.

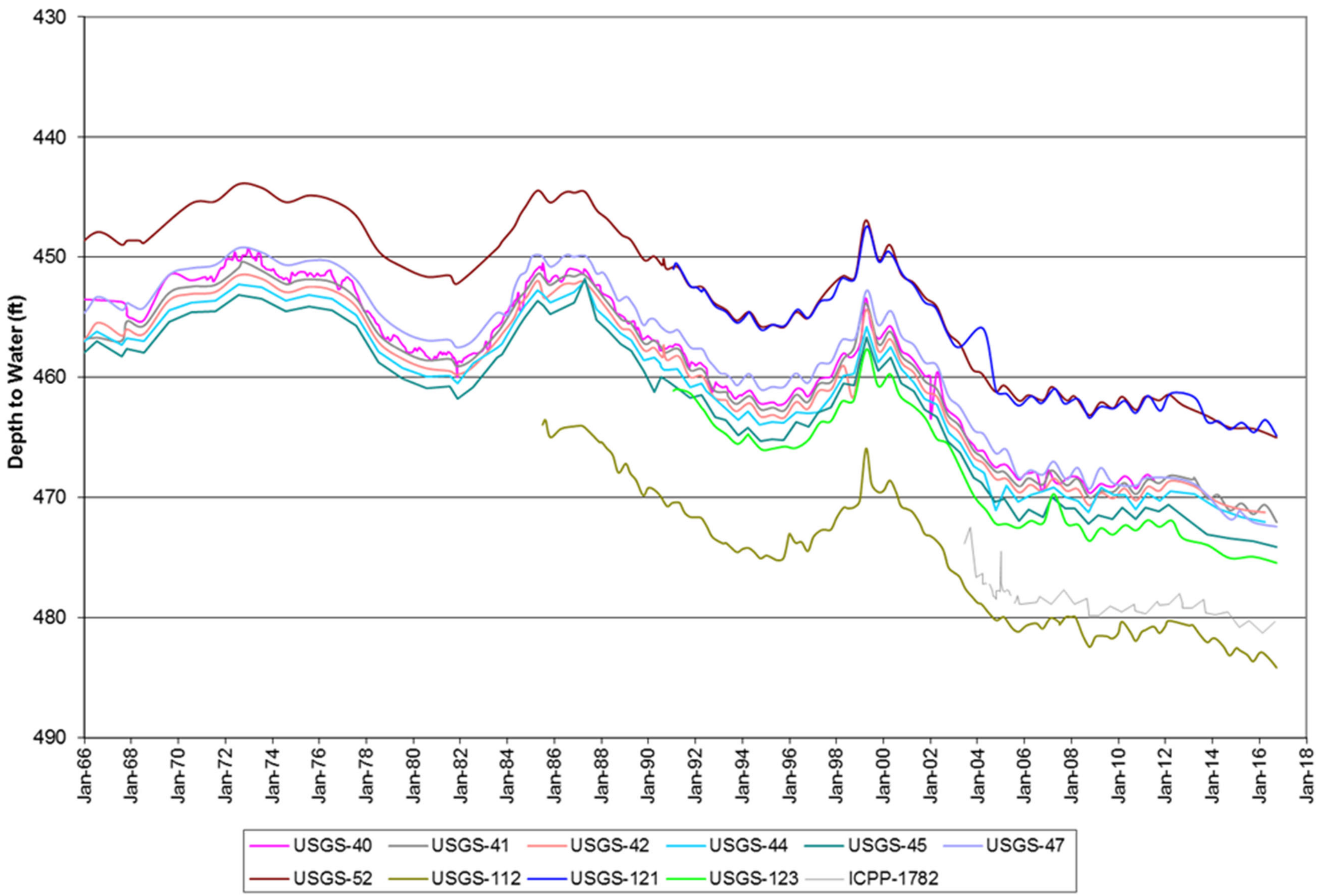

Figure 2-23. Hydrographs for selected Snake River Plain Aquifer wells (DOE-ID 2018c).

The water quality in the SRPA at and downgradient from INTEC has been adversely impacted due to past facility operations. The most significant water quality impacts resulted from the former INTEC injection well, CPP-03. The injection well was routinely used to discharge INTEC wastewater to the SRPA from 1952 to February 1984, when it was taken out of service as the percolation ponds became operational. During its operation, the injection well constituted a source of low-level radioactivity to the aquifer. The principal radionuclides of environmental significance discharged to the injection well were H-3, Sr-90, I-129, Cs-137, and Tc-99.

Since it was taken out of service in 1984, the former INTEC injection well no longer constitutes a continuing source of contaminants to the SRPA. However, drain-out of service waste from the deep unsaturated zone continues to contribute a slow flux of H-3, Sr-90, I-129, and other radionuclides to the SRPA. By 2003, H-3 and I-129 activities (derived from the injection well) had declined below their respective maximum contaminant levels (MCLs) in all SRPA monitoring wells downgradient of INTEC. As of 2017 reporting, Sr-90 and Tc-99 were the only radionuclides whose concentrations in groundwater still exceeded MCLs at and downgradient of INTEC (Shanklin, Forbes, and Lawrence 2018). However, concentrations of these contaminants are slowly declining because of radioactive decay and/or dilution and dispersion. Current trends indicate that concentrations of Sr-90 and Tc-99 from the former injection well will decline below the MCL before the Year 2095 (Cahn et al. 2006). However, other sources of Sr-90 in perched water pose a potential threat to the SRPA; these are being remediated under the OU 3-14 ROD (DOE-ID 2007a).

2.8 Geochemistry

Several studies have characterized the geochemistry of the perched water (EDF-5758) and the SRPA (McLing 1994; DOE-ID 2003a). These studies indicate that groundwater beneath the INL Site has multiple source regions and that these source regions have subtle chemical signatures (Johnson et al. 2000) that are defined by the geologic media in which the water originates. Waters originating from the Basin and Range valleys north of the INL Site that contain Paleozoic carbonates and siliciclastic sediments are therefore enriched in Ca-Mg-HCO_3. In contrast, waters originating in the Yellowstone Plateau volcanics contain more Na-K relative to those derived from the sedimentary terrain to the west. Two more-recent reports have expanded on this concept, defining two different groundwater quality types beneath the INL Site (western tributary water type and an eastern regional water type), as delineated by differences in lithium, boron, and fluoride concentrations in the SRPA (Bartholomay and Hall 2016; Fisher et al. 2012).

Based on isotopic evidence, groundwater flowing southward beneath INTEC and the CSSF is mostly derived from a fast flow corridor originating in the Little Lost River Valley (Roback et al. 2001). The fast flow zone is characterized by colder groundwater dominated by Ca-Mg-HCO_3. Bartholomay and Hall (2016) and Fisher et al. (2012) have also concluded that groundwater beneath INTEC is primarily derived from the western tributary valleys (Birch Creek and Little Lost River). SRPA groundwater is generally supersaturated with respect to calcite, which results in calcite precipitation in the sedimentary interbeds and as fracture and vesicle fillings within the basalt (McLing 1994).

Because most SRPA recharge comes in the form of snowmelt at high altitudes, groundwater entering the SRPA is quite cold ~9.0°C (48°F) (Smith and McLing 2001). Groundwater warms predictably as residence time increases in the SRPA. Near INTEC and the CSSF, SRPA groundwater temperatures generally fall between 11 and 13°C (51.8 and 55.4°F). In contrast, perched water temperatures at some INTEC wells are much warmer because of anthropogenic recharge from hot water leakage from steam and steam-condensate pipelines (Shanklin, Forbes, and Lawrence 2018).

Groundwater pH near INTEC and the CSSF ranges from approximately pH 7 to 8, with most values close to pH 7.9 (Shanklin, Forbes, and Lawrence 2018). The observed pH values are consistent with the presence of calcite ($CaCO_3$) in the SRPA matrix, which tends to buffer the pH in this range.

At most well locations, SRPA groundwater contains high concentrations of dissolved oxygen (>5 mg/L), which indicates that the groundwater is close to saturation with dissolved oxygen (Shanklin, Forbes, and Lawrence 2018). This is consistent with regional studies that confirm oxidizing conditions for the SRPA (McLing 1994).

Elevated concentrations of sodium and chloride persist in groundwater downgradient (south) of INTEC because of past service waste disposal at the former injection well and, later, at the former percolation ponds. Sodium and chloride levels at SRPA wells near the former percolation ponds remain several times above background concentrations more than a decade after the ponds were taken out of service. The observed slow decline in the salinity of groundwater beneath the former percolation ponds is evidence of continuing drain-out of high-salinity perched water and/or the diffusion of sodium and chloride out of low-permeability zones in the SRPA.

Total dissolved solids concentrations in SRPA groundwater typically fall within the range of 200 to 400 mg/L (Shanklin, Forbes, and Lawrence 2018), with the higher values observed at “skimmer wells” that are screened across only the upper few feet of the aquifer. As of 2017, the highest total dissolved solids value (391 mg/L) was observed at Well ICPP-MON-A-230 located north of the TFF. The elevated groundwater total dissolved solids at this location are attributable to past high-salinity waste releases in and around the tank farm area.

Figure 2-24 shows a Piper trilinear diagram that illustrates SRPA groundwater quality variations. The diagram shows that most SRPA groundwater is of the Ca-Mg-HCO_3 type, which falls toward the left side of the diamond-shaped Piper plot. This is typical of the SRPA, which contains an abundance of $CaCO_3$. Upgradient monitoring well USGS-121 represents the most extreme Ca-Mg-HCO_3 water type. In contrast, Well USGS-51 contains water of the $CaCl_2$ type because of residual chloride from the former percolation ponds.

2.9 Natural Resources

2.9.1 Geologic Resources

Geologic resources at the INL Site are very limited (DOE-ID 2011). INL Site mineral resources include sand, gravel, pumice, silt, clay, and aggregate (DOE-ID 2011). These resources are extracted at several quarries or pits at the INL Site and used for road construction and maintenance, waste burial activities, and ornamental landscaping (DOE-ID 2011). The geologic history of the ESRP makes the potential for petroleum products at the INL Site very low (DOE-ID 2011).

2.9.2 Water Resources

Groundwater from the SRPA aquifer supplies most of the water for the area surrounding the INL Site and essentially all drinking water consumed within the ESRP (IDEQ 2021). Water from the SRPA is used for agriculture, food processing, aquaculture, and domestic, rural, public, and livestock water supplies. Each year, approximately 1.1 million acre-ft of water is drawn from the SRPA. Approximately 95% is used for irrigation, 3% for domestic water, and 2% for industrial purposes (IDEQ 2006, 2021). It is estimated that more than 300,000 people depend on the SRPA for domestic and municipal water needs (IDEQ 2005).

The SRPA is the only source of water used at the INL Site, and the Federal Reserved Water Right for the INL Site is 35,000 acre-ft/year (1.14E+10 gal/year), not to exceed a maximum diversion rate of 80 ft^3/second (35,906 gpm). DOE works cooperatively with the Idaho Department of Water Resources under the 1990 Water Rights Agreement (Monson 1990) to provide information on water use at the INL Site. Annual reports summarize water usage, including the total volume of water diverted at the INL Site and for each facility, maximum diversion rate, and available pumping levels (water depth). Approximately 2.92E+09 L (7.72E+08 gal) of total annual water use was recorded at the INL Site in 2017, or approximately 6.8% of the water right (INL 2018). Generally, less than 10% of the water right is used each year.

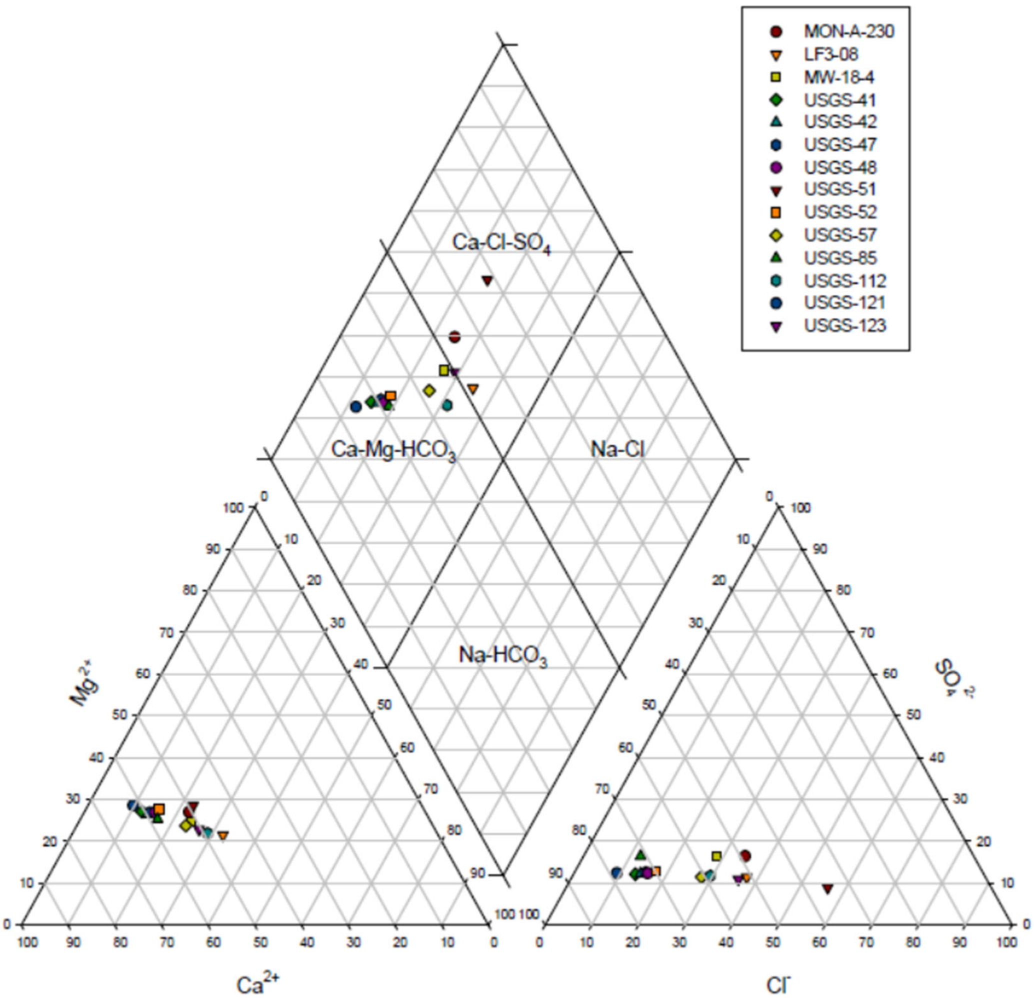

Figure 2-24. Piper trilinear diagram for selected Snake River Plain Aquifer wells near the Idaho Nuclear Technology and Engineering Center (DOE-ID 2022a).

2.10 Natural Background Radiation

Measurement programs are conducted to characterize radiological conditions in the environment at the INL Site and in the surrounding region (DOE-ID 2018a). Results of these programs show that exposures of workers and members of the public resulting from INL Site's airborne radionuclide emissions are well within applicable standards and are a small fraction of the dose from background sources (DOE-ID 2018a).

DOE has compared radiation levels monitored on and near the INL Site with those monitored at distant locations. Radiation levels at locations on the INL Site and at boundary community locations include contributions from background conditions and INL Site emissions. Data show that average radiation exposure levels for boundary locations were no different than those at distant stations. For 2017, the estimated potential population dose measured by the Environmental Surveillance, Education, and Research Program was 1.06E-02 person-rem (1.06E-04 person-Sv) (DOE-ID 2018a). This dose is approximately 0.000003% of that expected from exposure to natural background radiation of 127,411 person-rem (1,274 person-Sv) (DOE-ID 2022a).

The dose associated with radiological emissions is assessed annually to demonstrate compliance with the "National Emissions Standards for Hazardous Air Pollutants" (40 CFR 61) and "National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities" (40 CFR 61, Subpart H). The annual ED to the maximally exposed individual (MEI) resulting from radionuclide emissions from INL Site facilities during 2015 and 2016 was estimated at 0.0333 and 0.0143 mrem, respectively (DOE-ID 2016, 2017). These doses, as well as previous annual dose estimates, are well below both the EPA annual dose limit (10 mrem) and the annual dose received from background sources (approximately 620 mrem) (DOE-ID 2022a).

2.11 Idaho Nuclear Technology and Engineering Center Operational Background

INTEC, originally called the Idaho Chemical Processing Plant, began storing SNF in 1952 (DOE-ID 2022a). SNF was brought to INTEC from a variety of reactors throughout the world and stored in underwater or dry storage facilities for an interim period (DOE-ID 2022a). Beginning in 1953, some of the SNF was "reprocessed," a chemical treatment process that recovered enriched uranium and other products from the SNF (DOE-ID 2022a). SNF reprocessing and other INTEC support activities produced liquid radioactive waste that was stored in the TFF. Liquid waste was then sent to the waste calcining facilities to be converted into a solid, granular form called calcine (DOE-ID 2022a). Calcination consisted of spraying liquid wastes into a fluidized bed of thermally hot solids where the aqueous portion of the waste evaporated, leaving behind the dissolved constituents as the granular calcine material (Staiger and Swenson 2021). The calcination process produced a safer product for storage while reducing the volume of stored waste by an average factor of 7 (Staiger and Swenson 2021). The resulting granular solid (metallic oxides and fluorides) was pneumatically transported through transport air lines to the CSSF for storage (Staiger and Swenson 2021). Figure 2-25 depicts the former SNF reprocessing operations.

Calcine production began in 1963 with operation of the WCF (Staiger and Swenson 2021). WCF calcine production filled CSSFs 1 through 3 to or near capacity (Staiger and Swenson 2021). The WCF was shut down in March 1981, and in 1999, it was closed as a landfill with a HWMA/RCRA-compliant cap (DOE-ID 1997). Calcine production switched to the NWCF in August 1982 (Staiger and Swenson 2021). The NWCF operated until 2000 (Staiger and Swenson 2021), sequentially transferring calcine to CSSFs 4 through 6. CSSF 6 was only partially filled (713 m^3 [25,179 ft^3] of 1,506 m^3 [53,184 ft^3] usable capacity) before operations ended (Staiger and Swenson 2021). CSSF 7 was built to store future calcine but was never placed in service (Staiger and Swenson 2021). Operations ended after DOE decided to close the NWCF and treat remaining liquid waste with a different process (DOE-ID 1999b; DOE 2005). The portion of the NWCF used for calcination was closed under a partial closure plan (DOE-ID 2002); other portions of the NWCF are operated under HWMA/RCRA partial permits for waste treatment and storage (PER-109; PER-111). Closure is pending these other HWMA/RCRA-permitted operations ending at the NWCF (DOE-ID 2002). Figure 2-26 shows the location of the calcining facilities and CSSF.

The following subsections describe the design of CSSFs 1 through 6. Detailed descriptions and drawings are available in the CSSF HWMA/RCRA partial permit (PER-114), the CSSF Safety Analysis Report (SAR-105), and Stagier and Swenson (2021).

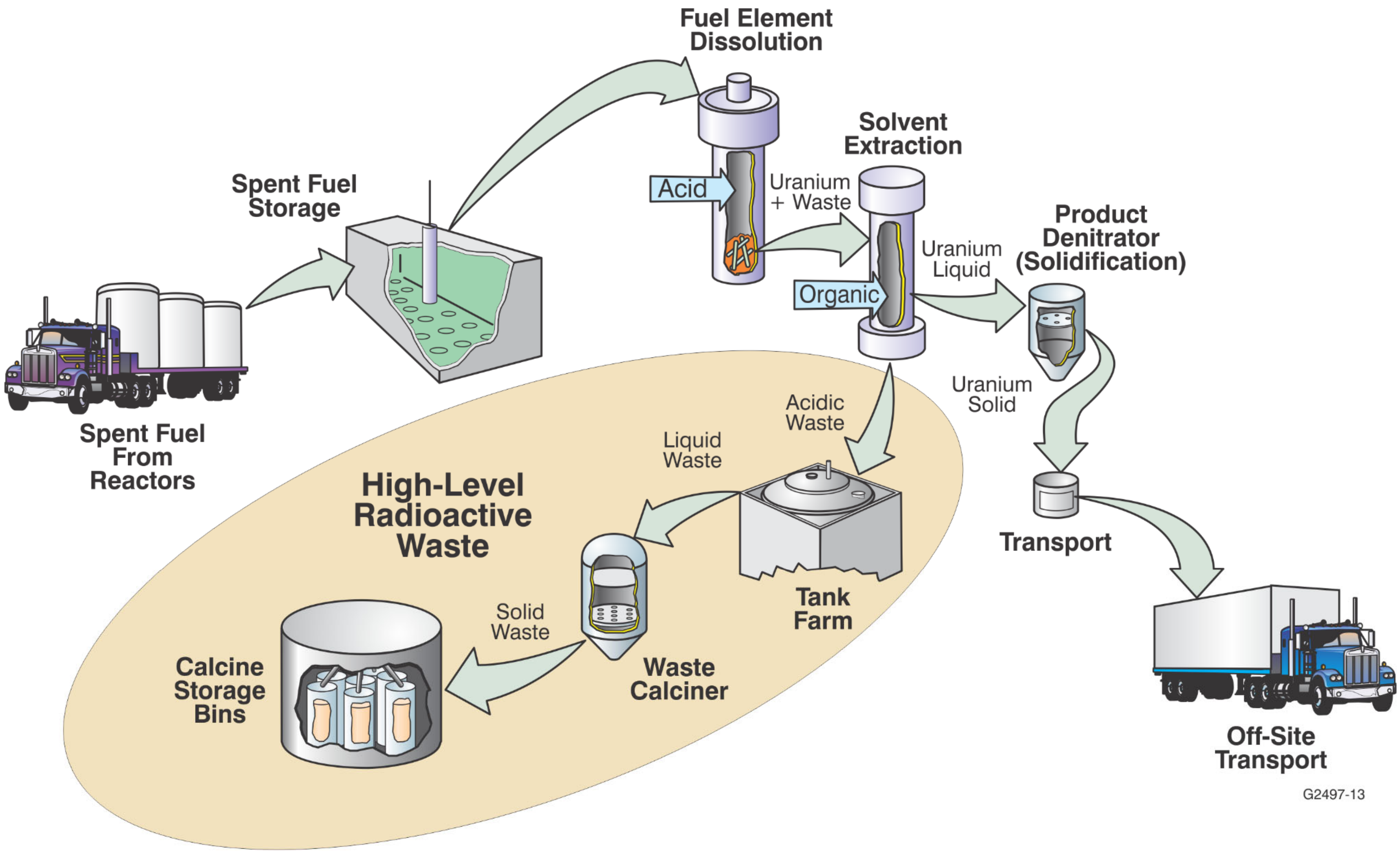

Figure 2-25. Former reprocessing of spent nuclear fuel at the Idaho Nuclear Technology and Engineering Center.

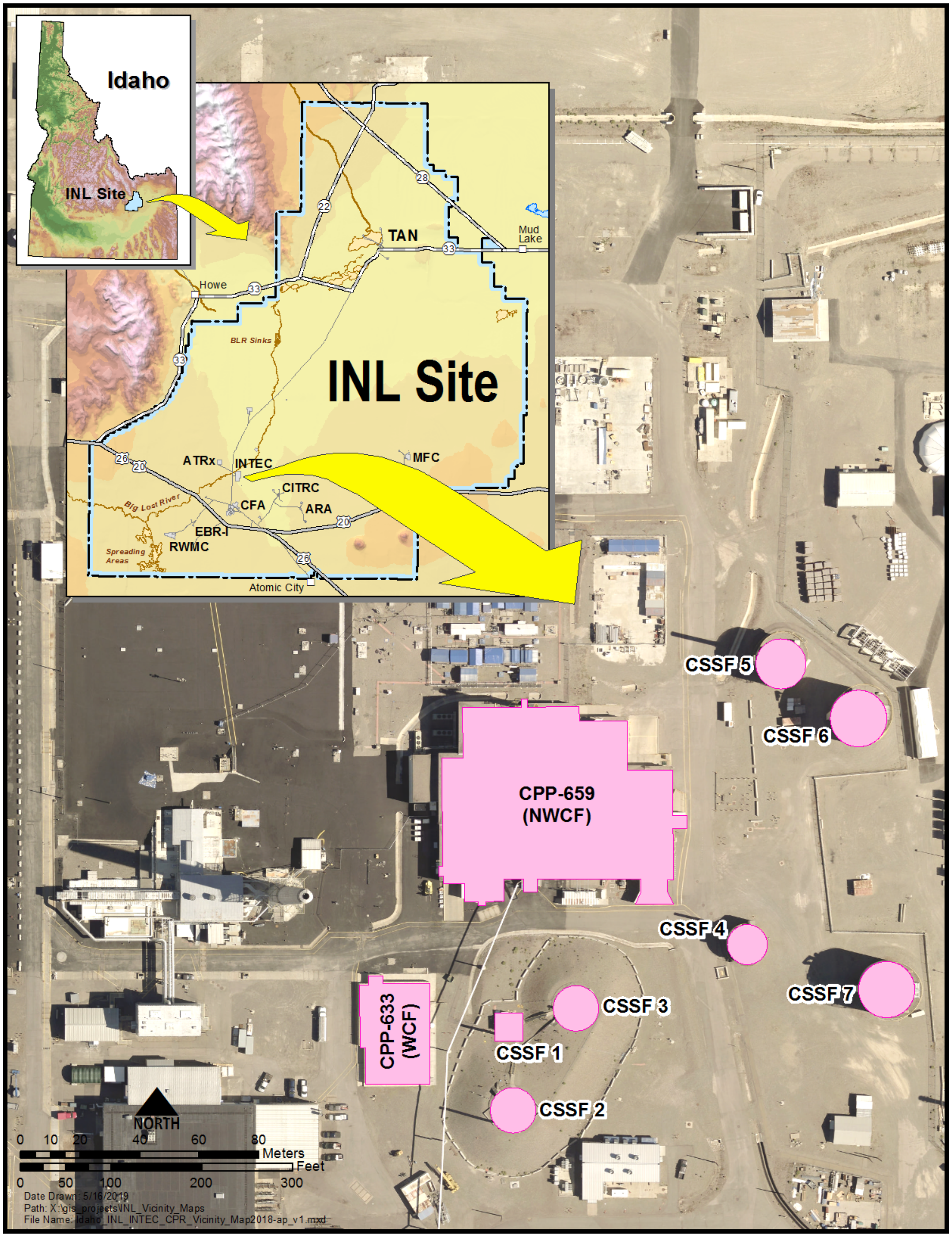

Figure 2-26. Calcined Solids Storage Facility location at the Idaho Nuclear Engineering and Technology Center.

2.11.1 CSSF Description

This section describes the configuration and characteristics of each CSSF as provided in Staiger and Swenson (2021), excluding CSSF 7. CSSF 7 never received or stored calcine, and therefore calcine removal from CSSF 7 will not be required.[21] This Draft CSSF 3116 Basis Document, however, only covers the CSSF bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine. In the CSSF PA/CA (DOE-ID 2022a), residual waste in the bins (including integral equipment) was assumed to be located on the bottom of the bins. This produced pessimistic[22] results in the analysis and is therefore bounding. The integral equipment in each CSSF will be evaluated prior to closure and will be either left in place (and covered by this Draft CSSF 3116 Basis Document) or removed and disposed of,[23] as identified in future State-approved closure plans.

CSSFs 1 through 6 consist of several stainless-steel bins housed in a reinforced-concrete vault (see Figure 2-27). Table 2-3 summarizes CSSF operational information and CSSF-specific configuration details relevant to this document, such as the number of bins, bin height, calcine volume, and usable capacity. The CSSFs have been referred to using either a Roman or Arabic numerical designator (e.g., CSSFs I, II, III or CSSFs 1, 2, or 3). This document uses Arabic numerical designators.

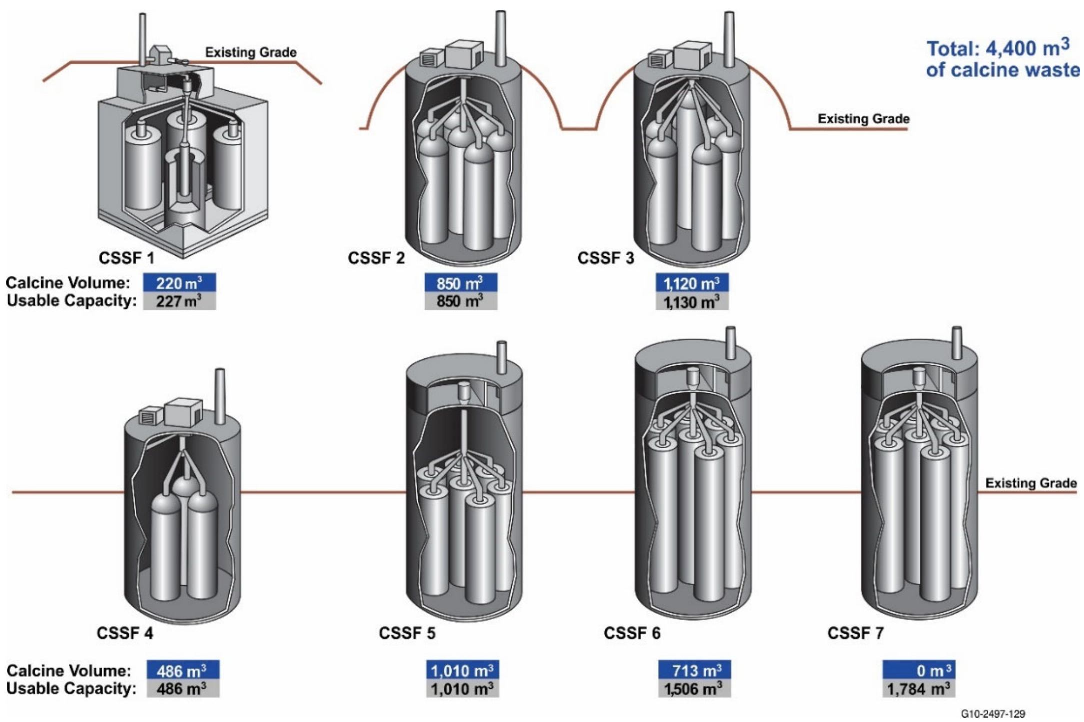

Figure 2-27. Illustration of the Calcined Solids Storage Facilities and their unique volumes and usable capacities (DOE-ID 2022a).

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21. HAD-353, "Hazard Assessment Document for the Calcined Solids Storage Facility 7" states that CSSF 7 contains no radioactive material. There are two valves on the transport lines to CSSF 7 (TAV-WS7-4 and TAV-WS7-5), which are the boundary for the transport lines. These values are closed and have a permanent device installed to prevent them from being opened.

22. Throughout this document, and in the CSSF PA/CA, the term *pessimistic* is used to reflect the intent to overpredict consequences.

23. Any waste generated during retrieval and closure of each CSSF will be appropriately characterized, classified (per DOE M 435.1-1 Chg 3), and verified to comply with the receiving facility's waste acceptance criteria. Such waste and its disposal are not covered by this Draft CSSF 3116 Basis Document.

Table 2-3. Calcined Solids Storage Facility operational information and construction details (DOE-ID 2022a).

CSSF	Construction Completion Date	Construction Material	Minimum Thickness of Bin Construction Material (mm [in.])	Thickness of Storage Vault Floor Slab (m [ft])	Year Filled	Number of Bins	Bin Design	Bin Height (m [ft])	Bin Diameter (m [ft])	Calcine Volume (m^3)	Usable Capacity (m^3)	Percent Full (%)
1	1959	405 stainless-steel plate	3.18 (0.125)	0.6 (2)	1964	4[a]	Concentric[a]	b	3.7 (12)	220	227	97
2	1965	304 stainless-steel plate	6.4 (0.25)	1.5 (5)	1972	7	Cylindrical	12.9 (42.3)	3.7 (12)	850	850	100
3	1969	304 stainless-steel plate	6.4 (0.25)	1.5 (5)	1981	7	Cylindrical	c	3.7 (12)	1,120	1,130	99
4	1976	304 stainless-steel plate	9.53 (0.375)	1.4 (4.5)	1983	3	Cylindrical	16.8 (55)	3.7 (12)	486	486	100
5	1978	304L stainless-steel plate	9.53 (0.375)	1.5 (5)	1992	7	Annular	16.8 (55)	3.7 (12)[d]	1,010	1,010	100
6	1980	304L stainless-steel plate	9.53 (0.375)	2 (6.5)	2000	7	Annular	20.8 (67.5)	4.1 (13.5)[e]	713	1,506	47

a. Each CSSF 1 bin consists of three concentric sub-bins—an inner cylindrical sub-bin and two outer annular sub-bins (see Figure 2-29). The concentric sub-bins are referred to as Bins A, B, and C, from inside to outside, respectively.

b. Three inner sub-bins (Bin A) are 7.6 m (25 ft) tall, and the fourth inner sub-bin is approximately 8.5 m (28 ft) tall. The outer annular sub-bins (Bins B and C) are approximately 6.1 m (20 ft) tall.

c. Six of the bins are approximately 16.2 m (53 ft) tall, and the seventh (center) bin is 18.6 m (61 ft) tall.

d. CSSF 5 bins are annular with a 1.2-m-diameter (4-ft-diameter) inner cylinder running through the length of the bin to provide more bin surface area for heat dissipation. The bin outer diameter is 3.7 m (12 ft).

e. CSSF 6 bins are annular with a 1.5-m-diameter (5-ft-diameter) inner cylinder running through the length of the bin to provide more bin surface area for heat dissipation. The bin outer diameter is 4.1 m (13.5 ft).

CSSF Calcined Solids Storage Facility

2.11.1.1 CSSF 1

CSSF 1 structures consist of the storage vault structure (CPP-741), cyclone vault (CPP-729), vault cooling air blower and instrument building (CPP-639), and solids cooling air stack (CPP-732). CSSF 1 consists of four composite bin groups (VES-WCS-115-1 through VES-WCS-115-4),[24] a distributor pipe (VES-WCS-3083), a cyclone (CYC-WCS-915), and transport lines to where they were cut and capped during HWMA/RCRA closure of the WCF. The CSSF transport lines are discussed in Subsection 2.11.1.7. The bins and distributor pipe are located wholly underground in the storage vault structure (CPP-741). Figure 2-28 shows a cross section of CSSF 1. This Draft CSSF 3116 Basis Document covers the CSSF 1 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe and cyclone.

The CSSF 1 bins are constructed of Type 405 stainless steel (Staiger and Swenson 2021). Each composite bin (VES-WCS-115-1 through VES-WCS-115-4) consists of three concentric sub-bins, numbered from inside to outside Bin A, B, and C (see Figure 2-29). The composite bin or nested bin configuration is also referred to as a “bin group.” The innermost sub-bin (Bin A) in each group is cylindrical. Each cylindrical sub-bin is surrounded by an annular sub-bin (Bin B), which is, in turn, surrounded by a second annular sub-bin (Bin C). Small gaps between the sub-bins provide a path for airflow, which removes decay heat from the radioactive calcine (Staiger and Swenson 2021).

Each cylindrical sub-bin (Bin A) has a diameter of approximately 0.9 m (3 ft) and is 7.6 m (25 ft) tall (VES-WCS-115-1 through VES-WCS-115-3) (Staiger and Swenson 2021). The cylindrical sub-bin (Bin A) of VES-WCS-115-4 is approximately 8.5 m (28 ft) tall (Staiger and Swenson 2021). The bin wall thickness varies from 3.18 mm (0.125 in.) at the bottom to 6.4 mm (0.25 in.) at the top. The bottom of the bin is 7.938 mm (0.3125 in.) thick, and the top of the bin is 6.4 mm (0.25 in.) thick. Each of the two outer annular sub-bins (Bins B and C) surrounding the central cylindrical bin is approximately 6.1 m (20 ft) tall and has a flat top and bottom. The inner annular bin (Bin B) has a 104-cm (41-in.) inside diameter and a 229-cm (90-in.) outside diameter. The inner annular bin (Bin B) has an inner wall thickness of 3.18 mm (0.125 in.), an outer wall thickness of 4.763 mm (0.1875 in.), a bottom thickness of 8.26 mm (0.325 in.), and a top thickness of 6.4 mm (0.25 in.). The outer annular bin (Bin C) has a 239-cm (94-in.) inside diameter and a 361-cm (142-in.) outside diameter. The outer annular bin (Bin C) has a wall thickness of 4.763 mm (0.1875 in.), a bottom thickness of 7.938 mm (0.3125 in.), and a top thickness of 4.763 mm (0.1875 in.) (Staiger and Swenson 2021). The usable calcine storage volume of CSSF 1 is approximately 227 m^3 (8,016.43 ft^3), of which 220 m^3 (7,769.23 ft^3) (97%) is used (Staiger and Swenson 2021). Figures 2-30 and 2-31 are historical photographs of the construction of CSSF 1 and placement of concentric bin VES-WCS-115-2 in the storage vault.

CSSF 1 bins contain internal obstructions such as thermowells, internally mounted wall stiffeners, bottom braces, and bin fill lines. Each of the cylindrical sub-bins (Bin A) has a centerline-mounted thermowell that extends from the top of the bin nearly to the bottom of the bin. The thermowell contains a series of thermocouples that were used to determine the calcine level while the bin was being filled and to monitor the calcine temperature thereafter. Each of the annular sub-bins (Bins B and C) contains two main thermowells and at least one secondary thermowell. The two main thermowells are located near the center of the annulus (midway between the inside and outside walls), on opposite sides of the bin, and extend from the top of the bin nearly to the bottom of the bin. Each of the bins contains internal stiffening rings on the outer bin wall. The annular bins also contain internal stiffening rings on the inner bin wall as well as stiffeners on the flat bin floor. The bin floor stiffening ribs are 3 in. tall. The stiffening rings on the outer and inner walls of the sub bins are flat bars that extend 2 to 3 in. from the walls. The outer wall stiffening rings in Bin C have a 2 in. face that extends 1 in. up and down from the ring. All stiffening rings are fabricated from Type 405 stainless steel.

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24. Refer to the nomenclature for a description of the acronyms used for the CSSF equipment numbering.

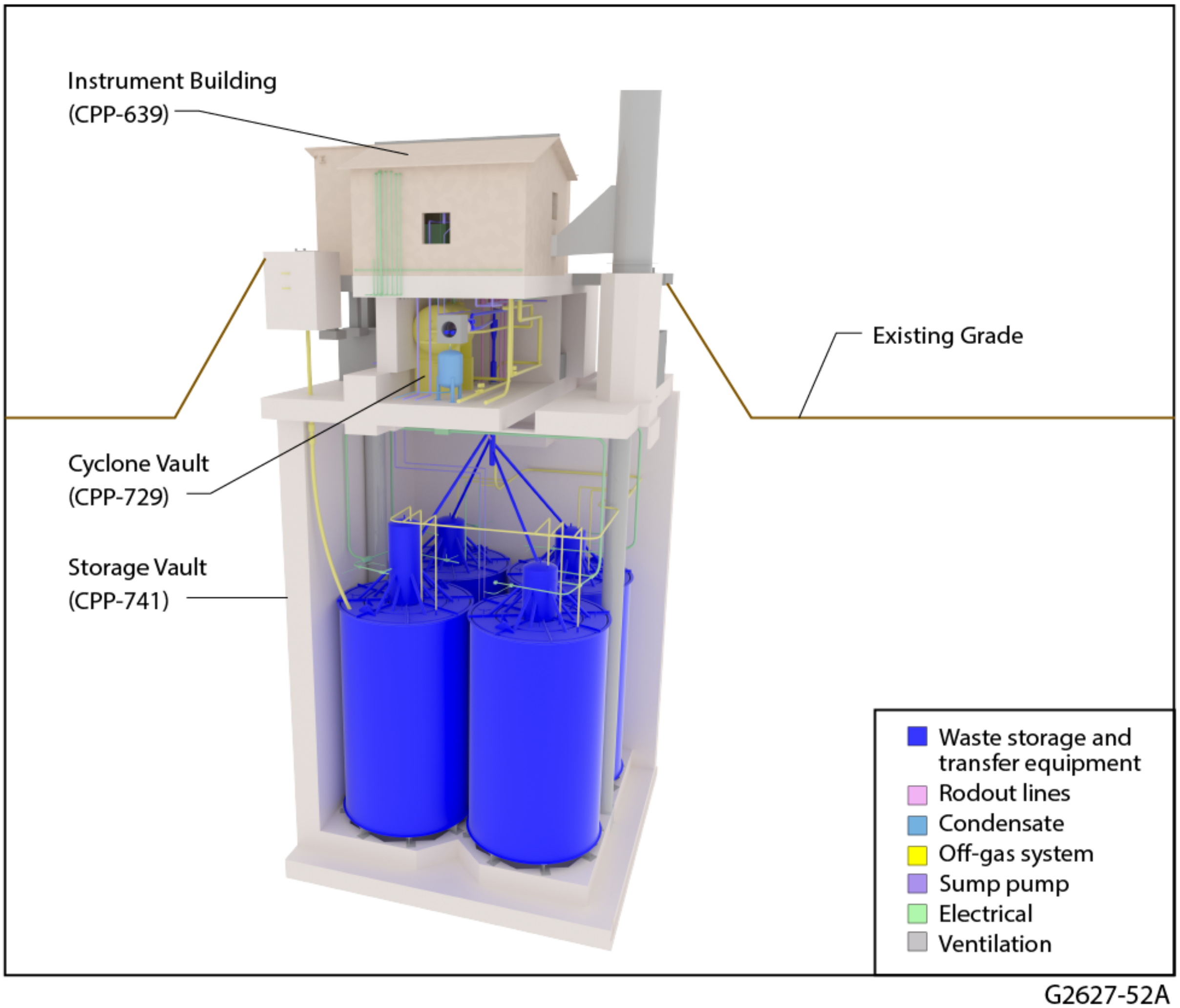

Figure 2-28. Calcined Solids Storage Facility 1 cross section (DOE-ID 2022a).

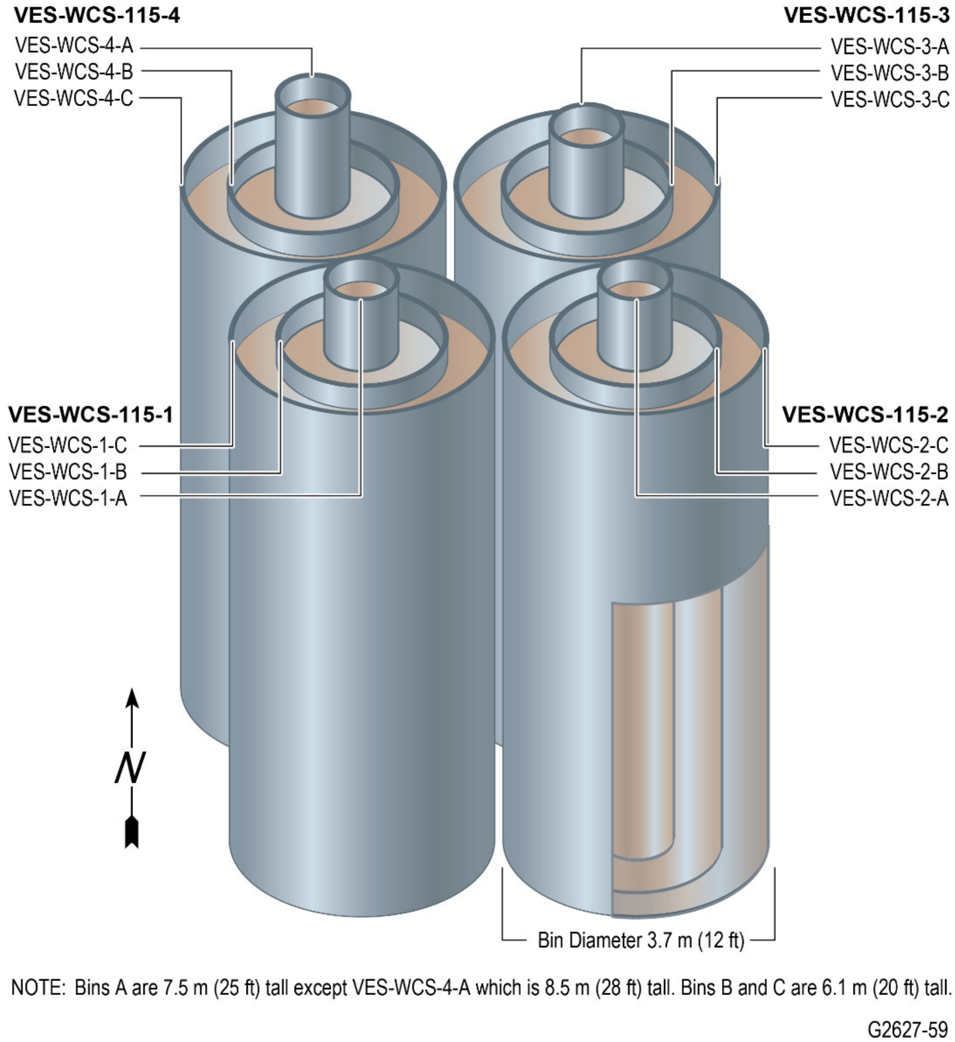

Figure 2-29. Illustration of the Calcined Solids Storage Facility 1 nested bin configuration (DOE-ID 2022a).

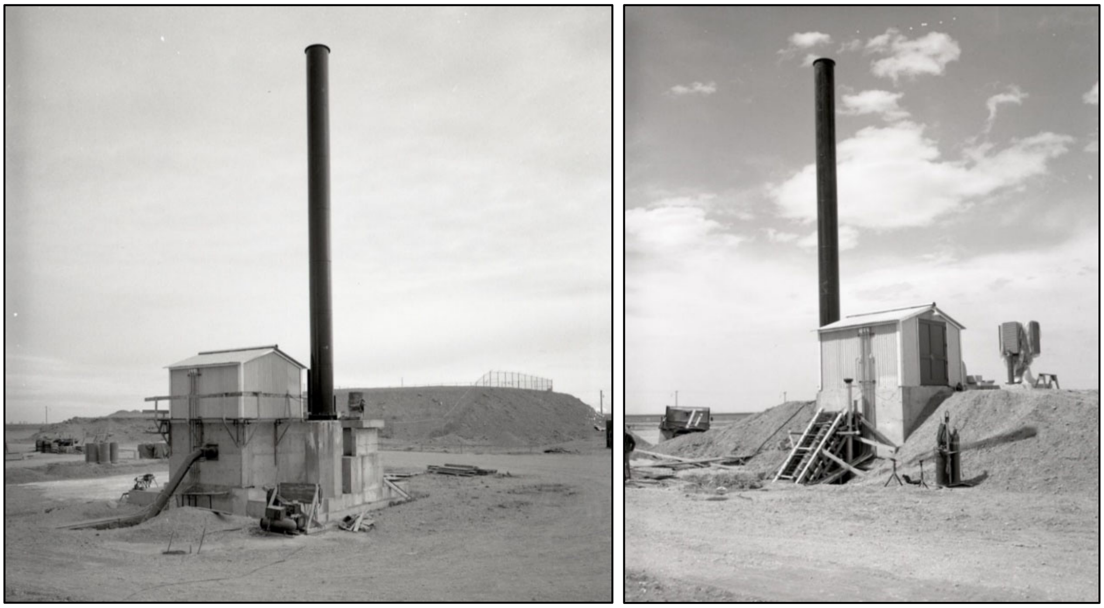

Figure 2-30. Historical photographs taken during construction of Calcined Solids Storage Facility 1 (EDMS ID 7188225).

Figure 2-31. Historical photograph of VES-WCS-115-2 placed in the Calcined Solids Storage Facility 1 storage vault (EDMS ID 7188225).

Each of the center sub-bins (Bin A) was filled sequentially from a calcine distribution piping system connected to the solids outlet of the CSSF 1 cyclone. Each center sub-bin acted as the solids distributor for its two annular sub-bins. The center bin is equipped with two sets of distribution piping, each at a different elevation, connected to the two outer annular bins. When calcine in the center bin reached the elevation of the lower of the two sets of distribution piping, it overflowed from the center bin, through the distribution piping, and into the first annular sub-bin (Bin B). When the Bin B (and its fill lines) filled with calcine, calcine again began filling Bin A until it reached the second, upper, set of distribution piping. At that point, the calcine overflowed into the outer, annular sub-bin (Bin C). When Bin C (and its fill lines) filled, calcine once again collected in Bin A until it was filled. The network of calcine distribution piping extends from Bin A, over the tops of Bins B and C. This calcine distribution piping is visible in Figure 2-31.

The CSSF 1 storage vault is a rectangular reinforced-concrete structure. The reinforced-concrete lower level of the vault is founded on bedrock approximately 16.6 m (54.3 ft) bls. The main chamber of the CSSF 1 storage vault has inside horizontal dimensions of 7.8 m (25.5 ft) × 7.8 m (25.5 ft) and a height of 12.6 m (41.3 ft). The walls are 0.76 m (2.5 ft) thick, the floor slab is 0.61 m (2 ft) thick, and the ceiling (partition between upper and lower vault sections) is 0.53 m (1.75 ft) thick. The top of the CSSF 1 vault ceiling slab is approximately at the same level as the surrounding roads. Access to the CSSF 1 storage vault is through the cyclone vault. This requires the removal of the two-part hatch cover to the cyclone vault.

In September 1973, the CSSF 1 instrument room (CPP-639) became contaminated because of a pressurization release (ACC 1973).[25] The solids storage bins became pressurized because a closed relief valve that was not completely reseated after it was opened several weeks earlier, and as a result, contamination leaked from a vacuum breaker relief valve located in CPP-639. CPP-639 survey results indicated 1,500 cpm ß + γ. Because CSSFs 1, 2, and 3 are interconnected, smears were taken of CSSFs 2 and 3. The CSSF 2 instrument room (CPP-646) was found to have contamination from 1,200 to 1,600 cpm ß + γ on the equipment and floor surfaces; however, the Significant Operating Occurrence Report (ACC 1973) concluded it was impossible to determine how the facility was contaminated because contamination may have been tracked from CSSF 1 or may have occurred from a separate event. Both CPP-646 and CPP-639 were decontaminated. It was determined contamination likely was not released to the environment (ACC 1973).

In April 1979, an estimated 250 to 750 gal of water entered the CSSF 1 vault. The source of the water was a nearby underground water line that froze and failed during construction of the NWCF. The water likely entered the CSSF 1 storage vault through belowgrade piping penetrations in the vault roof and through the joints between the vault's roof and walls. The CSSF 1 vault is the only CSSF vault that was not waterproofed with an exterior, belowgrade, bituminous coating to prevent in-leakage of water, and it is the only CSSF that has belowgrade piping penetrations. The ruptured water line was repaired, and the water level in the vault decreased over time without operational intervention. The water level did not rise high enough to touch the bins or interrupt convective cooling airflows. Samples taken of the water contained no radioactive contamination. In 1995, a remote video inspection of the vault confirmed the undisturbed condition of the bins and the good condition of the concrete vault walls (Staiger and Swenson 2021).

2.11.1.2 CSSF 2

CSSF 2 structures are composed of the storage vault (CPP-742), cyclone vault (CPP-744), instrument room (CPP-646), and cooling air stack. CSSF 2 consists of seven bins (VES-WCS-136-1 through VES-WCS-136-7), a distributor pipe (VES-WCS-137), a cyclone (CYC-WCS-911), and transport lines to

25. It is likely that ACC (1973) incorrectly described the release occurring in the CSSF 1 cyclone vault (CPP-729). The relief valves are located in the CSSF 1 instrument room (CPP-639), and a release in this location is corroborated by interviews with project personnel, reviews of CSSF 1 drawings, and radiological controls implemented in the instrument room.

where they were cut and capped during HWMA/RCRA closure of the WCF. The bins and distributor pipe are in the storage vault structure (CPP-742), which is located wholly underground within the surrounding berm. The CSSF transport lines are discussed in Subsection 2.11.1.7. Figure 2-32 shows a cross section of CSSF 2. This Draft CSSF 3116 Basis Document covers the CSSF 2 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe and cyclone.

Each CSSF 2 bin is a circular cylinder made of Type 304 stainless steel. The bins are approximately 12.9 m (42.3 ft) tall and 3.7 m (12 ft) in diameter. The bin walls are 6.4 mm (0.25 in.) thick. Total usable calcine storage volume of CSSF 2 is approximately 850 m^3 (30,017.47 ft^3), of which 100% is used. Figure 2-33 is a construction photograph that shows VES-WCS-136-2 prior to installation.

CSSF 2 bins are arranged with six bins forming a circle around the seventh, central, bin. The bin access lines from the six outer bins rise vertically from the calcine retrieval nozzle. The access line from the center bin (VES-WCS-136-1) rises at an angle of 6.5 degrees from vertical and goes through two 30-degree bends in a shielded structure on top of the CSSF 2 storage vault. This arrangement was necessary to terminate the riser in an accessible location on top of the vault instead of terminating the riser inside the concrete cell that houses the CSSF 2 cyclone, which is located directly above the center bin. Each bin has a 6-in. nozzle on the top of the bin, which provides access to the bin for calcine removal. Each nozzle is attached to an access riser that extends approximately 8.2 m (27 ft) through the roof of the storage vault to the top of the CSSF, where it terminates beneath a removable concrete shielding block.

The CSSF 2 cylindrical storage vault structure floor slab is approximately 1.5 m (5 ft) thick and 15.9 m (52 ft) in diameter. The slab is founded on the bedrock beneath the facility. The cylindrical storage vault structure exterior of CSSF 2 is approximately 18.8 m (61.8 ft) tall, excluding the base slab. CSSF 2 has an outer diameter of 15.2 m (50 ft) with a wall thickness of approximately 0.61 m (2 ft). The roof thickness is approximately 0.76 m (2.5 ft). The exterior height of CSSF 2 extends 6.9 m (22.7 ft) above the existing grade. CSSF 2 is surrounded by an earthen berm for radiation shielding that extends from the top of the storage vault structure exterior to the ground. This berm geometrically resembles an annular frustum that horizontally extends approximately 2.4 m (8 ft) from the top edge of CSSF 2 before sloping toward the ground.

Principal internal bin obstructions are thermowells, thermowell supports, and corrosion coupons.[26] The thermowell contains a series of thermocouples that were used to determine the calcine level while the bins were being filled and to monitor the calcine temperature thereafter. Each bin has four internal stiffening rings that extend 5.25 in. from the bin wall and are fabricated from Type 304 stainless steel. The four stiffening rings are spaced roughly equidistant down the bin wall.

Two bins, VES-WCS-136-1 and VES-WCS-136-4, have corrosion coupons that are mounted on cables suspended from wall-mounted hangers and J-hooks within the calcine retrieval riser. One bin has four coupon cables, and the other has three. Prior to sampling the calcine stored in VES-WCS-136-3 and VES-WCS-136-7 in 1978, temporary bin ventilation equipment was installed on the riser connected to VES-WCS-136-1. At that time, a cable containing corrosion coupons in the VES-WCS-136-1 riser was inadvertently dropped into the bin. The cable and coupons were never recovered, and they remain in the bin (Staiger and Swenson 2021).

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26. The purpose of these coupons is to provide data on performance of the materials of construction in terms of corrosion. During January 1966, 160 coupons were hung on 10 stainless-steel cables in two empty bins in CSSF 2 (i.e., VES-WCS-136-1 and VES-WCS-136-4). Each cable supported 16 coupons, four each of Type 405, 304, and 304L stainless steel and four of Type 1025 carbon steel. Each cable contained eight welded cylinders and plates fabricated from mill-certified steels. The cylinders were rolled from a 1/8-in.-thick plate into welded tubes 15 cm (6 in.) long. The plate coupons were fabricated from 1/4-in.-thick materials. For the plate coupons, two pieces of metal 3.8 cm (1-1/2 in.) wide by 13 cm (5 in.) long were butt-welded to form a 7.6- × 13-cm (3- × 5-in.) coupon. All coupons were welded using the tungsten inert gas process with the appropriate electrode for the different alloys.

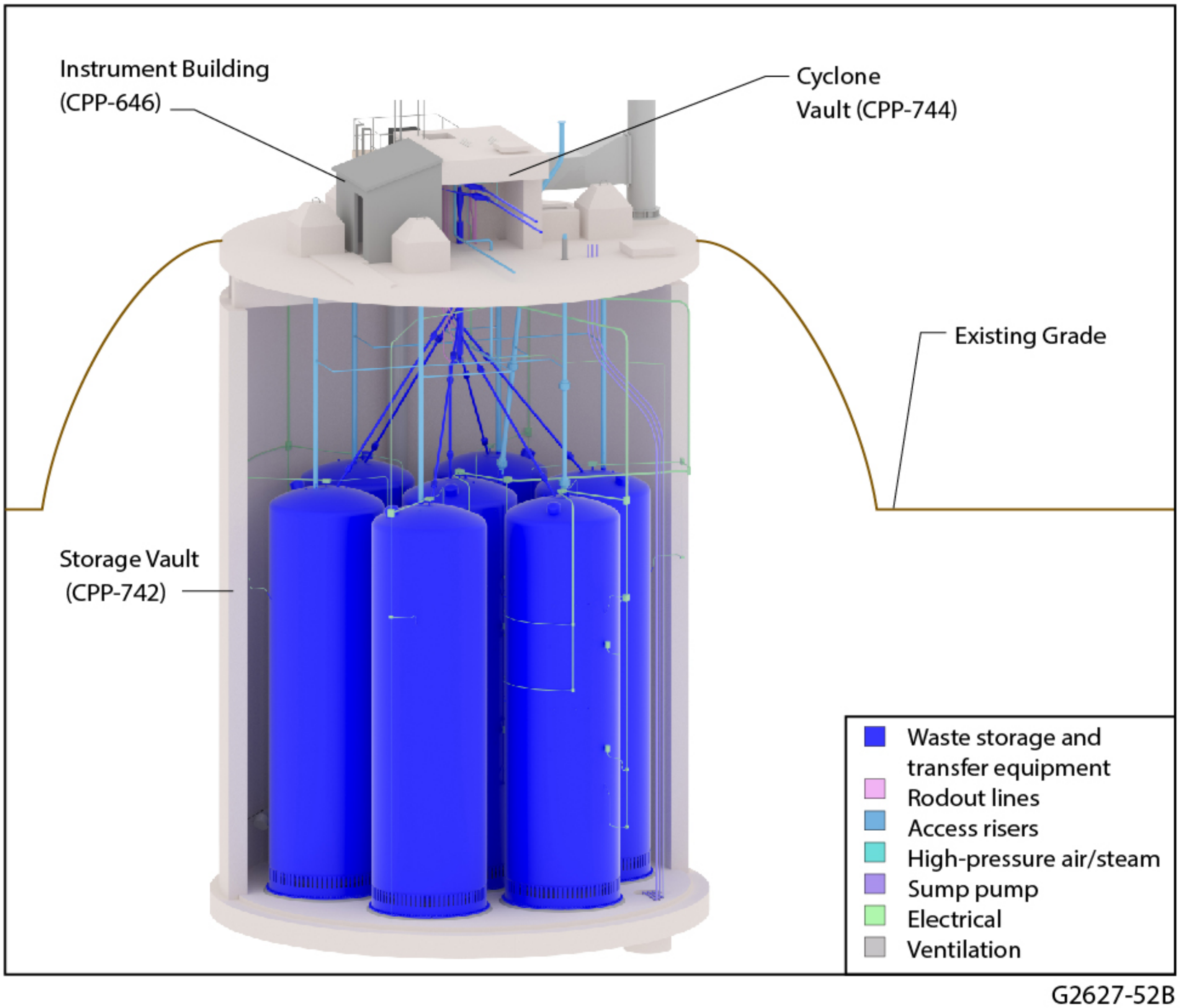

Figure 2-32. Calcined Solids Storage Facility 2 cross section (DOE-ID 2022a).

Figure 2-33. Historical photograph of VES-WCS-136-2 ready to be installed in Calcined Solids Storage Facility 2 (EDMS ID 7188225).

When CSSF 2 was nearly full, the calcine level was measured in two bins using a lead-weighted string. After its use, the weighted string was dropped into the bin to prevent the spread of radioactive contamination. Three such level measurements were made in VES-WCS-136-2: the first on September 20, 1971, the second on October 21, 1971, and the last on November 24, 1971. At least two such measurements were made in VES-WCS-136-6: one on September 20, 1971, and the second on October 21, 1971. The access risers on VES-WCS-136-2 and VES-WCS-136-6 originally had a welded cap. Modifications were made to those risers to accommodate the level measurements in those bins. Access to VES-WCS-136-2 was gained in 1972 for off-gas sampling through a 2-in. screw-cap nipple installed on the terminating weld cap of the retrieval line. Though not documented, the riser into VES-WCS-136-6 was likely modified in a similar fashion (Staiger and Swenson 2021).

In 1978, calcine was retrieved from VES-WCS-136-3 and VES-WCS-136-7 to analyze chemical and physical properties to determine whether, after several years in storage, chemical or physical changes to calcine would impede pneumatic retrieval. The welded caps were cut from the ends of the access risers for those two bins to insert the sample device into the bins. The calcine sampling report did not indicate how the risers were sealed when the sampling work was finished (Staiger and Swenson 2021). Data indicate the calcine remained unchanged after 12 years of storage, and a study concluded pneumatic retrieval of calcine is feasible (ICP 2017). During this sampling event, a section of the stainless-steel sample piping (a 45.7-cm [18-in.] length of 2-in. pipe) was dropped into VES-WCS-136-3 and was never recovered. The calcine sampling operation also found the bin off-gas system was plugged or restricted (Staiger and Swenson 2021).

According to Staiger and Swenson (2021), an erosion failure in the transport piping system or CSSF 2 cyclone vault may have occurred. About the time CSSF 2 was filled and CSSF 3 was placed in service, operational data indicate transport system parameter anomalies occurred. The anomalies are mentioned in the January 27 through February 26, 1972, monthly production report (ACC 1972) and, at the time, were assumed to indicate the bins were full. However, as postulated in Staiger and Swenson (2021), it is possible the anomalies were the result of filling CSSF 2 or breaching the cyclone and transport piping system in a manner like that which occurred in CSSF 3 (see Subsection 2.11.1.3). Staiger and Swenson (2021) describe CSSF 2 as being filled to operating capacity; however, bulk density calculations from the HPM[27] for the last bin filled (VES-WCS-136-6) is "markedly lower than that of the other segments, suggesting that segment and bin are not completely filled and the transport system anomalies may have been the result of a transport system failure." Possible failure of the transport piping is further supported by a release from CSSF 2 that occurred in December 1975 when the facility became slightly pressurized (ACC 1975). Radioactive contamination was found around the sleeves for the extension rods that operate the CSSF cyclone inlet and outlet valves. It was supposed that the valve bellows inside the CSSF 2 cyclone vault may have failed and released contamination when the facility pressurized. An inspection of the CSSF 2 cyclone vault to verify the integrity of the cyclone and associated piping and valves has never been performed.

2.11.1.3 CSSF 3

CSSF 3 is composed of the storage vault (CPP-746), cyclone vault (CPP-747), instrument room (CPP-647), and cooling air stack. The CSSF 3 system consists of seven bins (VES-WCS-140-1 through VES-WCS-140-6 and VES-WCS-139), a distributor pipe (VES-WCS-141), a cyclone (CYC-WCS-912), and transport lines to where they were cut and capped during HWMA/RCRA closure of the WCF. The bins and distributor pipe are in the storage vault (CPP-746), which is located wholly underground within the surrounding berm. The CSSF transport lines are discussed in Subsection 2.11.1.7. Figure 2-34 shows a cross section of CSSF 3. This Draft CSSF 3116 Basis Document covers the CSSF 3 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ.

27. The HPM uses multiple sources to provide definitive calcine volume, mass, and composition. The HPM is further described in Subsection 2.11.3.

Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe.

CSSF 3 contains seven bins fabricated from Type 304 stainless steel: VES-WCS-140-1 through VES-WCS-140-6 and VES-WCS-139. VES-WCS-139 has an equipment number that is not in sequence with the other CSSF 3 bins because it is physically different (longer) than the other six bins, which are identical in size and shape. However, VES-WCS-139 has been renamed VES-WCS-140-7 in some facility documents for numbering consistency with the other bins, which has generated some confusion. This Draft CSSF 3116 Basis Document uses VES-WCS-139.

CSSF 3 bins are arranged and constructed much like those in CSSF 2, except the center bin (VES-WCS-139) is taller than the other six, as depicted in Figure 2-35. Six of the 3.6-m-diameter (12-ft-diameter) bins are approximately 16.2 m (53 ft) tall. The seventh center bin (VES-WCS-139) is 18.6 m (61 ft) tall. The bin wall thickness varies from 6.4 mm (0.25 in.) at the top to 11.11 mm (0.4375 in.) at the bottom. Total usable calcine storage volume of CSSF 3 is approximately 1,130 m^3 (39,905.57 ft^3), of which 1,120 m^3 (39,552.43 ft^3) (99%) is used. Each bin is fitted with a 6-in. nozzle located on the top of the bin, which provides access to the bin for calcine removal. An access riser is attached to each nozzle and extends approximately 5.5 m (18 ft) up through the vault roof, where the riser terminates beneath a removable concrete shield block. The access line for the center bin (VES-WCS-139) is angled 13.5 degrees from vertical, like that of CSSF 2, to avoid having the access line terminate inside the cyclone vault, which is located directly above the center bin.

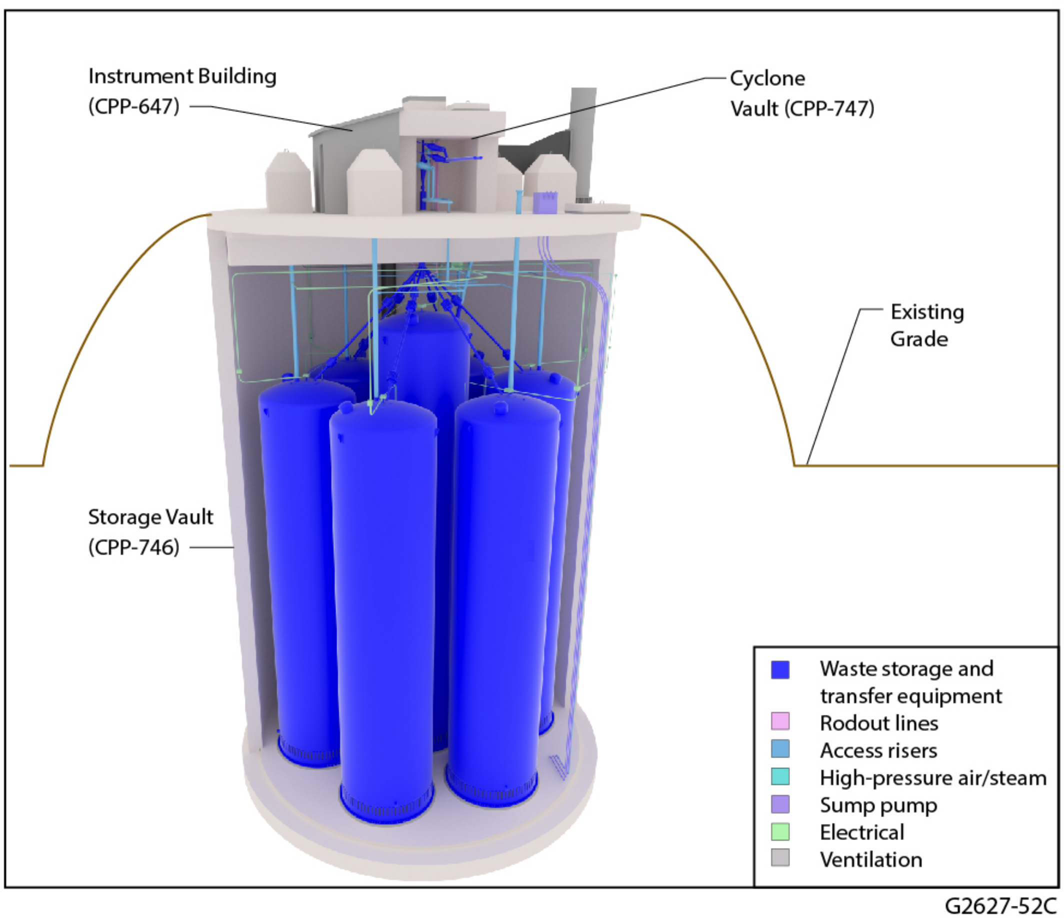

Figure 2-34. Calcined Solids Storage Facility 3 cross section (DOE-ID 2022a).

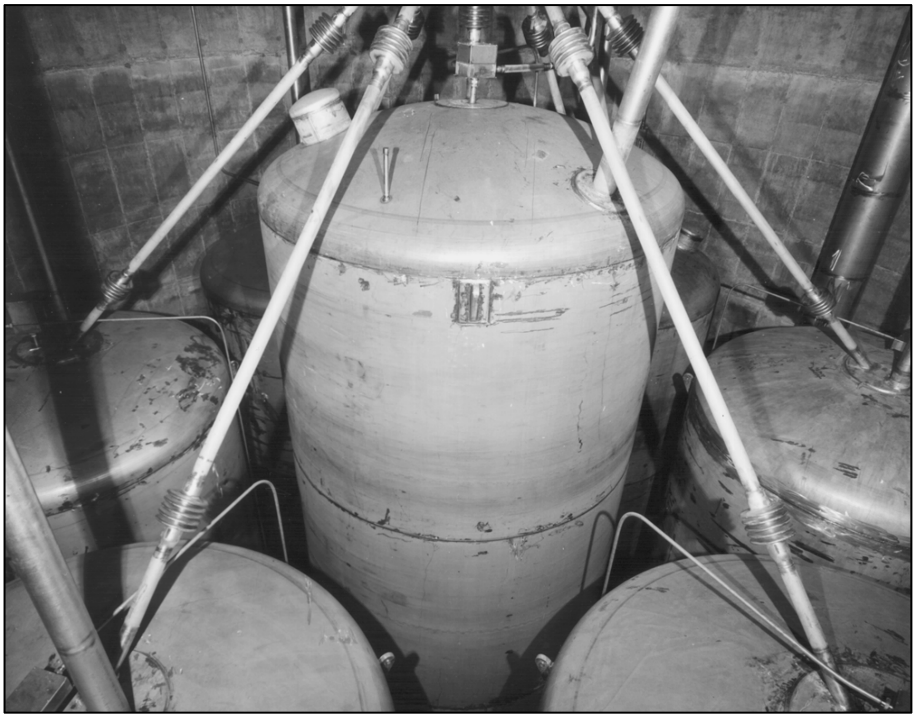

Figure 2-35. Historical photograph of center bin VES-WCS-139 surrounded by six other bins in Calcined Solids Storage Facility 3 (EDMS ID 7188225).

The CSSF 3 cylindrical storage vault structure floor slab is approximately 1.5 m (5 ft) thick and 15.9 m (52 ft) in diameter. The slab is founded on the bedrock beneath the facility. The cylindrical storage vault structure exterior of CSSF 3 is approximately 20.6 m (67.7 ft) tall, excluding the base slab. CSSF 3 has an outer diameter of 15.2 m (50 ft) with wall thickness of approximately 0.61 m (2 ft). The roof thickness is approximately 0.76 m (2.5 ft). The exterior height of CSSF 3 extends 8.2 m (26.8 ft) above the existing grade. CSSF 3 is surrounded by an earthen berm for radiation shielding that extends from the top of the storage vault structure exterior to the ground. This berm geometrically resembles an annular frustum that horizontally extends approximately 2.4 m (8 ft) from the top edge of CSSF 3 before sloping toward the ground.

The principal internal obstructions in CSSF 3 are thermowells, thermowell supports, and corrosion coupons. Each bin contains a centerline-mounted thermowell that extends from the top of the bin to nearly the bottom of the bin. The thermowell contains a series of thermocouples that were used to determine the calcine level while the bins were being filled and to monitor the calcine temperature thereafter. In addition to the centerline thermowell, VES-WCS-140-1 has four wall-mounted thermowells, two on each side of the vessel that enter the vessel wall near the middle of the bin and are attached to the inside surface of the bin wall approximately 1.3 m (4 ft) above and below the point at which they enter the bin. VES-WCS-140-1 also has an array of seven thermocouples that were designed to monitor the calcine temperature profile horizontally through the bin, like those found in two of the CSSF 2 bins (i.e., VES-WCS-136-1 and VES-WCS-136-4). Bins VES-WCS-140-1 through VES-WCS-140-6 each have six internal stiffening rings, while bin VES-WCS-1139 has eight, that are all spaced roughly equidistant down the bin walls. Each of the stiffening rings extend 5.25-in. from the bin wall and are fabricated from Type 304 stainless steel. Bin VES-WCS-140-1 has five sets of Type 405, 304, and 304L stainless steel and Type 1025 carbon steel corrosion coupons, similar to the coupons previously described for the CSSF 2 bins. The corrosion coupons are installed on stainless-steel cables that are secured with 1/4-in. stainless-steel J-hooks welded to the inside of the bin access riser 10.1 to 12.7 cm (4 to 5 in.) from the top of the riser.

Frequent plugging of the calcine transport system and CSSF 3 cyclone occurred while filling CSSF 3. The piping was cleared with an auger-type device on several occasions. Although no documentary evidence was found, workers remember losing sections of wire auger cable on at least two occasions (Staiger and Swenson 2021).

Two erosion failures and releases of calcine occurred in the CSSF 3 cyclone vault at the elbow of the cyclone inlet solids transport line. Erosion also caused a third failure, but no calcine was released because the second repair involved installation of a backup wear pad, which prevented the release of any contamination (Staiger and Swenson 2021). The elbow first failed sometime between May and July 1976 (ACC 1976a) and again a year later in December 1977 (INEL 1978). The first failure was discovered in October 1976 during cold startup of the WCF. At that time, the cyclone (CYC-WCS-912) became restricted with nonradioactive startup material (dolomite) (ACC 1976b). While clearing the cyclone, the cyclone momentarily pressurized, causing a release of dolomite that contaminated the vault roof and portions of the berm northeast of CSSF 3. Further investigation of the cyclone release revealed significant accumulation of calcine in the cyclone vault. The accumulation was determined to be the result of an elbow failure that likely occurred while processing liquid tank farm waste earlier in the year (ACC 1976a). The cyclone vault floor was covered with approximately 3 ft of calcine and a layer of nonradioactive startup material. The cyclone vault required extensive cleanup before the failed piping could be replaced and the outside surfaces of CSSF 3 could be decontaminated. The contaminated berm was covered with a layer of soil. The contaminated berm was later identified in the FFA/CO (DOE-ID 1991) as CERCLA Site CPP-13, and it subsequently was remediated under the OU 3-13 ROD in 2009 (DOE-ID 1999a; DOE-ID 2009). The contaminated soil was excavated and disposed of at the ICDF. The area was then backfilled with clean fill material and contoured to the original surface grade (DOE-ID 2009).

2.11.1.4 CSSF 4

CSSF 4 is composed of the storage vault (CPP-760), cyclone vault and filter room (CPP-761), instrument building (CPP-658), and cooling air stack. The CSSF 4 system consists of three bins (VES-WS4-142 through VES-WS4-144) and a distributor pipe (VES-WS4-145), a cyclone (CYC-WS4-916), and transport lines. The bins and distributor pipe are in the storage vault (CPP-760). The distributor pipe, cyclone, and a portion of the bin system are above the existing grade at INTEC. The CSSF transport lines are discussed in Subsection 2.11.1.7. Figure 2-36 shows a cross section of CSSF 4. This Draft CSSF 3116 Basis Document covers the CSSF 4 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe and cyclone.

CSSF 4 contains three bins fabricated from Type 304 stainless steel: VES-WS4-142, -143, and -144. The CSSF 4 bins are cylindrical, like those of CSSFs 2 and 3, except that the stiffening rings are attached externally. Each bin is approximately 16.8 m (55 ft) tall and 3.7 m (12 ft) in diameter. The bin walls range in thickness from 9.53 mm (0.375 in.) at the top to 15.9 mm (0.625 in.) at the bottom. The total usable calcine storage volume of CSSF 4 is approximately 486 m^3 (17,162.93 ft^3), of which 100% is used. Each bin is fitted with two 6-in. nozzles located on the top of the bin that provide access to the bin for calcine removal. Attached to each nozzle is a 5.5-m-long (18-ft-long) bin access riser, which penetrates the roof of the bin vault where it is covered with a removable concrete shield block. Figure 2-37 shows Bin VES-WS4-142 during placement in the CSSF 4 storage vault.

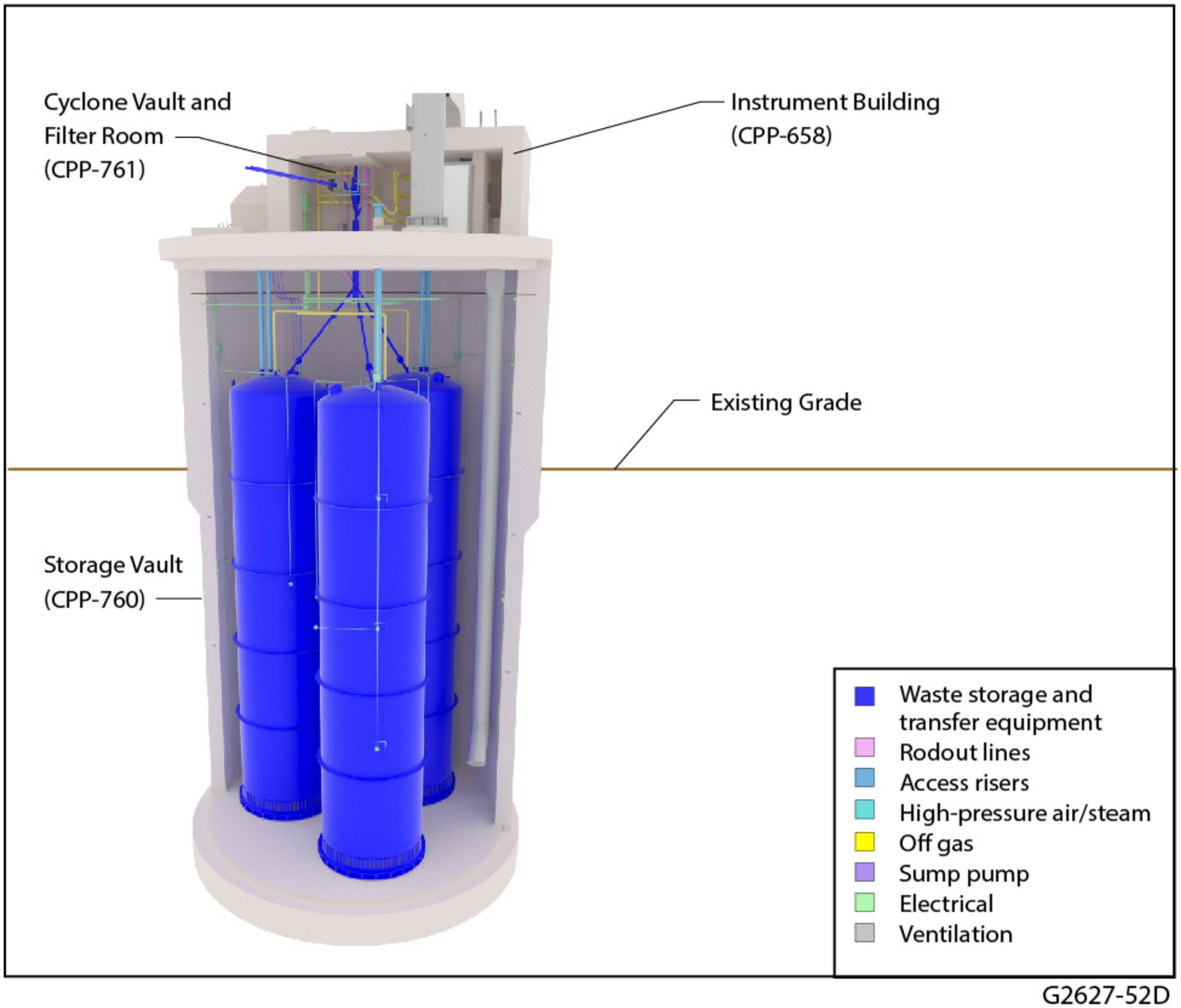

Figure 2-36. Calcined Solids Storage Facility 4 cross section (DOE-ID 2022a).

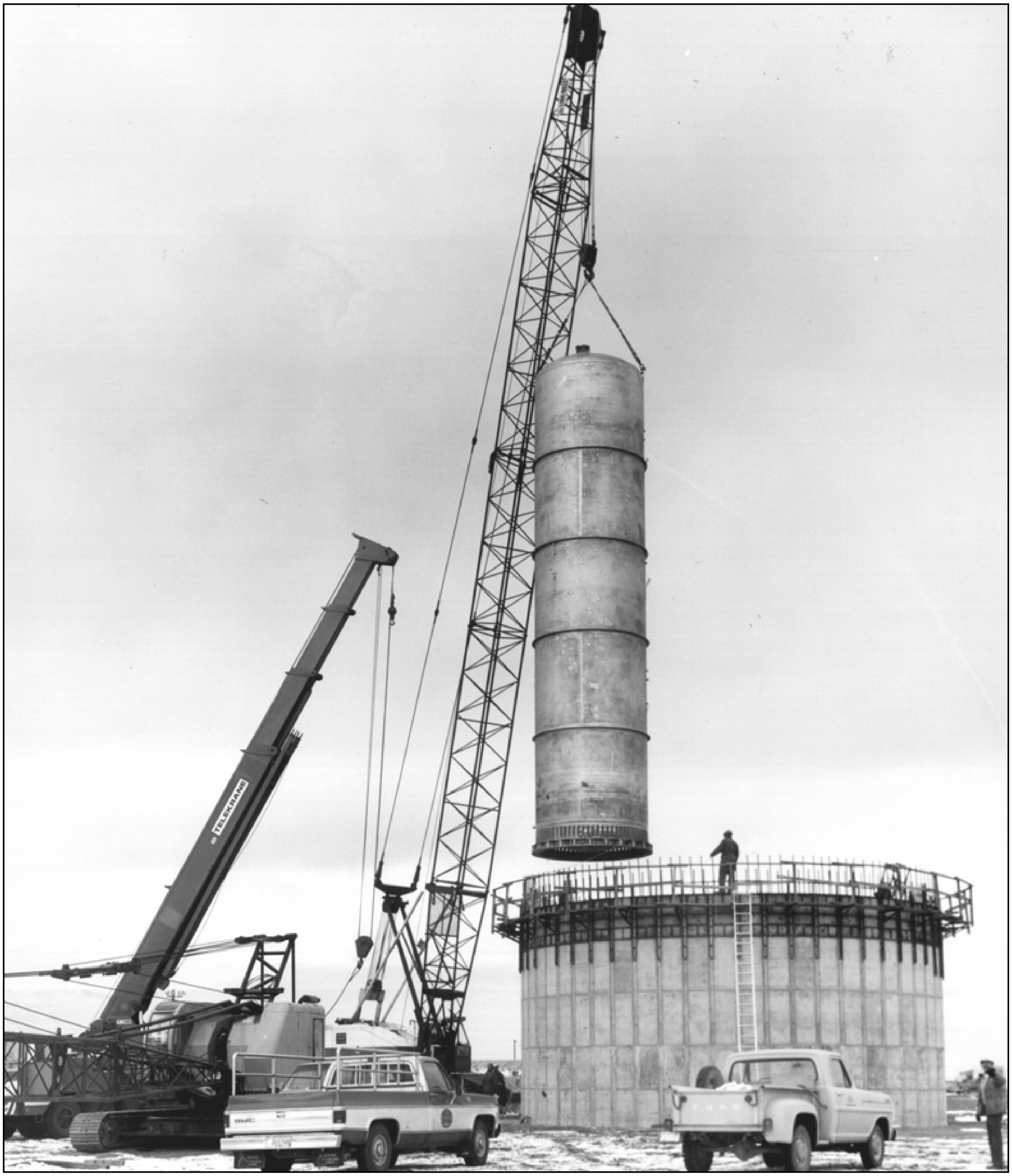

Figure 2-37. Historical photograph of installation of Bin VES-WS4-142 in the Calcined Solids Storage Facility 4 storage vault (EDMS ID 7188224).

The CSSF 4 cylindrical storage vault floor slab is approximately 1.4 m (4.5 ft) thick and 12.8 m (42 ft) in diameter. The slab is founded on the bedrock beneath the facility. The exterior of CSSF 4 is 21.4 m (70 ft) tall, excluding the base slab, and has an overall inner diameter of 11.0 m (36 ft). The lower storage vault structure section wall thickness is 0.61 m (2 ft) and vertically extends 12.2 m (40 ft) from the base slab. The outer diameter of this lower storage vault structure section is 12.2 m (40 ft). The upper storage vault structure section wall thickness is 1.1 m (3.5 ft) and extends from 12.2 m (40 ft) above the base slab to the rooftop of CSSF 4. The outer diameter of this upper storage vault structure section is 13.1 m (43 ft). The exterior height of CSSF 4 extends 8.0 m (26.3 ft) above the existing grade, with the remaining 13.3 m (43.7 ft) located below the existing grade. Six precast reinforced-concrete beams support the storage vault structure roof.

Internal obstructions include thermowells, thermowell supports, and corrosion coupons. Each CSSF 4 bin contains a centerline-mounted thermowell that extends from the top of the bin nearly to the bottom of the bin. The thermowell contains a series of thermocouples that were used to determine calcine levels while the bins were being filled and to monitor calcine temperature thereafter. Type 304L stainless-steel corrosion coupons, similar to the coupons previously described for the CSSF 2 bins, are installed in one of the two access nozzles in each bin. The coupons are secured to 1/4-in. stainless-steel J hooks that are welded to the inside of the riser 10.1 to 12.7 cm (4 to 5 in.) from its opening.

CSSF 4 bin vent filters and associated pressure relief valves are highly contaminated. They were contaminated during a total loss of electrical power at the NWCF in October 1982 during the first NWCF operating campaign (Staiger and Swenson 2021). The NWCF off-gas blowers failed with the loss of electrical power, but the compressed air system continued to add air to the calciner process systems via sparges, instrument air purges, jets, airlifts, and other equipment. As a result, the entire calciner process system pressurized, and the pressure was relieved through the CSSF 4 bin pressure relief valves and vent filters. The filter system contamination was discovered in January 1983, and the contact radiation reading on the filters was 125 R/hour. In 1986, consideration was given to changing the contaminated filters. A radiation survey found the vent filters had a contact radiation reading of up to 50 R/hour and a general body field of 12 R/hour at .46 m (1.5 ft) from the filter. A decision was made not to change the contaminated equipment at that time because of the high radiation fields and as low as reasonably achievable (ALARA) concerns (SAR-105).

2.11.1.5 CSSF 5

CSSF 5 comprises the storage vault, the cyclone vault, and the upper storage vault structure, which contains the instrument room and cooling blower (CPP-765); the cyclone vault access building (CPP-671); and cooling air stack. The CSSF 5 system consists of seven bins (VES-WS5-146 through VES-WS5-152), a distributor pipe (VES-WS5-153), a cyclone (CYC-WS5-917), and transport lines. The bins and distributor pipe are in the storage vault (CPP-765). The distributor pipe, cyclone, and a portion of the bin set system are above the existing grade at INTEC. The CSSF transport lines are discussed in Subsection 2.11.1.7. Figure 2-38 shows a cross section of CSSF 5. This Draft CSSF 3116 Basis Document covers the CSSF 5 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe and cyclone.

CSSF 5 contains seven annular bins fabricated from Type 304L stainless steel: VES-WS5-146 through VES-WS5-152, arranged much like those in CSSFs 2 and 3. CSSF 5 fabrication documents identified the bins with the identifier “WCS” instead of “WS5” that is used in INTEC documents for CSSF 5. This Draft CSSF 3116 Basis Document uses “WS5.”

CSSF 5 bins are significantly different from those of CSSFs 2, 3, and 4. CSSF 5 bins have an annular instead of cylindrical design. They are a cylinder with a hole running through the length of the bin, like an elongated donut. The annular bin design provides more bin surface area for heat dissipation and could accommodate waste with a higher heat generation than that of a cylindrical design.

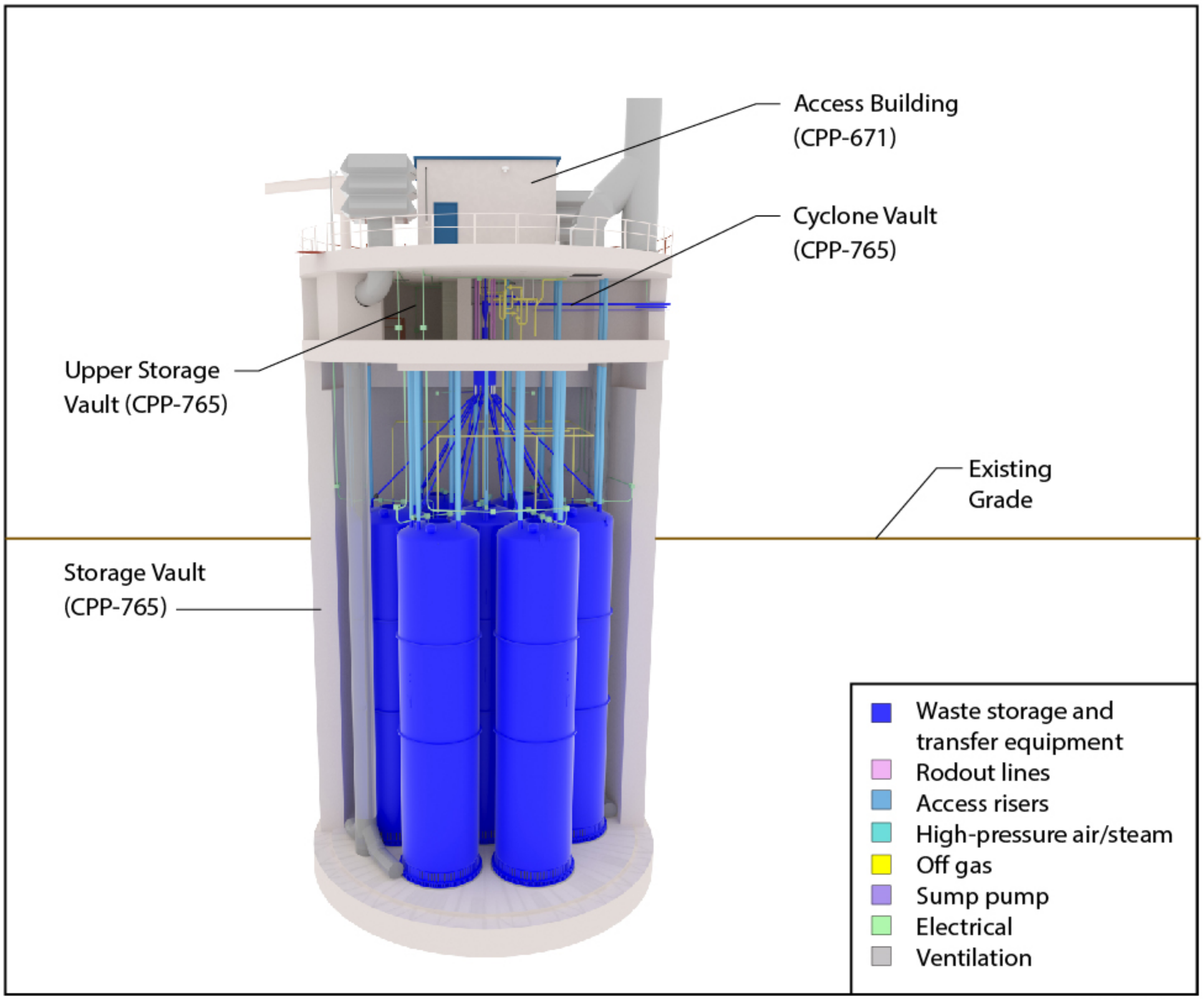

Figure 2-38. Calcined Solids Storage Facility 5 cross section (DOE-ID 2022a).

Each CSSF 5 bin is approximately 16.8 m (55 ft) tall with a 3.7-m (12-ft) outer and a 1.2-m (4-ft) inner diameter. The outer wall thickness varies from 15.9 mm (0.625 in.) at the bottom to 9.53 mm (0.375 in.) at the top, while the inner wall thickness is 9.53 mm (0.375 in.) throughout. The usable calcine storage volume of CSSF 5 is approximately 1,010 m^3 (1,321 yd^3), of which 100% is used. Each bin is fitted with four 8-in. nozzles located approximately 90 degrees apart, which provide access to the bin for calcine removal. Each of the nozzles is connected to an 8-in. access riser that extends vertically from the nozzle and is capped with a blinded, weld-neck flange. Twenty-two of the risers are approximately 7.3 m (24 ft) long (all the risers from Bins VES-WS5-148 through -152 and two of the risers from VES-WS5-146) and terminate in the roof of the bin vault (the CSSF 5 instrument room floor), where they are covered by removable shield plugs. The four risers from Bin VES-WS5-147 are approximately 11 m (36 ft) long, extend through the cyclone vault, and terminate in the CSSF 5 roof, where they are covered with a removable shield plug. Two risers from Bin VES-WS5-146 are approximately 11.3 m (37 ft) long, extend through the cyclone vault, and terminate on the CSSF 5 roof in the CSSF 5 equipment building, where they are equipped with a shielded plug.

The CSSF 5 cylindrical storage vault structure floor slab is approximately 1.5 m (5 ft) thick and 17.4 m (57 ft) in diameter. The storage vault structure slab is founded on the bedrock beneath the facility. The exterior of CSSF 5 is approximately 28.1 m (92.2 ft) tall, excluding the base slab, and has an outer diameter of 16.8 m (55 ft). The storage vault wall thickness is 1.2 m (4 ft). The CSSF 5 storage vault structure has been designed and constructed to prevent any water infiltration. The storage vault walls were placed in one continuous pour from the base slab up to the transition to the access cell wall using the slip-form technique (a moving formwork so that the concrete can be poured continuously). The storage vault structure excavation was backfilled to slightly above grade. Backfill was added in layers and compacted to at least 90% of maximum density. There are no storage vault wall penetrations at an elevation that would allow entry of water, even during a design basis flood, which is based on the failure of the Mackay Dam (Ostenaa and O'Connell 2005).

Figure 2-39 is a construction photograph that shows the top of VES-WS5-147 in the CSSF 5 storage vault. It shows the four 8-in. bin access nozzles and associated bin access risers (vertical lines), a 3-in. vent line, two (sloping) bin fill lines (attached to the bin access risers just above the top of the bin), and two thermowells that penetrate the top of the bin. Construction plywood surrounds the outer portion of the bin and covers the annular hole through the center of the bin.

Figure 2-39. Historical photograph of the top of VES-WS5-147 in the Calcined Solids Storage Facility 5 storage vault (EDMS ID 7188224).

Internal obstructions include thermowells, thermowell supports, and corrosion coupons. Each CSSF 5 bin contains two thermowells that extend from the top of the bin nearly to the bottom. The thermowells are mounted on the centerline of the annular portion of the bin (midway between the outer and inner wall), with one thermowell on each side of the annular bin. The thermowells contain a series of thermocouples that were used to determine the calcine level while the bins were being filled and to monitor the calcine temperature thereafter. Each bin also has several thermocouples mounted on the outside of the bin that monitor the temperature of both the inner and outer bin wall.

Bins VES-WS5-149 and -151 contain five sets of Type 304L stainless-steel corrosion coupons similar to those of CSSFs 2, 3, and 4. However, unlike CSSFs 2, 3, and 4, where the corrosion coupons are in the bin access nozzles and risers designated for calcine removal, the corrosion coupons in CSSF 5 are in their own dedicated 6-in. nozzles and risers, separate from the 8-in. calcine retrieval nozzles and risers.

There are no known historical releases from CSSF 5 to the facility or environment.

2.11.1.6 CSSF 6

CSSF 6 is composed of the storage vault (CPP-791), a service building (CPP-673) (which houses the cyclone cell room, the cyclone cell access room, the fan room, the instrument room, the off-gas filter room, the exhaust plenum room, the inlet plenum room, and equipment building), and a cooling air stack. The CSSF 6 system consists of seven bins (VES-WS6-154 through VES-WS6-160), a distributor pipe (VES-WS6-161), a cyclone (VES-WS6-918), and transport lines. The bins and distributor pipe are in the storage vault (CPP-791). The distributor pipe, cyclone, and a portion of the bins are above the existing grade at INTEC. The CSSF transport lines are discussed in Subsection 2.11.1.7. Figure 2-40 shows a cross section of CSSF 6. This Draft CSSF 3116 Basis Document covers the CSSF 6 bins (including integral equipment), transport lines, and any residual calcine therein that will be disposed of in situ. Integral equipment includes any piping or equipment that has had contact with calcine, such as the distributor pipe.

CSSF 6 contains seven annular bins fabricated from Type 304L stainless steel: VES-WS6-154 through VES-WS6-160. Except for some dimension changes, the configuration of the CSSF 6 bins is very similar to those of CSSF 5. Each CSSF 6 bin is approximately 20.8 m (67.5 ft) tall, with a 4.1-m (13.5-ft) outer and a 1.5-m (5-ft) inner diameter. The outer wall thickness varies from 25.4 mm (1 in.) at the bottom to 9.53 mm (0.375 in.) at the top, while the inner wall thickness is 14.29 mm (0.5625 in.) throughout. The top of each bin is fitted with four 8-in. nozzles, located approximately 90 degrees apart that provide access to the bin for calcine removal. Each of the nozzles is attached to an 8-in. access riser that is capped and blinded with a weld-neck flange. Four of the access risers (from Bin VES-WS6-154) are 11.9 m (39 ft) long and extend to the roof of the CSSF, where they terminate in a recess covered by a removable shield plug. The 24 access risers to the other six bins are 7.9 m (26 ft) long and extend into the calcine storage vault roof (instrument room floor), where they terminate in a recess covered by a removable shield plug. The usable calcine storage volume of CSSF 6 is approximately 1,506 m^3 (1,970 yd^3), of which 713 m^3 (933 yd^3) (47%) is used.

Figure 2-41 is a construction photograph of CSSF 6 and shows the annular bins within the vault; the sloping, bin-fill piping; the vertical calcine retrieval risers; and the thermowells penetrating the tops of the bins. The annular bin design is typical of bins in CSSFs 5 and 6.

The CSSF 6 cylindrical storage vault structure floor slab is approximately 2 m (6.5 ft) thick and 23 m (76 ft) in diameter. The storage vault structure slab is founded on the bedrock beneath the facility. The storage vault structure is approximately 34 m (112 ft) high with an 18.6-m (61-ft) outer diameter. The storage vault walls are 1.3 m (4.25 ft) thick. The storage vault excavation is backfilled to prevent water from pocketing around the storage vault. The CSSF 6 storage vault structure has no penetrations at elevations low enough to allow water ingress during flood events, including the overtopping of Mackay Dam, which represents the hypothetical flood considered to be the most severe flood event possible at the CSSF (SAR-100-1). Required construction joints and belowgrade structural sections are sealed with a polyvinyl water-stop product.

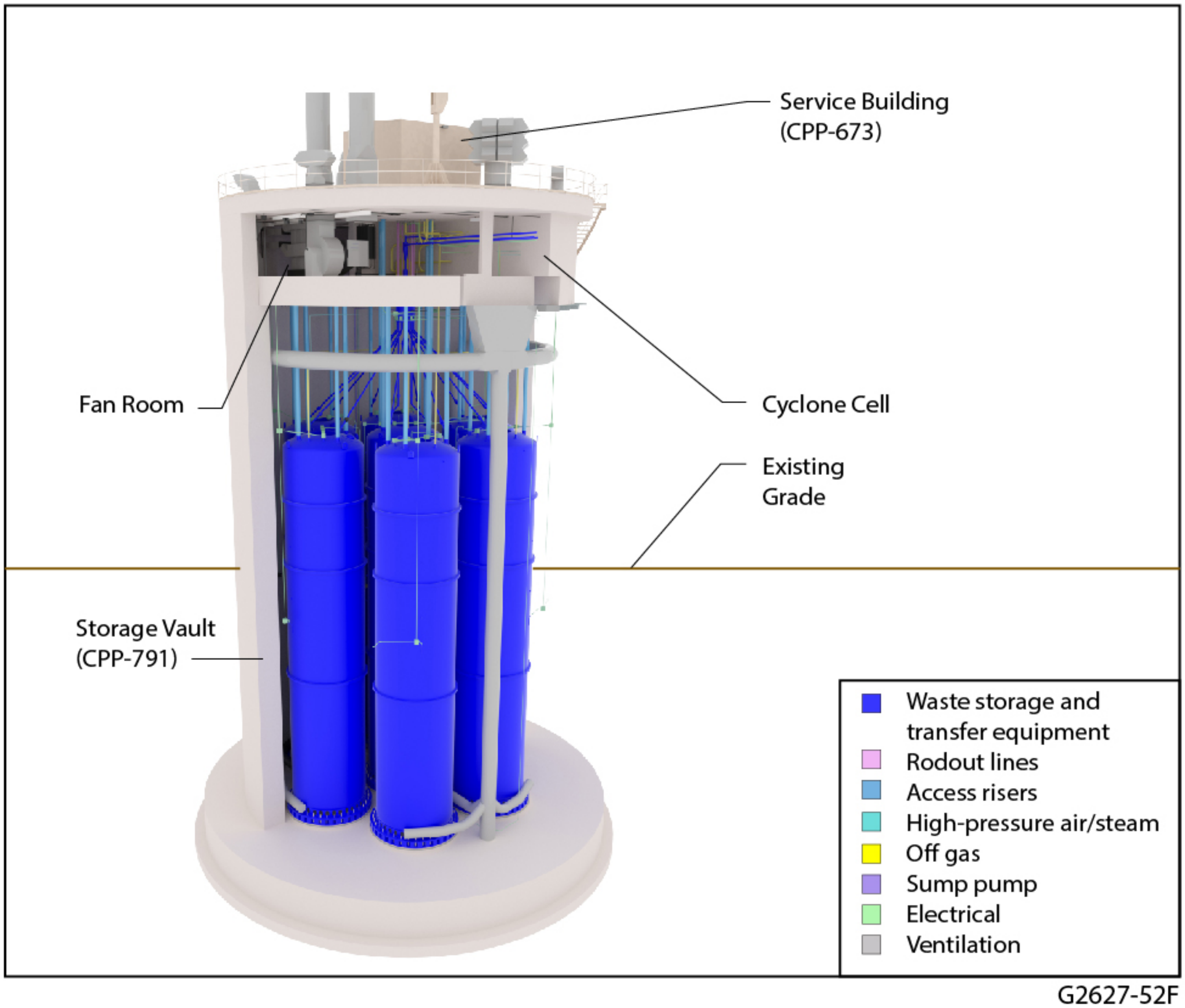

Figure 2-40. Calcined Solids Storage Facility 6 cross section (DOE-ID 2022a).

Figure 2-41. Historical photograph of sloped bin fill piping, vertical retrieval piping, and annular bins in Calcined Solids Storage Facility 6 (EDMS ID 7188224).

Internal obstructions in CSSF 6 include thermowells, thermowell supports, and corrosion coupons. Each CSSF 6 bin contains two thermowells that extend nearly to the bottom of the bin. The thermowells are mounted on the centerline of the annular bin (midway between the outer and inner wall), with one thermowell on each side of the annular bin. The thermowells contain a series of thermocouples that were used to determine the calcine level while the bins were being filled and to monitor the calcine temperature thereafter. Each bin also has several thermocouples mounted on the outside of the bin that monitor the temperature of both the inner and outer bin wall.

Two CSSF 6 bins (VES-WS6-156 and VES-WS6-159) contain five sets of Type 304L stainless-steel corrosion coupons, similar to the coupons previously described for the CSSF 2 bins. As with CSSF 5, the CSSF 6 corrosion coupons are in their own dedicated, 6-in., Schedule 40 risers, separate from the 8-in. calcine retrieval risers.

In August 1984, a release of calcine into a working tent occurred while connecting CSSF 6 transport lines to the main trunk line from the NWCF. The transport line pressurized during welding operations, blowing calcine dust into the work tent and inside the CSSF 6 bin area (Mairson 1984). It was reported that little or no contamination escaped to the environment, based on survey results (WINCO 1984), and the contaminated areas were decontaminated after the event.

2.11.1.7 CSSF Transport Lines

Historical WCF and NWCF liquid waste processing operations were evaluated to assess the functionality of the pneumatic transport lines, their configuration, integrity, and potential residual calcine that may remain in the lines after calcining operations ended. Results of the evaluation are documented in EDF-11119, "Calcined Solids Storage Facility Transport Line Evaluation Summary." The following subsections summarize details from the above-referenced evaluation except as otherwise noted.

2.11.1.7.1 Transport Line Configuration—Calcine was pneumatically transferred from the calcining facilities to the CSSF. Like the bin sets, the transport lines vary in size, length, and depth because of the different construction times of, and design changes to, each CSSF. Generally, though, the transport lines share some common features: (1) each CSSF has a set of two 3-in. stainless-steel lines—a solids transport line that traveled to the CSSF and an air return line that traveled to the calcining facility; (2) the two transport lines (solids transport and air return) were placed in a containment pipe fabricated from either a 14-in. carbon- or stainless-steel pipe or an 18-in. stainless-steel pipe; and (3) reinforced-concrete shielding encases the transport lines and containment pipe. Figure 2-42 shows a generic cross section of the transport line configuration. A 3-in. spare transport line that connected to the NWCF was added to CSSFs 5, 6, and 7.

The transport lines are more than 3 m (10 ft) below the existing grade, starting from the calcining facility buildings; they then travel upward to the cyclone vaults above the existing grade. Figures 2-30 and 2-43 present historical photographs taken during construction, showing the steel containment pipe to the cyclone vault and the concrete form and rebar in place around the steel pipe in preparation for pouring the concrete encasement. The concrete shielding varies in thickness from 15.2 cm (6 in.) at CSSFs 1 through 4, 49.5 cm (19.5 in.) at CSSF 5, and 67.3 cm (26.5 in.) at CSSF 6. The two layers of steel pipe and reinforced concrete results in a robust configuration for the transport lines. Detailed descriptions of the transport line configurations, maps, and drawings are provided in EDF-11119.

CSSF 7 has never received or stored calcine and, therefore, is not subject to regulatory closure. HAD-353, "Hazard Assessment Document for the Calcined Solids Storage Facility 7," states that CSSF 7 contains no radioactive material. In addition, two valves (TAV-WS7-4 and TAV-WS7-5) on the transport lines to CSSF 7 are the boundary for the transport lines. These valves are closed and have a permanent device installed to prevent them from being opened.

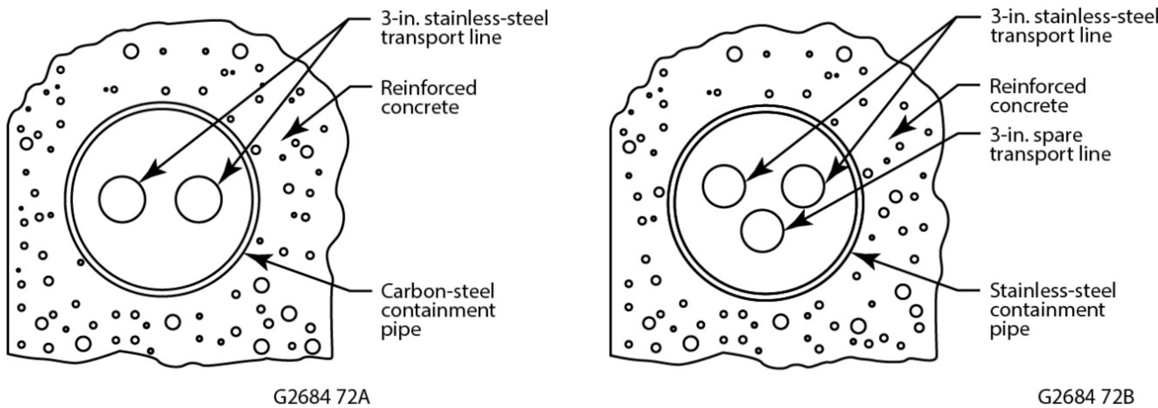

Figure 2-42. Cross section showing typical transport line configuration at the Waste Calcining Facility (left) and New Waste Calcining Facility (right).

Figure 2-43. Calcined Solids Storage Facility 6 historical photograph taken during construction, showing the rebar and concrete form in place around the steel pipe in preparation for pouring the concrete encasement.

2.11.1.7.2 Transport Line Operations—The pneumatic transport system operated with an air velocity high enough to prevent solids from falling out (salting) into the transport line, and generally, the last material sent through each transport line was transport air. During both closures of the calcining facilities (WCF and NWCF), the systems (calcining, processing, and transfer equipment) were scoured with high-velocity air or nonradioactive material to remove residual waste.

In some instances during startup operations, nonradioactive material was sent through the calcining and transport systems, and in other instances, operations switched to the next CSSF without shutting down WCF or NWCF or sending nonradioactive material through the system. The air transport system operated in such a way that deposits or "residual accumulation" of material may have developed in dead space in the transport lines, such as dead legs or solids transport lines no longer in use.[28] Potential deposit or residual-accumulation locations identified in EDF-11119, and discussion regarding whether the deposits would consist of radioactive or nonradioactive material, are included in the following subsections.

2.11.1.7.2.1 Waste Calcining Facility Transport Line Operations—The calcined waste generated as a result of WCF operations filled CSSFs 1, 2, and 3. The transport lines are HWMA/RCRA-permitted components up to where they were cut and capped during HWMA/RCRA closure of the WCF. A portion of the CSSF 1 transport lines (approximately a 6.1- to 9.1-m [20- to 30-ft] section) was removed during the HWMA/RCRA closure because it contained deposits with CSSF 2 nonradioactive startup material. Figure 2-44 shows the portion that was removed, as well as the two sections (west from the excavation to the WCF and west from CSSF 1 to the excavation) that were grouted and left in place.

Aside from the few potential locations where residual material may have accumulated, the transport lines are assumed to only have a film of calcine residue on their internal surfaces. During the 1981 shutdown of the WCF, high-pressure air was introduced into the system to sweep loose solids out of the WCF system to the calcine storage bins. A significant amount of solids was removed from the processing system, and prior to the final processing run in 1981 that preceded shutdown, the WCF processing system was flushed and cleaned (with air) for substantial maintenance and construction work (Archibald and Demmer 1995). Analyses provided in Archibald and Demmer (1995) calculates residual calcine in the WCF processing system. Although the calculations did not evaluate the transport lines outside the WCF, transfer piping within the facility was evaluated and the method used is assumed to be applicable to future residual calculations for the transport lines due to similar processes used in both groups of piping. The approach to calculate residual waste in the WCF system was accepted by the Idaho DEQ for closure of the WCF (Monson 1997).

Potential deposits or residual accumulation locations are in the CSSF 2 solids transport line 3" TAA-3030 and the stub-outs of CSSF 3 solids transport line 3" TAA-3039 where system flushing during closure of WCF may not have completely removed any residual accumulations (see Figure 2-44). Each of these potential deposits is likely radioactive material because processing operations switched to filling CSSFs 2 and 3 without using nonradioactive material (see Table 2-4). Table 2-4 contains the evaluation summary for transport lines connecting the WCF to CSSFs 1 through 3. The "length" in Table 2-4 refers to the total length of the line for each transport line (solids transport and air return) from where the transport lines were cut and capped to where they enter/exit the cyclone vaults. "Comments" in Table 2-4 refer to information used to infer more about the contents of each line. This information is useful in identifying the type of material likely present (nonradioactive or radioactive) in the transport lines as well as the length of time the waste has been occupying the lines.

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28. Previous reports used the word "plugged" for potential deposits of calcine in the transport lines. This report uses the terms "deposits" or "residual accumulation" to refer to the potential accumulations of calcine (radioactive or non-radioactive) in the transport lines, because the calcining systems operated with an air velocity high enough to prevent solids from falling out (salting) and the calcining systems, including the transport lines, were cleared with high-pressure air or scouring material as part of the HWMA/RCRA closure process.

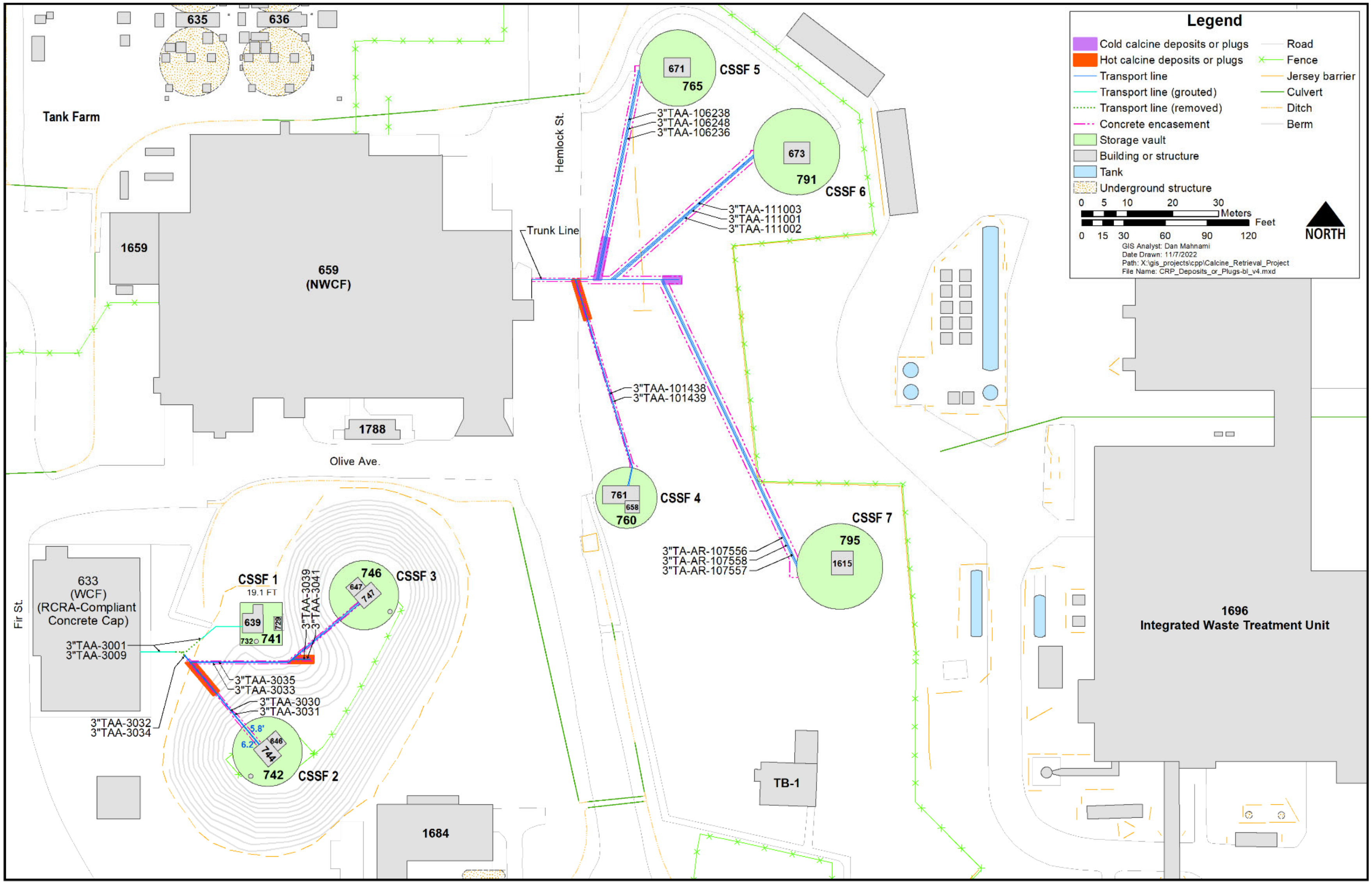

Figure 2-44. Locations of potential deposits of material that may have accumulated in dead spaces of the transport lines during calcining operations.

Table 2-4. Waste Calcining Facility transport lines evaluation summary.

CSSF	Transport Line Number	Transport Line Function	Nominal Pipe Size (in.)	Length m (ft)	Depth m (ft)	Comments
CSSF 1	3” TAA-3001	Air return	3	10.0 (32.8)[a]	a	• CSSF 1 transport lines are Schedule 40 stainless-steel pipe and travel in a 14-in. carbon-steel containment pipe. The transport lines and containment pipe are surrounded by a minimum of 15.2 cm (6 in.) of reinforced concrete. • Filling operations began in 1963 with nonradioactive startup material. • The last material sent to CSSF 1 was radioactive waste in 1964. • In 1966, 3” TAA-3009 was partially filled with nonradioactive material from the CSSF 2 startup operation. This section was later removed during the WCF HWMA/RCRA closure in 1999 (Wessman 1999). A 6.1- to 9.1-m (20- to 30-ft) section of encased transport lines was removed, placed in the WCF operating corridor, and grouted with other building components. 3” TAA-3009 was grouted on both sides of the removed section—west from the excavation point to the WCF and west from CSSF 1 to the excavation point. 3” TAA-3001 was grouted west from CSSF 1 to the WCF (Wessman 1999).
	3” TAA-3009	Solids transport	3	10.0 (32.8)[a]		
CSSF 2	3” TAA-3030	Solids transport	3	31.5 (103.4)[b]	b	• CSSF 2 transport lines are constructed of Schedule 40 stainless steel and travel in a 14-in. carbon-steel containment pipe. The transport lines and containment pipe are surrounded by a minimum of 15.2 cm (6 in.) of reinforced concrete. • Filling operations began in 1966 with nonradioactive startup material. • The last material sent to CSSF 2 was radioactive waste in 1972. • When CSSF 2 was filled, operations switched from CSSF 2 to CSSF 3 with no shutdown; thus, no nonradioactive material was sent through transport lines after CSSF 2 was filled (Staiger and Swenson 2021). It is possible that radioactive material partially filled 3” TAA-3030 (EDF-11119), filling a portion of the line—similar to CSSF 1 transport solids line 3” TAA-3009 when WCF startup
	3” TAA-3031	Air return	3	31.5 (103.4)[b]		

Table 2-4. (continued).

CSSF	Transport Line Number	Transport Line Function	Nominal Pipe Size (in.)	Length m (ft)	Depth m (ft)	Comments
						operations switched from CSSF 1 to CSSF 2 with nonradioactive material.
CSSF 3	3” TAA-3033	Solids transport	3	45.2 (148.4)[c]	c	• Filling operations began in 1972 with radioactive waste. • The last material sent to CSSF 3 was radioactive waste in 1981. • During WCF shutdown in 1981, high-pressure air was introduced into the WCF process equipment to move loose solids into the transport system (Archibald and Demmer 1995). • 3” TAA-3039 and 3” TAA-3041 are stub-outs off the primary transport lines to CSSF 3. The stub-outs were designed and installed as a potential future tie point. • 3” TAA-3039 is likely filled with radioactive material that was deposited and remained when operations switched from CSSF 2 to CSSF 3 while processing radioactive waste (EDF-11119).
	3” TAA-3035	Air return	3	45.2 (148.4)[c]		
	3” TAA-3039	Solids transport	3	5.5 (18.0)[c]		
	3” TAA-3041	Air return	3	5.5 (18.0)[c]		
CSSF 2/3	3” TAA-3032	Solids transport	3	2.0 (6.7)	3.0–3.5 (10–11.5)	• 3” TAA-3032 and 3” TAA-3034 comprise a 2.0-m (6.7-ft) segment between CSSF 1 transport lines and CSSF 2 and 3 transport lines. A portion of the line may have been removed during the WCF closure (see comments specific to CSSF 1 transport lines).
	3” TAA-3034	Air return	3	2.0 (6.7)	3.0–3.5 (10–11.5)	

a. CSSF 1 transport lines from the point of excavation (where the lines were cut and capped) to CSSF 1 are approximately 3 m (10 ft) below the existing grade. The transport lines travel approximately 10.0 m (32.8 ft) before entering the cyclone vault at approximately 1.2 m (3.9 ft) below the existing grade. Total length does not include the 7.8-m (25.6-ft) portion (from the point of excavation to WCF) that was grouted and left in place during HWMA/RCRA closure of the WCF. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

b. CSSF 2 transport lines are approximately 3.5 m (11.5 ft) below the existing grade, where CSSF 2 and 3 transport lines branch. CSSF 2 concrete-encased lines travel 12.2 m (40 ft) southeast to where the transport lines are approximately 6.6 m (21.5 ft) below the existing grade. From that point, the transport lines travel 11.3 m (37 ft) under the berm, sloping upward until they extend beyond the berm. The last section of line after exiting the berm and before entering CSSF 2 is approximately 8.0 m (26.1 ft) long and is 1.9 m (6.2 ft) above the existing grade. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

c. CSSF 3 transport lines are approximately 3.5 m (11.5 ft) below the existing grade, where CSSF 2 and 3 branch. CSSF 3 concrete-encased lines travel 21.5 m (70.7 ft) east and then 2.9 m (9.5 ft) northeast to where the lines are approximately 4.7 m (15.3 ft) below the existing grade. From that point, the transport lines travel 11.3 m (37 ft) under the berm, sloping upward until they extend beyond the berm. The last section of line after exiting the berm and before entering CSSF 3 is approximately 9.5 m (31.2 ft) long and is 1.8 m (5.8 ft) above the existing grade. The GIS database indicates the depth of the 5.5-m (18-ft) dead leg (3” TAA-3039 and -3041) is 10.2 m (33.5 ft) below the original grade; however, this measurement could represent the depth from the foundation bottom. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

CSSF Calcined Solids Storage Facility
GIS geographical information system
HWMA Hazardous Waste Management Act
RCRA Resource Conservation and Recovery Act
WCF Waste Calcining Facility

The transport lines originated approximately 3 m (10 ft) below grade traveling at length, nearly horizontal, with minimal upward slope before taking a more drastic slope upward to enter the cyclone vaults. The cyclone vaults were constructed above the original grade at INTEC. The CSSF 1 storage vault and cyclone vault are wholly covered by a berm. The berm covers CSSFs 2 and 3 storage vaults but does not cover the cyclone vaults. CSSF 1 transport lines are below the berm, and portions of the CSSFs 2 and 3 transport lines are above the berm. Depths of the transport lines are provided in greater detail in the footnotes of Table 2-4.

2.11.1.7.2.2 New Waste Calcining Facility Transport Line Operations—The NWCF transferred calcine to CSSFs 4 through 6. The NWCF calcining system, including the transport lines, was certified as closed under a partial closure plan (Monson 2004), and remaining portions of the building are used for various HWMA/RCRA-permitted operations. Final closure is pending conclusion of current HWMA/RCRA-permitted operations at the NWCF. The transport trunk line, which travels east and west connecting CSSFs 4 through 7 to the NWCF, exits the NWCF approximately 4.2 m (13.9 ft) below grade on the east side of the building (see Figure 2-44 and Table 2-5). The 3-in. stainless-steel transport lines (solids transfer, air return, and spare line) run in a 14- or 18-in. stainless-steel pipe to the cyclone vaults. The stainless-steel secondary containment pipe is completely encased in concrete shielding of varying thickness.

The NWCF transport lines are assumed to have only a film of calcine residue on their internal surfaces (Swenson 2000). After the last waste processing campaign, the NWCF was flushed with a scrub solution that was blended with aluminum nitrate and recycled as final feed to remove radioactivity from the system (Staiger and Swenson 2021). Swenson (2000) calculated residual calcine in NWCF transport lines, and it was assumed based on process knowledge that minimal residues would be in the transport lines due to the "scouring action" of the nonradioactive material last sent through the system. The approach for estimating residual calcine in the NWCF was the same approach used in Archibald and Demmer (1995) for the WCF. The approach was accepted by the Idaho DEQ to support HWMA/RCRA partial closure of the NWCF (Swenson 2000). However, radioactive material that potentially accumulated in the CSSF 4 solids transport line 3" TAA-101438 when operations switched from CSSF 4 to CSSF 5 while processing liquid waste (EDF-11119) may not have been completely flushed out during closure of the NWCF. In addition, the CSSF 5 solids transport line 3" TAA-106238 may be partially filled with nonradioactive material from CSSF 6 startup operations, as is the trunk line beyond CSSF 6, which was likely filled with nonradioactive material during initial NWCF startup operations (EDF-11119). Figure 2-44 shows potential locations of deposits of material.

Table 2-5 contains the evaluation summary for the transport lines connecting NWCF to CSSFs 4 through 7, as well as the CSSF trunk line. Several transport lines associated with storage bins were never used and are denoted as "Spare" under "Transport Line Function" in Table 2-5. The CSSF trunk line refers to the main trunk used to transport solids to CSSFs 4 through 7 from the NWCF. The remaining transport lines branch from this trunk to their respective storage bins. CSSF 7 and its transport lines were never used for the transport and storage of calcine and are considered clean. "Comments" in Table 2-5 refer to information used to infer more about the contents of the transport lines. The information helps identify the type of material likely present (radioactive or nonradioactive) in the lines as well as the length of time the material has been occupying the lines.

Generally, the NWCF transport lines traveled at length, nearly horizontal, with minimal upward slope to the outside of the storage vault. At the storage vault, the transport lines traveled up the outside of the storage vault before entering the cyclone vault. Vertical lengths are accounted for in the total length in Table 2-5. Total length of each transport line (solids transport and air return) is measured from where the transport lines exit and enter the NWCF and cyclone vaults. The depth and length of the transport lines are given in greater detail in the footnotes of Table 2-5.

Table 2-5. New Waste Calcining Facility transport lines evaluation summary.

CSSF	Transport Line Number	Transport Line Function	Nominal Pipe Size (in)	Length m (ft)	Depth m (ft)	Comments
CSSF 4	3” TAA-101438	Solids transport	3	56.2 (184.3)[a]	a	• CSSF 4 transport lines are constructed of Schedule 40 stainless steel and travel in a 14-in. stainless-steel containment pipe. The transport lines and containment pipe are surrounded by a minimum of 15.2 cm (6 in.) of reinforced concrete. • Filling operations began in 1982 with nonradioactive material. • The last material sent to CSSF 4 was radioactive waste in 1983. • When CSSF 4 was filled, operations switched from CSSF 4 to CSSF 5 with no shutdown; thus, no nonradioactive material was sent through the lines after CSSF 4 was filled (Staiger and Swenson 2021). It is possible that radioactive material may have accumulated in 3” TAA-101438 when operations switched from CSSF 4 to CSSF 5 and the residual accumulation may not have been completely flushed out during closure of NWCF.
	3” TAA-101439	Air return	3	56.2 (184.3)[a]		
CSSF 5	3” TAA-106236	Air return	3	62.8 (205.9)[b]	b	• CSSF 5 transport lines are constructed of Schedule 40 stainless steel and travel in an 18-in. stainless-steel containment pipe. The transport lines and containment pipe are surrounded by a minimum of 49.5 cm (19.5 in.) of reinforced concrete. • Filling operations began in 1983 with radioactive waste. • The last material sent to CSSF 5 was nonradioactive material in 1992. • 3” TAA-106238 may have some residual accumulation of nonradioactive material that was deposited and remained when NWCF operations switched from CSSF 5 to CSSF 6 with nonradioactive feed (EDF-11119) and was not completely flushed out during closure of NWCF. • 3” TAA-106248 was never used for transport.
	3” TAA-106238	Solids transport	3	62.8 (205.9)[b]		
	3” TAA-106248	Spare	3	62.8 (205.9)[b]		
CSSF 6	3” TAA-111002	Solids transport	3	61.1 (200.3)[c]	c	• CSSF 6 transport lines are constructed of Schedule 40 stainless steel and travel in an 18-in. stainless-steel containment pipe. The transport lines and containment pipe are surrounded by a minimum of 67.3 cm (26.5 in.) of reinforced concrete. • Filling operations began in 1992 with nonradioactive material.
	3” TAA-111003	Air return	3	61.1 (200.3)[c]		
	3” TAA-111001	Spare	3	61.1 (200.3)[c]		

Table 2-5. (continued).

CSSF	Transport Line Number	Transport Line Function	Nominal Pipe Size (in)	Length m (ft)	Depth m (ft)	Comments
						• The last material sent to CSSF 6 was nonradioactive material in 2000 as a final scour. • 3” TAA-111001 was never used for transport.
CSSF 7	3” TAA-107556	Air return	3	87.8 (288.1)[d]	d	• Material was neither sent to nor stored in CSSF 7.
	3” TAA-107557	Solids transport	3	87.8 (288.1)[d]		
	3” TAA-107558	Spare	3	87.8 (288.1)[d]		
CSSF trunk line	3” TAA-101438	Solids transport	3	32.3 (106.1)	e	• CSSF trunk lines have the same transport line numbers as CSSF 4, but the trunk lines represent the portion of lines that travel east and west, connecting CSSFs 4 through 7 to the NWCF. • Trunk lines were scoured with high-velocity air or nonradioactive material to remove residual waste during final shutdown in 2000. • The trunk line (solids transport 3” TAA-101438) beyond CSSF 6 likely has some residual accumulation of nonradioactive material from NWCF startup operations (EDF-11119).
	3” TAA-101439	Air return	3	32.3 (106.1)		
	3” TAA-106120	Spare	3	32.3 (106.1)		

a. CSSF 4 transport lines are approximately 3.8 m (12.6 ft) below the existing grade, where they branch off the main trunk. The lines travel 43.1 m (141.3 ft) southeast, where the stainless-steel- and concrete-encased lines are approximately 3.0 m (10 ft) below the existing grade. The transport lines then travel up the side of CSSF 4 approximately 13.1 m (43 ft) before entering the cyclone vault. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

b. CSSF 5 transport lines are approximately 3.8 m (12.6 ft) below the existing grade, where they branch off the main trunk. The lines travel 16.9 m (153.9 ft) northeast, and the stainless-steel- and concrete-encased lines are approximately 2.8 m (9.3 ft) below the existing grade. The transport lines then travel up the side of CSSF 5 approximately 15.8 m (52 ft) before entering the cyclone vault. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

c. CSSF 6 transport lines are approximately 3.7 m (12.2 ft) below the existing grade, where they branch off the main trunk. The lines travel 41.2 m (135.3 ft) northeast, and the stainless-steel- and concrete-encased lines are approximately 2.4 m (8 ft) below the existing grade. The transport lines then travel up the side of CSSF 6 approximately 19.8 m (65 ft) before entering the cyclone vault. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

d. CSSF 7 transport lines are approximately 3.7 m (12.2 ft) below the existing grade, where they branch off the main trunk. The lines travel 69.2 m (227.1 ft) southeast, and the stainless-steel- and concrete-encased lines are approximately 2.4 m (8 ft) below the existing grade. The transport lines then travel up the side of CSSF 7 approximately 18.6 m (61 ft) before entering the cyclone vault. Detailed maps and drawings are provided in EDF-11119 and Appendix C of EDF-11132.

e. The main trunk that connects to CSSFs 4 through 7 is 4.2 to 3.7 m (13.9 ft to 12.2 ft) below the existing grade.

CSSF Calcined Solids Storage Facility
NWCF New Waste Calcining Facility

2.11.1.7.3 Summary of Potential Partially Filled Transport Lines—Based on information provided in EDF-11119, three potential radioactive waste deposits may have accumulated in the transport lines near CSSFs 2, 3 and 4 and may not have been flushed out during closure of the calcining facilities (Figure 2-44). It was assumed in the CSSF PA/CA (DOE-ID 2022a) that one-twenty-fifth (3.9%) of the transport line volume was filled with residual waste. The transport line contamination is based on the potential estimated length of 23.8 m (78 ft) of accumulated material averaged over the total length of 613.3 m (2,012 ft) of piping at the CSSF. Though some of the lines may have areas where calcine accumulated, the likelihood of drilling into one of these areas is small in comparison to the overall length of lines at the CSSF (see Subsection 7.2.3). At CSSF 1, a portion of the transport lines (approximately a 6.1- to 9.1-m [20- to 30-ft] section) was removed during HWMA/RCRA closure of WCF because that portion was partially filled with CSSF 2 cold startup material and the remaining transport lines between CSSF 1 and the WCF were grouted in place. Therefore, no waste deposits remain in the CSSF 1 transport lines.

2.11.2 CSSF Waste Origin and Management

Prior to the actual reprocessing of SNF, the fuel cladding was removed (called decladding).[29] Thereafter, SNF was reprocessed, which typically began by dissolving the fuel. This created an aqueous solution containing fission products, activation products, and uranium (Staiger and Swenson 2021). Uranium was chemically separated from the dissolver solution in the first-cycle uranium extraction process (Staiger and Swenson 2021). The raffinate from the first-cycle extraction process contained the bulk of the fission products. Other waste sources included fuel cladding, equipment decontamination, uranium purification (second- and third-cycle raffinates), and support operations such as ion-exchange systems, laboratory analyses of radioactive materials, and off-gas treatment systems (Staiger and Swenson 2021). Reprocessing SNF and other activities (described above) at INTEC generated millions of gallons of liquid radioactive waste.

Differences in the fuel configuration, especially the fuel-cladding material, dictated the use of different chemicals to process the various types of fuel (Staiger and Swenson 2021). Processing varying types of fuels generated chemically different liquid wastes and, consequently, chemically different calcine. Liquid wastes and calcine were often named for the cladding of the fuel from which they were derived. Names such as “aluminum” and “zirconium” waste were applied to wastes generated by the dissolution of aluminum- and zirconium-clad fuels, respectively (Staiger and Swenson 2021). Aluminum is a major component of “Al calcine” and zirconium is a major component of “Zr calcine” (Staiger and Swenson 2021). Sodium-bearing waste was a term applied to wastes that contained relatively high concentrations (1 to 2 molar) of sodium (Staiger and Swenson 2021). The high sodium concentration came from processes that used alkali metal salts, such as sodium permanganate, sodium hydroxide, and sodium carbonate (Staiger and Swenson 2021). Sodium-bearing wastes included most of the wastes generated by equipment decontamination and support systems (e.g., ion exchangers, off-gas systems, scrubbers, and laboratory analyses) (Staiger and Swenson 2021).

Fluidized-bed calcination was a unique process developed to treat liquid radioactive wastes at INTEC. Calcination converted liquid wastes into a solid form in a high-temperature (400 to 600°C [752 to 1,112°F]) fluidized bed. During calcination, liquid radioactive wastes were atomized with air and sprayed

29. DOE M 435.1-1 Chg 3 defines reprocessing as: “Actions necessary to separate fissile elements (U-235, Pu-239, U-233, and Pu-241) and/or transuranium elements (e.g., Np, Pu, Am, Cm, Bk) from other materials (e.g., fission products, activated metals, cladding) contained in spent nuclear fuel for the purposes of recovering desired materials. Separation processes include aqueous separation processes, e.g., the Redox and the Purex processes, and nonaqueous processes, e.g., pyrometallurgical and pyrochemical processes. Wastes that are produced upstream of these separations processes, from processes such as chemical or mechanical decladding, cladding separations, conditioning, or accountability measuring, are not high-level waste. Such wastes are considered processing wastes and should be managed in accordance with the appropriate Chapters of DOE M 435.1-1, as either transuranic, mixed low-level, or low-level waste. Likewise, wastes that are produced downstream of these separations processes, from such processes as decontamination, rinsing, washing, treating, vitrifying, or solidifying, are also not high-level waste and should be managed accordingly. Upstream and downstream wastes are not high-level waste because they do not result from reprocessing.”

into a heated bed of air-fluidized, granular solids[30] (Staiger and Swenson 2021). The principal calcination reactions were evaporation and thermal decomposition of the solutions to form metallic oxides and fluorides, water vapor, and nitrogen oxides (Staiger and Swenson 2021). Solids dissolved in the liquid wastes built up in layers on the fluidized-bed particles. Gases and some of the smaller solids (less than 0.15 mm [0.006 in.] in diameter) were swept from the vessel with the fluidizing air. The average bed particle size was kept at the desired value (typically 0.3 to 0.7 mm [0.012 to 0.027 in.] in diameter) by controlled attrition of bed particles. This was achieved by varying the volume of air used to atomize the liquid waste as it was sprayed into the calciner (Staiger and Swenson 2021). Parameters such as feed composition also affected the calcine particle size. Most of the solids formed in the calcination process were nonreprocessing, nonradioactive oxides of aluminum and zirconium from the fuel cladding and calcium fluoride formed from the calcium nitrate added to fluoride-bearing wastes during the calcination process to prevent fluoride volatility. Only a small fraction (less than 1 wt%) of the calcine was composed of radioactive elements from reprocessing SNF (Staiger and Swenson 2021).

Two different methods were used to supply heat for the calcination process (Staiger and Swenson 2021). Heat for the first three WCF operating campaigns (1963 through 1969) was supplied by circulating a liquid sodium-potassium eutectic metal alloy (NaK) through a heating coil located within the fluidized bed. Heat for the remaining six WCF campaigns and all four of the NWCF campaigns was supplied by the "in-bed" combustion of oxygen-atomized kerosene (Bendixsen 1970).

The calcination process included an extensive off-gas cleanup system that had a cyclone, wet-scrubbing system, adsorbers, and multiple stages of filtration that removed fine particulates from the calciner off-gas before exhausting the gases to the atmosphere. Details of the calcination process used in the WCF and NWCF are discussed extensively in historical calcining facility safety analysis reports (Lakey and Bower 1963; LMITCO 1999). Figure 2-45 depicts the calcination process (without the off-gas cleanup system) and its interconnection with the CSSF.

Calcine was pneumatically transported from two sources (a high-efficiency cyclone and a calciner) within the calcining facilities to each CSSF (Staiger and Swenson 2021) (see Figure 2-45). Relatively large calcine particles (product) were removed from the fluidized-bed portion of the calciner vessel via lines connecting the calciner with the pneumatic transport system. Very small calcine particles were elutriated from the calciner and separated from the off-gas with a cyclone. The small particles were transferred from the cyclone to the pneumatic transport system, where they joined the large particles from the calciner vessel. The combined stream of small and large particles was pneumatically transported from the calcination facilities to each CSSF. In each CSSF, a cyclone separated the calcined solids from the transport air. The solids fell by gravity from the CSSF cyclone into the calcine storage bins. The pneumatic transport air returned to the calciner, where the transport air joined the calciner off-gas for cleanup prior to discharge to the atmosphere.

The WCF converted 15,486,877 L (4,091,000 gal) of aqueous radioactive waste into 2,189 m^3 (77,300 ft^3) of calcined solids (Staiger and Swenson 2021). The NWCF converted 13,787,226 L (3,642,000 gal) of aqueous waste into 2,209 m^3 (78,000 ft^3) of calcined solids (Staiger and Swenson 2021). The total volume of calcined solids stored in the CSSF is approximately 4,400 m^3 (155,300 ft^3) (Staiger and Swenson 2021).

30. "Granular solids" were nonradioactive startup bed materials that were composed of alumina calcine from WCF testing during the first calcination campaign when CSSF 1 was filled. Midway through the second calcination campaign, when CSSF 2 was being filled, WCF switched to dolomite ($CaMg(CO_3)_2$) as the startup bed material. From June 1979 to March 1980, fluorapatite ($Ca_{10}(PO_4)_6F_2$) was used as startup, but then the calcination process switched back to dolomite for the remaining calcining operations.

The volume of calcine stored at INTEC has been reported in Staiger and Swenson (2021). Table 2-6 is a detailed summary of the volume of liquid waste calcined, the volume of calcine generated, and the liquid/solid volume ratio (also called the volume reduction factor) for each of the 13 calcination campaigns. The volume reduction factor is the volume reduction achieved by converting the liquid waste into a solid form.

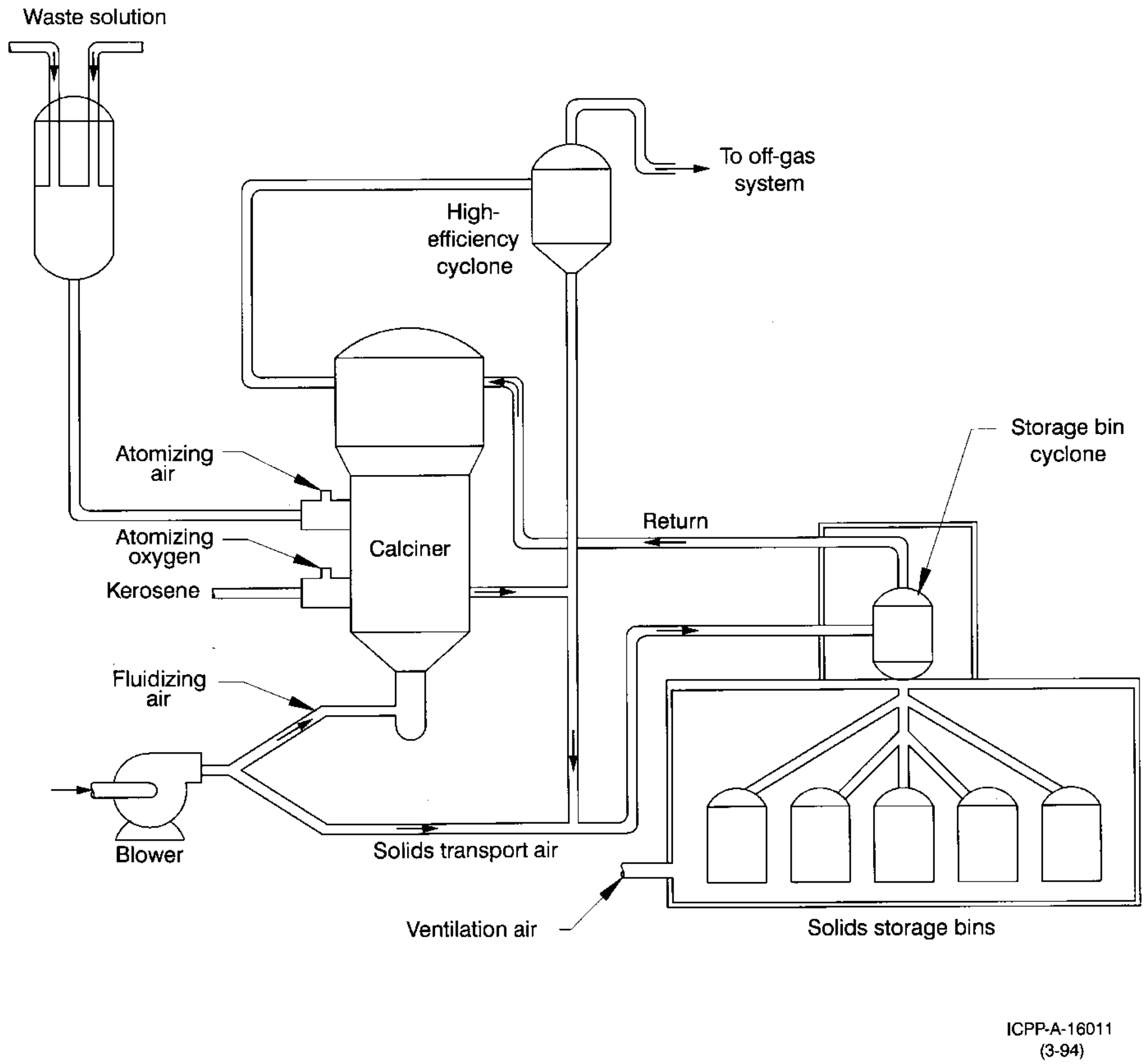

Figure 2-45. Simplified schematic of the calcining process at the Idaho Nuclear Technology and Engineering Center.

Table 2-6. Summary of liquid waste processed and calcine generated during calcination campaigns (from EDF-11126, Table 1).

Operating Campaign	Calciner Operating Date	TFF Waste Calcined gal (L)	Solid Calcine in Storage[a] ft^3 (m^3)	Liquid/Solid Volume Ratio[b]
WCF 1	Nov 1963 to Oct 1964	513,000 (1,941,916)	7,760 (220)	8.8
WCF 2	Mar 1966 to Mar 1968	986,000 (3,732,416)	13,000 (367)	10.1
WCF 3	Aug 1968 to Jun 1969	340,100 (1,287,419)	5,810 (165)	7.8
WCF 4	May 1970 to Jan 1971	229,500 (868,752)	5,120 (145)	6.0
WCF 5	Sept 1971 to May 1972	293,200 (1,109,883)	7,000 (198)	5.6
WCF 6	May 1973 to May 1974	401,900 (1,521,357)	9,060 (256)	5.9
WCF 7	May 1975 to Jan 1977	381,200 (1,442,999)	9,270 (263)	5.5
WCF 8	Sept 1977 to Sept 1978	469,000 (1,775,358)	10,700 (303)	5.9
WCF 9	Jun 1979 to Mar 1981	477,300 (1,806,777)	9,530 (270)	6.7
NWCF 1	Aug 1982 to Jun 1984	1,531,300 (5,796,601)	27,400 (777)	7.5
NWCF 2	Sept 1987 to Dec 1988	829,000 (3,138,106)	16,900 (480)[c]	6.6[c]
NWCF 3	Dec 1990 to Nov 1993	776,700 (2,940,129)	16,000 (452)	6.5
NWCF 4	May 1997 to May 2000	505,200 (1,912,390)	17,600 (500)[d]	3.8
Total[e]		7,733,400 (29,274,103)	155,300 (4,400)	Average 6.7

a. Solids include startup bed and solids from both radioactive and nonradioactive feed material.

b. Ratio of TFF liquid to calcined solids. Solids include startup bed and nonradioactive feed associated with each operating campaign.

c. Includes 330 ft^3 (9.34 m^3) of startup bed and nonradioactive feed material from attempted calciner operation in May/June 1989.

d. Includes 40 ft^3 (1.13 m^3) of nonradioactive calcine from the INTEC pilot plant calciner added to Bin WS6-154 in April 1986.

e. Totals may differ slightly from the sum of the columns because of rounding.

INTEC Idaho Nuclear Technology and Engineering Center
NWCF New Waste Calcining Facility
TFF Tank Farm Facility
WCF Waste Calcining Facility

Table 2-7 provides a dated, historical summary of the capacity of each CSSF and the volume of calcine stored in each. As indicated in Table 2-7, CSSFs 2, 4, and 5 are completely full, CSSFs 1 and 3 are nearly full, and CSSF 6 is approximately half full. The first WCF calcining campaign (known as Campaign 1) ended just prior to filling CSSF 1 (Staiger and Swenson 2021) (see Table 2-6). WCF Campaign 1 stopped short of filling CSSF 1 because CSSF 2 did not yet exist, so no bin set was available to receive additional calcine if CSSF 1 had been filled. Similarly, CSSF 3 is not quite full because WCF Campaign 9 ended just prior to filling the CSSF 3 bins. WCF Campaign 9 ended the operation of the WCF, which was not connected to the next CSSF (i.e., CSSF 4) (Staiger and Swenson 2021). The NWCF was not connected to CSSF 3, so CSSF 3 could not be filled by NWCF. CSSFs 2, 4, and 5 are filled to capacity because they were filled during calciner operating campaigns when the next bin set was available to receive calcine (Staiger and Swenson 2021). When those three bin sets were filled, the next bin set was available for service, so there were no concerns about having enough room to store the calciner bed or dissolve the bed and return waste to the TFF. CSSF 6 is approximately half full because DOE stopped calcining waste in May 2000 and decided to treat the sodium-bearing waste remaining in the TFF with different technology (Staiger and Swenson 2021; DOE-ID 1999b; DOE 2005).

Table 2-7. Summary of filling of the Calcined Solids Storage Facility (from EDF-11126, Table 2).

CSSF	Date When Calcine Storage Began[a]	Date When CSSF Was Filled/Filling Stopped	Total Calcine Volume in Storage ft^3 (m^3)	Percent Full
1	November 1963	October 1964	7,760 (220)	97
2	March 1966	February 1972	30,000 (850)	100
3	February 1972	March 1981	39,500 (1,120)	99
4	August 1982	July 1983	17,200 (486)	100
5	July 1983	January 1992	35,600 (1,010)	100
6	December 1992[b]	May 2000	25,200 (713)	47

a. The initial service date is when the nonradioactive calciner startup began for CSSFs 1, 2, 4, and 6 and when radioactive service began for CSSFs 3 and 5.

b. The initial service date does not include 40 ft^3 (1.13 m^3) of nonradioactive calcine from the INTEC pilot plant calciner added to Bin VES-WS6-154 in April 1986. CSSF 6 was placed in service in December 1992 when the startup bed was added, but feed solution to the calciner and CSSF 6 began in January 1993.

CSSF Calcined Solids Storage Facility
INTEC Idaho Nuclear Technology and Engineering Center

2.11.3 CSSF Radionuclide Inventory

Calcined Waste Storage at the Idaho Nuclear Technology Engineering Center (Staiger and Swenson 2021) provides the volume, mass, and composition (chemical and radioactivity) of calcine stored at INTEC in the CSSF. The Staiger and Swenson (2021) report uses historical liquid waste sample data, the volume of liquid waste calcined, calciner operating data, and CSSF operating data to quantify the calcine inventory and composition in each bin. These data are compiled and the calcine composition is calculated in several large Microsoft Excel databases and spreadsheets that are collectively called the Historical Processing Model (HPM).[31]

INTEC calcine composition data from the HPM has been used to support regulatory compliance, safety analysis reports, calcine retrieval and treatment designs, and compliance with storage and disposal acceptance criteria. Development of the HPM began in the late 1990s as a part of the effort to close the TFF.[32] The calcine in each bin contains multiple layers of chemically and radiologically different calcine, as shown in Figure 2-46. Although not presented in Staiger and Swenson (2021), the HPM calculates the composition of each of the nearly 7,000 batches of material (including waste not generated in the reprocessing of SNF such as nonradioactive startup bed material, radioactive liquid reprocessing waste from the TFF) charged to the calciners.

The HPM, its databases, and calculation techniques have been reviewed for accuracy and completeness. Several of the reviews have been formally documented (Wood et al. 2003; Swenson 2018a), and to validate HPM calculations, comparisons of HPM calcine composition calculations with calcine sample data have been made (Wood et al. 2003; Swenson and Thomas 2006; Swenson 2018b). In general, agreement is excellent between the HPM calcine composition calculations and the validated data from calcine samples. Because the HPM is used extensively as the source for calcine composition, it has undergone in-depth reviews and has been independently verified (Wood et al. 2003; Swenson, Nenni, and

31. To preserve essential calcine composition data calculated using the HPM, the data were transferred to the ICP Environmental Data Warehouse for storage. Additionally, the spreadsheets that make up the HPM have been archived on the ICP Electronic Document Management System as “read-only” supporting information for the Staiger and Swenson (2021) report. Preservation of these data in the Environmental Data Warehouse and Electronic Document Management System allows for static accessibility to the data, removing any concerns related to data corruption and version tracking.

32. In 2006, the Secretary of Energy determined that pursuant NDAA Section 3116(a) of the “Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005,” the stabilized residuals in the TFF and the TFF system are not HLW and may be disposed of in place at the INL Site (DOE 2006).

Young 2018; Swenson 2018b), providing accurate, detailed composition data for calcine. Additional details are discussed in the CSSF PA/CA (DOE-ID 2022a).

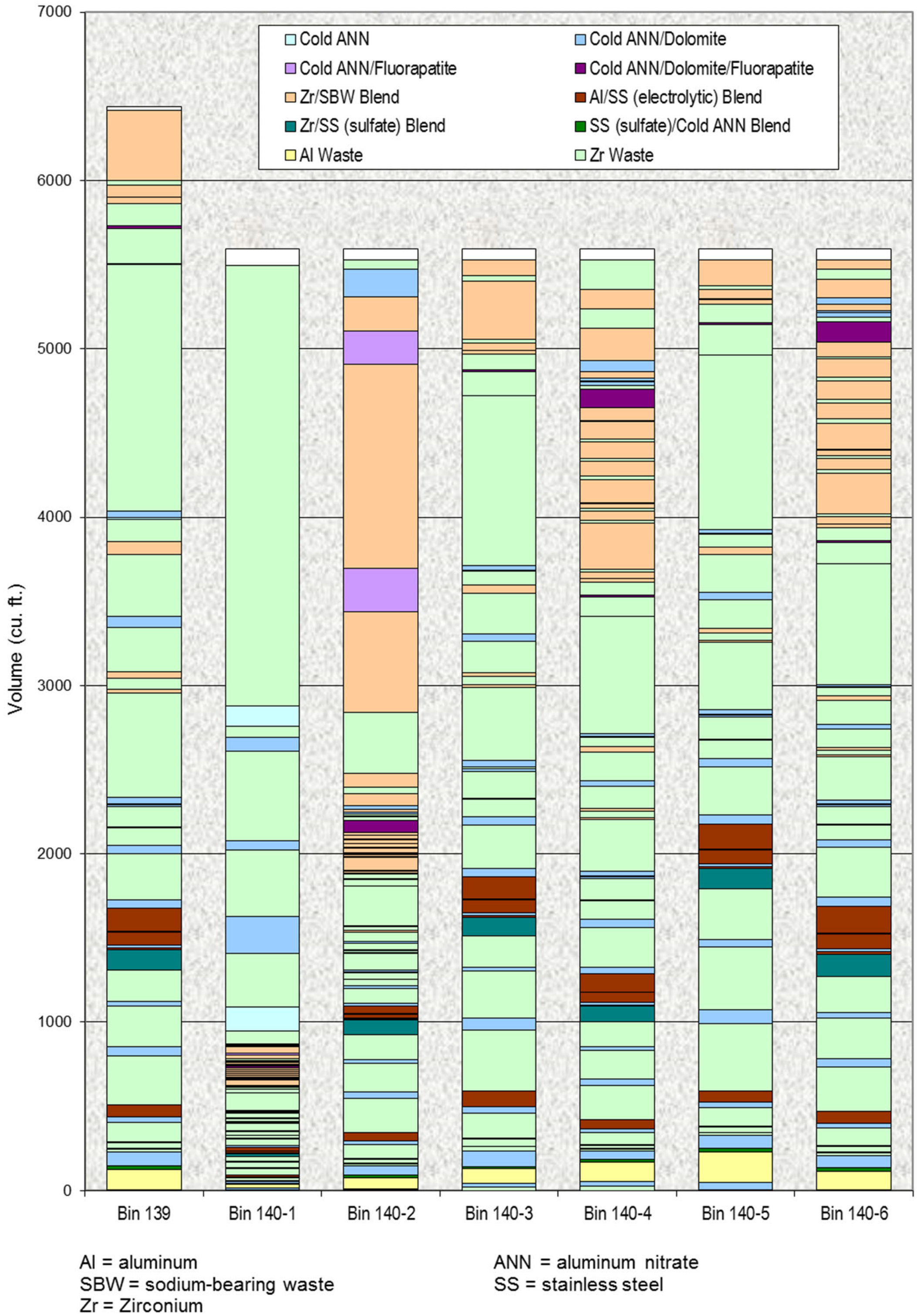

Figure 2-46. Simplified schematic of chemically different calcine layers in Calcined Solids Storage Facility 3.

Calcine in the bins is heterogeneous and contains multiple layers with different chemical and radiological characteristics, as shown in Figure 2-46. Though sampling and analysis of calcine have been conducted in the past, collecting and analyzing samples are difficult (e.g., obtaining a representative sample is difficult, high radiation levels can cause inaccurate results, samples may not sufficiently characterize the calcine inventory, and calcine samples are limited). Thus, the HPM provides an excellent alternative method to determine the CSSF radionuclide inventory in addition to the use of historical sampling and analysis results of solid calcine.

The HPM and supporting documentation were used to calculate the assumed radionuclide inventory remaining in the CSSF at the time of closure.[33] Subsections 2.3 through 2.5 of the CSSF PA/CA (DOE-ID 2022a) present and repeat key sections in Staiger and Swenson (2021) necessary to understand the process used to characterize calcine. In addition, these subsections of the CSSF PA/CA provide details regarding the method used to develop the radionuclide inventory (radionuclide activity and volume) for residual calcine, which was assumed to be 5.1 cm (2 in.) (depth) after calcine removal (see Subsection 2.11.3.3). EDF-11126, "Calcine Radionuclide Inventory Calculations for the CSSF Performance Assessment," which is a supporting document to the CSSF PA/CA, contains a list and description of the files necessary to run the HPM and perform operations, such as decaying radioactivity to various dates or finding the activity of radionuclides or quantities of trace elements not listed in the summary tables of the Staiger and Swenson (2021) report. EDF-11126 and associated files provide a record, maintained in the ICP Electronic Document Management System, that will allow results to be reproduced if necessary.

The following subsections present the radionuclide inventory developed using HPM data to support the CSSF PA/CA dose calculations referenced in this Draft CSSF 3116 Basis Document.

2.11.3.1 Radionuclides Evaluated for the CSSF Performance Assessment

To calculate the calcine radionuclide inventory, the HPM uses calcine composition data based on calciner and CSSF operating data, liquid waste sample analyses, volumes of liquid waste calcined, ORIGEN2- and ORIGEN-ARP-based radioactivity calculations (Croff 1980), and process knowledge of INTEC fuel and waste-processing systems. Of the hundreds of radionuclides that exist, the 148 radionuclides listed in Table 2-8 were evaluated for the CSSF dose assessment. For the purposes of the CSSF dose assessment, the inventory is calculated for radionuclides with radioactive decay half-lives greater than 5 years. Five years is one-twentieth the 100-year institutional control period assumed for the purposes of analysis in the CSSF PA/CA (DOE-ID 2022a),[34] and any radionuclide with less than a 5-year radioactive decay half-life will decay by more than a factor of 1 million during that assumed 100-year institutional control period. Therefore, 73 short-lived radionuclides are eliminated from further analysis for the CSSF dose assessment. The 148 radionuclides and their radionuclide half-lives are shown in Table 2-8. Table 2-8 is reproduced from Table 2-18 in the CSSF PA/CA and modified as necessary for this Draft CSSF 3116 Basis Document.

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33. The Staiger and Swenson (2018) report was used to calculate the radionuclide inventory for the CSSF PA/CA. In 2021, the Staiger and Swenson report was revised to document the process to retire the HPM, which was managed as a software program under ICP Software Quality Assurance requirements, and to archive the HPM data in the ICP Environmental Data Warehouse. HPM data in the 2018 report (Revision 5) and the 2021 report (Revision 6) are the same because HPM data are static (no additions or removals from the bins have occurred). Although supporting documents, such as EDF-11126, "Calcine Radionuclide Inventory Calculations for the CSF Performance Assessment," reference the 2018 report, this document references the more recent 2021 report.

34. In the CSSF PA/CA, the analysis assumed a 100-year institutional control period. Future land use likely will be similar to current uses, with research facilities within INL Site boundaries and agricultural and open land surrounding the INL Site. DOE expects to retain ownership and control of the INL Site until at least 2095 and will continue to manage portions that cannot be released for unrestricted land use beyond 2095 (INL 2016).

Table 2-8. Radionuclides evaluated for the calcine inventory analysis with associated half-lives.

Radionuclide	Half-Life (yr)[a]	Is Half-Life >5 yr?[b]
Ac-225	2.74E-02	—
Ac-227	2.18E+01	Yes
Ac-228	7.02E-04	—
Ag-108	4.51E-06	—
Ag-108m	4.18E+02	Yes
Ag-109m	1.25E-06	—
Ag-110	7.80E-07	—
Ag-110m	6.84E-01	—
Am-241	4.32E+02	Yes
Am-242	1.83E-03	—
Am-242m	1.41E+02	Yes
Am-243	7.37E+03	Yes
At-217	1.02E-09	—
Ba-137m	4.85E-06	—
Be-10	1.51E+06	Yes
Bi-210	1.37E-02	—
Bi-210m	3.04E+06	Yes
Bi-211	4.07E-06	—
Bi-212	1.15E-04	—
Bi-213	8.67E-05	—
Bi-214	3.78E-05	—
C-14	5.73E+03	Yes
Cd-109	1.26E+00	—
Cd-113m	1.41E+01	Yes
Ce-142[c]	Stable	Yes
Ce-144	7.80E-01	—
Cf-249	3.51E+02	Yes
Cf-250	1.31E+01	Yes
Cf-251	8.98E+02	Yes
Cf-252	2.65E+00	—
Cm-242	4.46E-01	—
Cm-243	2.91E+01	Yes
Cm-244	1.81E+01	Yes
Cm-245	8.50E+03	Yes
Cm-246	4.76E+03	Yes
Cm-247	1.56E+07	Yes
Cm-248	3.48E+05	Yes
Co-60	5.27E+00	Yes
Cs-134	2.07E+00	—
Cs-135	2.30E+06	Yes
Cs-137	3.01E+01	Yes
Eu-150	3.69E+01	Yes
Eu-152	1.35E+01	Yes
Eu-154	8.59E+00	Yes
Eu-155	4.76E+00	—
Fe-55	2.74E+00	—
Fr-221	9.32E-06	—
Fr-223	4.18E-05	—
Gd-152	1.08E+14	Yes
Gd-153	6.58E-01	—
H-3	1.23E+01	Yes
Ho-166m	1.20E+03	Yes
I-129	1.57E+07	Yes
In-115	4.41E+14	Yes
Kr-81[d]	2.29E+05	Yes
Kr-85[d]	1.08E+01	Yes
La-138	1.05E+11	Yes
Nb-93m	1.61E+01	Yes
Nb-94	2.03E+04	Yes
Nd-144	2.29E+15	Yes
Ni-59	7.60E+04	Yes
Ni-63	1.00E+02	Yes
Np-235	1.08E+00	—
Np-236	1.54E+05	Yes
Np-237	2.14E+06	Yes
Np-238	5.80E-03	—
Np-239	6.45E-03	—
Np-240m	1.37E-05	—
Pa-231	3.28E+04	Yes
Pa-233	7.38E-02	—
Pa-234	7.64E-04	—
Pa-234m	2.22E-06	—
Pb-209	3.71E-04	—
Pb-210	2.23E+01	Yes
Pb-211	6.86E-05	—
Pb-212	1.21E-03	—
Pb-214	5.10E-05	—
Pd-107	6.50E+06	Yes
Pm-146	5.53E+00	Yes
Pm-147	2.62E+00	—
Po-210	3.79E-01	—
Po-211	1.64E-08	—
Po-212	9.47E-15	—
Po-213	1.16E-13	—

Table 2-8. (continued).

Radionuclide	Half-Life (yr)[a]	Is Half-Life >5 yr?[b]
Po-214	5.21E-12	—
Po-215	5.64E-14	—
Po-216	4.60E-09	—
Po-218	5.89E-06	—
Pr-144	3.29E-05	—
Pr-144m	1.37E-05	—
Pu-236	2.86E+00	—
Pu-238	8.77E+01	Yes
Pu-239	2.41E+04	Yes
Pu-240	6.56E+03	Yes
Pu-241	1.43E+01	Yes
Pu-242	3.73E+05	Yes
Pu-243	5.65E-04	—
Pu-244	8.00E+07	Yes
Ra-223	3.13E-02	—
Ra-224	1.00E-02	—
Ra-225	4.08E-02	—
Ra-226	1.60E+03	Yes
Ra-228	5.75E+00	Yes
Rb-87	4.75E+10	Yes
Rh-102	5.67E-01	—
Rh-106	9.44E-07	—
Rn-219	1.25E-07	—
Rn-220	1.76E-06	—
Rn-222	1.05E-02	—
Ru-106	1.02E+00	—
Sb-125	2.76E+00	—
Sb-126	3.41E-02	—
Sb-126m	3.64E-05	—
Se-79	1.10E+06	Yes
Sm-146	1.03E+08	Yes
Sm-147	1.06E+11	Yes
Sm-148	7.00E+15	Yes
Sm-149[c]	Stable	Yes
Sm-151	9.00E+01	Yes
Sn-119m	8.02E-01	—
Sn-121m	5.50E+01	Yes
Sn-126	1.00E+05	Yes
Sr-90	2.88E+01	Yes
Tc-98	4.20E+06	Yes
Tc-99	2.11E+05	Yes
Te-123	6.00E+14	Yes
Te-125m	1.57E-01	—
Th-227	5.13E-02	—
Th-228	1.91E+00	—
Th-229	7.34E+03	Yes
Th-230	7.54E+04	Yes
Th-231	2.91E-03	—
Th-232	1.41E+10	Yes
Th-234	6.60E-02	—
Tl-207	9.07E-06	—
Tl-208	5.80E-06	—
Tl-209	4.11E-06	—
Tm-171	1.92E+00	—
U-232	6.89E+01	Yes
U-233	1.59E+05	Yes
U-234	2.46E+05	Yes
U-235	7.04E+08	Yes
U-236	2.34E+07	Yes
U-237	1.85E-02	—
U-238	4.47E+09	Yes
U-240	1.61E-03	—
Y-90	7.30E-03	—
Zr-93	1.53E+06	Yes

a. Radionuclide half-lives were obtained from the ICP Integrated Waste Tracking System database; its data source is the National Nuclear Data Center, Brookhaven Laboratory.

b. Dashes in this column indicate the half-life is less than 5 years and the radionuclide was not retained for further analysis. See the CSSF PA/CA (DOE-ID 2022a) for additional discussion.

c. A stable isotope.

d. Radionuclide is a gas.

CA composite analysis
CSSF Calcined Solids Storage Facility
ICP Idaho Cleanup Project
PA performance assessment

2.11.3.2 CSSF Total Radionuclide Inventory

Seventy-five of the 148 radionuclides in the original inventory had half-lives greater than 5 years and were retained for further analysis in the CSSF radionuclide inventory analysis, as shown in Table 2-9.

The total inventory of the 75 radionuclides in each CSSF (decayed to Year 2016) is shown in Table 2-9. Inventories were calculated using ORIGN2-based models, the mass in each bin, the fuel-type-specific ratios to the Cs-137 concentrations, and the fuel-type-specific concentration of Cs-137. The noble gases (krypton) and stable isotopes (Ce-142 and Sm-149) are retained in the table, but the inventory is either listed as NA (not available) for noble gases or zero for the stable isotopes. Subsections 2.4 and 2.5 of the CSSF PA/CA (DOE-ID 2022a) provide details on the approach used to calculate the CSSF radionuclide inventories.

Table 2-9 also shows the total radionuclide activity in each CSSF decayed to January 1, 2016. The HPM was originally developed and its companion report written in the late 1990s. At that time, 2016 was identified as the earliest date when a future project might retrieve calcine from the CSSF, so the radionuclide data were decayed to that date. Subsequent revisions of the HPM and its report have maintained the 2016 decay date, which currently overestimates the radioactivity.[35] Thus, the 2016 decay date provides upper-bound estimates of the activity values for calculations.

Table 2-9. Calcined Solids Storage Facility total inventory (pre-retrieval operations) decayed until Year 2016 for radionuclides with decay half-lives greater than 5 years (from EDF-11126, Table 4).

Nuclide	CSSF 1 Inventory (Ci)	CSSF 2 Inventory (Ci)	CSSF 3 Inventory (Ci)	CSSF 4 Inventory (Ci)	CSSF 5 Inventory (Ci)	CSSF 6 Inventory (Ci)	CSSF Total Inventory (Ci)
Ac-227	1.09E-02	1.49E-02	1.50E-03	8.75E-04	1.88E-03	5.05E-04	3.06E-02
Ag-108m	6.71E-06	2.32E-05	4.25E-05	2.48E-05	5.33E-05	1.43E-05	1.65E-04
Am-241	1.22E+02	1.13E+03	2.42E+03	1.54E+03	2.88E+03	4.01E+02	8.49E+03
Am-242m	1.75E-02	5.30E-01	1.50E+00	8.75E-01	1.88E+00	5.05E-01	5.31E+00
Am-243	8.65E-03	8.39E-02	3.08E-01	1.96E-01	3.44E-01	1.12E-01	1.05E+00
Be-10	1.70E-04	3.21E-04	2.88E-04	1.68E-04	3.61E-04	9.71E-05	1.41E-03
Bi-210m	4.21E-20	1.14E-17	3.35E-17	1.95E-17	4.20E-17	1.13E-17	1.18E-16
C-14	6.82E-04	8.97E-04	7.77E-07	4.54E-07	9.74E-07	2.62E-07	1.58E-03
Cd-113m	4.86E+01	1.22E+02	1.73E+02	1.01E+02	2.16E+02	5.82E+01	7.19E+02
Ce-142[a]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cf-249	2.25E-14	5.34E-10	1.58E-09	9.23E-10	1.98E-09	5.33E-10	5.55E-09
Cf-250	1.91E-15	2.50E-10	7.38E-10	4.31E-10	9.26E-10	2.49E-10	2.59E-09
Cf-251	5.99E-17	9.52E-12	2.82E-11	1.64E-11	3.53E-11	9.49E-12	9.89E-11
Cm-243	8.34E-04	6.42E-02	1.87E-01	1.09E-01	2.34E-01	6.29E-02	6.58E-01
Cm-244	1.28E-02	8.88E-01	2.41E+00	1.49E+00	2.79E+00	8.44E-01	8.44E+00
Cm-245	4.41E-06	3.74E-04	1.09E-03	6.36E-04	1.36E-03	3.67E-04	3.83E-03

35. In Staiger and Swenson (2021), the date 2016 was selected as the earliest date when a future project might retrieve calcine from the CSSFs. The year 2016 is conservative for the calcine inventory because the CSSFs will be closed after 2016 and not all CSSFs will be closed at the same time. Thus, the 2016 date provides worst-case activity values for the CSSF PA/CA calculations due to reduced radioactive decay. In addition, minimal ingrowth of progeny between the 2016 inventory date and future closure dates would not significantly impact the CSSF PA/CA dose results.

Table 2-9. (continued).

Nuclide	CSSF 1 Inventory (Ci)	CSSF 2 Inventory (Ci)	CSSF 3 Inventory (Ci)	CSSF 4 Inventory (Ci)	CSSF 5 Inventory (Ci)	CSSF 6 Inventory (Ci)	CSSF Total Inventory (Ci)
Cm-246	1.01E-07	3.97E-05	1.17E-04	6.83E-05	1.47E-04	3.94E-05	4.11E-04
Cm-247	3.61E-14	6.10E-11	1.80E-10	1.05E-10	2.26E-10	6.08E-11	6.33E-10
Cm-248	1.15E-14	8.55E-11	2.53E-10	1.48E-10	3.17E-10	8.53E-11	8.89E-10
Co-60	3.82E-01	2.24E+01	3.61E+01	3.48E+01	7.10E+02	1.09E+02	9.13E+02
Cs-135	1.07E+01	2.61E+01	3.51E+01	2.12E+01	4.26E+01	1.07E+01	1.46E+02
Cs-137	8.09E+05	1.69E+06	1.84E+06	1.07E+06	2.31E+06	6.19E+05	8.33E+06
Eu-150	7.56E-05	4.04E-04	9.00E-04	5.26E-04	1.13E-03	3.03E-04	3.34E-03
Eu-152	6.92E+00	3.96E+01	6.61E+01	3.97E+01	8.07E+01	2.14E+01	2.54E+02
Eu-154	4.31E+02	2.39E+03	2.14E+03	1.99E+03	6.62E+03	1.42E+03	1.50E+04
Gd-152	1.91E-11	9.36E-11	2.03E-10	1.18E-10	2.54E-10	6.83E-11	7.56E-10
H-3	1.05E+03	1.51E+03	3.82E+02	2.23E+02	4.80E+02	1.29E+02	3.77E+03
Ho-166m	4.11E-04	2.35E-03	5.35E-03	3.12E-03	6.71E-03	1.80E-03	1.97E-02
I-129	6.88E-03	1.25E-02	1.22E-02	7.08E-03	1.54E-02	4.11E-03	5.82E-02
In-115	9.07E-10	1.37E-09	5.23E-10	3.05E-10	6.56E-10	1.76E-10	3.94E-09
Kr-81[b]	NA	NA	NA	NA	NA	NA	NA
Kr-85[b]	NA	NA	NA	NA	NA	NA	NA
La-138	1.59E-08	2.85E-08	2.23E-08	1.30E-08	2.80E-08	7.52E-09	1.15E-07
Nb-93m	1.14E+02	2.95E+02	4.30E+02	2.51E+02	5.39E+02	1.45E+02	1.77E+03
Nb-94	1.33E-03	1.64E+01	4.86E+01	2.84E+01	6.10E+01	1.64E+01	1.71E+02
Nd-144	8.19E-08	1.61E-07	1.56E-07	9.13E-08	1.96E-07	5.27E-08	7.39E-07
Ni-59	0.00E+00	2.04E+01	6.04E+01	3.53E+01	7.57E+01	2.03E+01	2.12E+02
Ni-63	0.00E+00	1.09E+03	2.78E+03	1.81E+03	3.19E+03	5.85E+02	9.46E+03
Np-236	8.93E-06	3.01E-04	8.56E-04	5.00E-04	1.07E-03	2.88E-04	3.03E-03
Np-237	1.09E+00	1.76E+00	8.00E+00	1.95E+01	3.75E+01	5.50E+00	7.34E+01
Pa-231	1.46E-02	2.00E-02	2.23E-03	1.30E-03	2.79E-03	7.51E-04	4.17E-02
Pb-210	2.02E-03	2.66E-03	1.32E-05	7.72E-06	1.66E-05	4.46E-06	4.71E-03
Pd-107	9.16E-01	1.69E+00	1.44E+00	8.38E-01	1.80E+00	4.84E-01	7.16E+00
Pm-146	2.98E-02	3.26E-01	8.49E-01	4.96E-01	1.07E+00	2.86E-01	3.05E+00
Pu-238	3.16E+02	8.11E+03	1.66E+04	1.65E+04	3.23E+04	4.99E+03	7.89E+04
Pu-239	4.27E+01	1.82E+02	4.45E+02	4.61E+02	8.96E+02	3.34E+02	2.36E+03
Pu-240	1.71E+01	1.44E+02	3.21E+02	2.97E+02	6.32E+02	1.80E+02	1.59E+03
Pu-241	1.19E+02	4.06E+03	8.54E+03	7.93E+03	1.71E+04	4.90E+03	4.27E+04
Pu-242	9.86E-03	3.36E-01	8.18E-01	8.30E-01	1.48E+00	3.76E-01	3.85E+00
Pu-244	1.92E-10	5.28E-10	8.14E-10	4.75E-10	1.02E-09	2.74E-10	3.30E-09
Ra-226	4.38E-03	5.77E-03	5.25E-05	3.07E-05	6.59E-05	1.77E-05	1.03E-02
Ra-228	4.15E-08	6.59E-08	3.36E-08	1.96E-08	4.21E-08	1.13E-08	2.14E-07
Rb-87	1.62E-03	3.03E-03	2.64E-03	1.54E-03	3.31E-03	8.90E-04	1.30E-02
Se-79	2.72E+00	5.02E+00	4.99E+00	2.91E+00	6.28E+00	1.67E+00	2.36E+01

Table 2-9. (continued).

Nuclide	CSSF 1 Inventory (Ci)	CSSF 2 Inventory (Ci)	CSSF 3 Inventory (Ci)	CSSF 4 Inventory (Ci)	CSSF 5 Inventory (Ci)	CSSF 6 Inventory (Ci)	CSSF Total Inventory (Ci)
Sm-146	9.95E-07	6.53E-06	1.54E-05	9.02E-06	1.94E-05	5.21E-06	5.66E-05
Sm-147	5.80E-04	9.68E-04	6.08E-04	3.55E-04	7.62E-04	2.05E-04	3.48E-03
Sm-148	6.14E-10	2.40E-09	4.70E-09	2.75E-09	5.90E-09	1.59E-09	1.79E-08
Sm-149[a]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sm-151	1.64E+04	2.43E+04	1.35E+04	6.85E+03	1.90E+04	5.94E+03	8.60E+04
Sn-121m	2.26E+00	1.15E+02	3.31E+02	1.93E+02	4.15E+02	1.12E+02	1.17E+03
Sn-126	1.10E+01	2.02E+01	2.01E+01	1.17E+01	2.53E+01	6.76E+00	9.51E+01
Sr-90	6.72E+05	1.49E+06	1.57E+06	9.94E+05	2.13E+06	5.35E+05	7.40E+06
Tc-98	3.90E-05	1.40E-04	2.62E-04	1.53E-04	3.28E-04	8.83E-05	1.01E-03
Tc-99	4.25E+02	7.68E+02	7.39E+02	4.28E+02	9.34E+02	2.49E+02	3.54E+03
Te-123	1.20E-12	8.25E-11	2.39E-10	1.40E-10	3.00E-10	8.07E-11	8.44E-10
Th-229	9.50E-06	1.88E-05	1.87E-05	1.09E-05	2.35E-05	6.31E-06	8.78E-05
Th-230	1.01E-01	1.22E-01	6.13E-03	1.17E-03	7.22E-02	2.89E-02	3.32E-01
Th-232	4.18E-08	6.65E-08	3.43E-08	2.00E-08	4.30E-08	1.16E-08	2.17E-07
U-232	8.02E-05	8.83E-03	9.77E-02	6.93E-02	9.71E-02	1.35E-02	2.86E-01
U-233	1.57E-04	2.37E-04	1.26E-03	3.13E-03	5.36E-03	7.29E-04	1.09E-02
U-234	2.96E+00	6.69E+00	1.97E+00	1.81E+00	6.67E+00	2.96E+00	2.31E+01
U-235	2.06E-02	3.99E-02	1.94E-02	1.61E-02	8.63E-02	7.29E-02	2.55E-01
U-236	4.78E-02	1.01E-01	5.16E-02	4.38E-02	2.67E-01	1.63E-01	6.74E-01
U-238	1.17E-03	2.27E-03	3.43E-03	8.68E-03	4.72E-02	5.74E-02	1.20E-01
Zr-93	1.28E+02	3.40E+02	5.10E+02	2.98E+02	6.39E+02	1.72E+02	2.09E+03

a. Stable isotope retained in table, but the inventory is listed as zero in Table 4 of EDF-11126.
b. Noble gas retained in table, but the inventory is listed as not available in Table 4 of EDF-11126.
CSSF Calcined Solids Storage Facility
NA not available

2.11.3.3 CSSF Residual Radionuclide Inventory at Closure

In order to support the dose assessment calculations, it was assumed that 5.1 cm (2 in.) (depth) of residual calcine will remain in the bins after calcine removal. The 5.1-cm (2-in.) depth of residual calcine was chosen based on three historical, full-scale calcine retrieval tests performed in 1981, 1995, and 2005 (ENICO 1981; Westra 1982; Griffith 1996; AEA Technology 2006), respectively. Each test used a different retrieval method, but results for each test were similar, with residual calcine depths ranging from less than 2.5 cm (1 in.) to a depth of 5.1 cm (2 in.). Based on these historical tests, a 5.1-cm (2-in.) residual after waste-retrieval operations was assumed to be achievable and an appropriate assumption for the CSSF PA/CA calculations. Calcine remaining in the bins after waste-retrieval operations represents the residual calcine inventory assumed for the dose assessment calculations of the CSSF PA/CA (DOE-ID 2022a). In the CSSF PA/CA, residual waste in the bins, including waste on stiffening rings or in the distributor pipe, for example, was assumed to be located on the bottom of the bins. In addition, the CSSF storage vaults have never contained calcine waste and the equipment and structures above the storage vaults will be removed, as appropriate, and disposed of as identified in future State-approved closure plans. The inventory assumed in the analysis for the transport lines, analyzed separately from the bins, is provided in Subsection 7.3.

Each CSSF has a different number of bins, has different sizes of bins, and is filled to different levels. Therefore, the residual percent of calcine remaining after closure is different for each CSSF. For the calculations, all the bins are represented by cylinders or annular cylinders. Table 2-10 shows:

- CSSF parameters, including the number of bins, bin heights, bin outside diameters, annular air space diameter (if necessary), and volume of calcine currently in the bins
- Calculated volume of calcine in the 5.1-cm (2-in.) residual
- Percent of total calcine remaining after waste retrieval operations.

Estimates for the residual activity in the CSSF decayed to January 1, 2016, are shown in Table 2-11. These inventories were used in the CSSF PA/CA base case dose assessment calculations.

Tables 2-10 and 2-11 show that approximately 99% of the total radionuclide inventory (volume and curies), including HRRs, will be removed from each CSSF. To develop a residual radionuclide inventory for the CSSF after waste retrieval activities are complete, the volume of calcine is assumed to be 5.1 cm (2 in.) (depth) based on historical tests as described above. For the removal of calcine, the retrieval demonstration has confirmed the efficiency of the pneumatic transfer system. See Subsections 2.11.4.2 and 5.2 for additional retrieval technology information. The residual inventory is assumed to be a linear function of the remaining height of the calcine, so alternative inventories can be easily calculated based on the Table 2-11 inventory, which represents the 5.1-cm (2-in.) calcine residual.

Table 2-10. Summary of Calcined Solids Storage Facility bin set parameters, volume of calcine, and residual calcine based on a 5.1-cm (2 in.) depth (from EDF-11126, Table 15).

CSSF	Bin Parameters in Each CSSF				Volume of Calcine		Residual Percent Remaining[b]
	Number of Bins	Height m (ft)	Diameter				
			Bin m (ft)	Annular Opening in Center of Bin m (ft)	Total m^3 (ft^3)	Post-Waste Removal 5.1-cm (2-in.) Residual m^3 (ft^3)[a]	
1	4[c]	6.1/6.1/7.6–8.5 (20/20/25–28)[c]	3.7 (12)	NA	220 (7,760)	2.1 (75.4)	0.97%
2	7	12.9 (42.3)	3.7 (12)	NA	850 (30,000)	3.7 (131.9)	0.44%
3	7	16.2/18.6 (53/61)[d]	3.7 (12)	NA	1,120 (39,500)	3.7 (131.9)	0.33%
4	3	16.8 (55)	3.7 (12)	NA	486 (17,200)	1.6 (56.5)	0.33%
5	7	16.8 (55)	3.7 (12)	1.2 (4)	1,010 (35,600)	3.3 (117.3)	0.33%
6	7	20.6 (67.5)	4.1 (13.5)	1.5 (5)	713 (25,200	4.1 (144.1)	0.57%

a. For CSSFs 1 through 4, residual waste volume (ft^3) was calculated as follows: $\left(\pi * \left(\frac{bin\ diameter\ (ft)}{2}\right)^2 * \left(\frac{2\ in.residual\ waste}{12\ in./ft}\right) * \#\ bins\right)$.

For CSSFs 5 and 6, residual waste volume (ft^3) was calculated as follows:

$$\left(\pi * \left[\left(\frac{bin\ diameter\ (ft)}{2}\right)^2 - \left(\frac{annular\ opening\ (ft)}{2}\right)^2\right] * \left(\frac{2\ in.residual\ waste}{12\ in./ft}\right) * \#\ bins\right).$$

b. The residual percent remaining was calculated as follows: $\left(\frac{Total\ Volume\ of\ Calcine\ (ft^3)}{Residual\ Volume\ of\ Calcine\ (ft^3)} * 100\right)$.

c. Each composite bin consists of three concentric sub-bins, numbered from inside to outside, A, B, and C (see Figure 2-29). The innermost sub-bin (A) in each group is cylindrical. Each cylindrical sub-bin is surrounded by an annular sub-bin (B), which is, in turn, surrounded by a second annular sub-bin (C). Small gaps between the sub-bins provide a path for airflow, which removes radiolytic decay heat from the calcine. One bin, VES-WCS-115-4A, is approximately 8.5 m (28 ft) tall.

d. In CSSF 3, six bins are 16.1 m (53 ft) tall and one is 18.6 m (61 ft) tall.

CSSF Calcined Solids Storage Facility
NA not applicable

Table 2-11. Calcined Solids Storage Facility residual inventory (post-retrieval operations) decayed to January 1, 2016, based on the assumption that 5.1 cm (2 in.) of calcine will be left in the bins after waste has been removed (from EDF-11126, Table 16).

Radionuclide	5.1-cm (2-in.) CSSF Residual Inventories[a]						
	CSSF 1 (Ci)	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)	Radionuclide Total (Ci)[b]
Ac-227	1.06E-04	6.55E-05	5.01E-06	2.88E-06	6.19E-06	2.89E-06	1.89E-04
Ag-108m	6.52E-08	1.02E-07	1.42E-07	8.15E-08	1.75E-07	8.18E-08	6.48E-07
Am-241	1.19E+00	4.99E+00	8.07E+00	5.06E+00	9.48E+00	2.29E+00	3.11E+01
Am-242m	1.70E-04	2.33E-03	5.01E-03	2.88E-03	6.19E-03	2.89E-03	1.95E-02
Am-243	8.40E-05	3.69E-04	1.03E-03	6.44E-04	1.13E-03	6.41E-04	3.90E-03
Be-10	1.65E-06	1.41E-06	9.63E-07	5.53E-07	1.19E-06	5.55E-07	6.33E-06
Bi-210m	4.09E-22	5.00E-20	1.12E-19	6.43E-20	1.38E-19	6.45E-20	4.29E-19
C-14	6.63E-06	3.95E-06	2.60E-09	1.49E-09	3.21E-09	1.50E-09	1.06E-05
Cd-113m	4.72E-01	5.38E-01	5.77E-01	3.31E-01	7.13E-01	3.33E-01	2.96E+00
Ce-142[c]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cf-249	2.18E-16	2.35E-12	5.28E-12	3.03E-12	6.53E-12	3.05E-12	2.02E-11
Cf-250	1.86E-17	1.10E-12	2.47E-12	1.42E-12	3.05E-12	1.42E-12	9.45E-12
Cf-251	5.82E-19	4.19E-14	9.41E-14	5.41E-14	1.16E-13	5.43E-14	3.61E-13
Cm-243	8.11E-06	2.82E-04	6.23E-04	3.58E-04	7.71E-04	3.60E-04	2.40E-03
Cm-244	1.25E-04	3.91E-03	8.04E-03	4.91E-03	9.20E-03	4.83E-03	3.10E-02
Cm-245	4.29E-08	1.64E-06	3.64E-06	2.09E-06	4.50E-06	2.10E-06	1.40E-05
Cm-246	9.78E-10	1.74E-07	3.91E-07	2.25E-07	4.83E-07	2.25E-07	1.50E-06
Cm-247	3.51E-16	2.68E-13	6.02E-13	3.46E-13	7.45E-13	3.47E-13	2.31E-12
Cm-248	1.12E-16	3.76E-13	8.45E-13	4.86E-13	1.05E-12	4.87E-13	3.24E-12
Co-60	3.72E-03	9.87E-02	1.20E-01	1.14E-01	2.34E+00	6.25E-01	3.30E+00
Cs-135	1.04E-01	1.15E-01	1.17E-01	6.96E-02	1.40E-01	6.12E-02	6.08E-01
Cs-137	7.86E+03	7.41E+03	6.14E+03	3.53E+03	7.59E+03	3.54E+03	3.61E+04
Eu-150	7.35E-07	1.78E-06	3.01E-06	1.73E-06	3.72E-06	1.73E-06	1.27E-05
Eu-152	6.72E-02	1.74E-01	2.21E-01	1.30E-01	2.66E-01	1.23E-01	9.81E-01
Eu-154	4.19E+00	1.05E+01	7.14E+00	6.54E+00	2.18E+01	8.12E+00	5.83E+01
Gd-152	1.86E-13	4.12E-13	6.77E-13	3.89E-13	8.37E-13	3.91E-13	2.89E-12
H-3	1.02E+01	6.62E+00	1.28E+00	7.34E-01	1.58E+00	7.37E-01	2.11E+01
Ho-166m	3.99E-06	1.03E-05	1.79E-05	1.03E-05	2.21E-05	1.03E-05	7.49E-05
I-129	6.68E-05	5.51E-05	4.08E-05	2.33E-05	5.07E-05	2.35E-05	2.60E-04
In-115	8.81E-12	6.02E-12	1.75E-12	1.00E-12	2.16E-12	1.01E-12	2.08E-11
Kr-81[d]	NA	NA	NA	NA	NA	NA	NA
Kr-85[d]	NA	NA	NA	NA	NA	NA	NA
La-138	1.55E-10	1.25E-10	7.45E-11	4.28E-11	9.21E-11	4.30E-11	5.33E-10
Nb-93m	1.11E+00	1.30E+00	1.44E+00	8.25E-01	1.78E+00	8.28E-01	7.27E+00

Table 2-11. (continued).

Radionuclide	5.1-cm (2-in.) CSSF Residual Inventories[a]						
	CSSF 1 (Ci)	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)	Radionuclide Total (Ci)[b]
Nb-94	1.29E-05	7.23E-02	1.62E-01	9.33E-02	2.01E-01	9.37E-02	6.22E-01
Nd-144	7.96E-10	7.06E-10	5.22E-10	3.00E-10	6.46E-10	3.01E-10	3.27E-09
Ni-59	0.00E+00	8.98E-02	2.02E-01	1.16E-01	2.49E-01	1.16E-01	7.73E-01
Ni-63	0.00E+00	4.80E+00	9.29E+00	5.96E+00	1.05E+01	3.34E+00	3.39E+01
Np-236	8.68E-08	1.32E-06	2.86E-06	1.64E-06	3.54E-06	1.65E-06	1.11E-05
Np-237	1.06E-02	7.75E-03	2.67E-02	6.43E-02	1.24E-01	3.14E-02	2.64E-01
Pa-231	1.42E-04	8.79E-05	7.44E-06	4.28E-06	9.20E-06	4.29E-06	2.55E-04
Pb-210	1.96E-05	1.17E-05	4.42E-08	2.54E-08	5.46E-08	2.55E-08	3.14E-05
Pd-107	8.90E-03	7.43E-03	4.80E-03	2.76E-03	5.93E-03	2.77E-03	3.26E-02
Pm-146	2.89E-04	1.43E-03	2.84E-03	1.63E-03	3.51E-03	1.64E-03	1.13E-02
Pu-238	3.07E+00	3.57E+01	5.56E+01	5.43E+01	1.06E+02	2.85E+01	2.84E+02
Pu-239	4.15E-01	8.01E-01	1.49E+00	1.52E+00	2.95E+00	1.91E+00	9.08E+00
Pu-240	1.66E-01	6.33E-01	1.07E+00	9.76E-01	2.08E+00	1.03E+00	5.96E+00
Pu-241	1.15E+00	1.79E+01	2.85E+01	2.61E+01	5.64E+01	2.80E+01	1.58E+02
Pu-242	9.58E-05	1.48E-03	2.73E-03	2.73E-03	4.88E-03	2.15E-03	1.41E-02
Pu-244	1.87E-12	2.32E-12	2.72E-12	1.56E-12	3.36E-12	1.57E-12	1.34E-11
Ra-226	4.25E-05	2.54E-05	1.75E-07	1.01E-07	2.17E-07	1.01E-07	6.85E-05
Ra-228	4.03E-10	2.90E-10	1.12E-10	6.45E-11	1.39E-10	6.47E-11	1.07E-09
Rb-87	1.58E-05	1.33E-05	8.82E-06	5.07E-06	1.09E-05	5.09E-06	5.90E-05
Se-79	2.65E-02	2.21E-02	1.67E-02	9.56E-03	2.07E-02	9.53E-03	1.05E-01
Sm-146	9.67E-09	2.87E-08	5.16E-08	2.97E-08	6.38E-08	2.98E-08	2.13E-07
Sm-147	5.64E-06	4.26E-06	2.03E-06	1.17E-06	2.51E-06	1.17E-06	1.68E-05
Sm-148	5.97E-12	1.05E-11	1.57E-11	9.03E-12	1.94E-11	9.06E-12	6.98E-11
Sm-149[c]	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sm-151	1.60E+02	1.07E+02	4.51E+01	2.25E+01	6.27E+01	3.40E+01	4.31E+02
Sn-121m	2.20E-02	5.06E-01	1.11E+00	6.36E-01	1.37E+00	6.38E-01	4.28E+00
Sn-126	1.07E-01	8.90E-02	6.72E-02	3.85E-02	8.34E-02	3.87E-02	4.24E-01
Sr-90	6.53E+03	6.53E+03	5.26E+03	3.27E+03	7.03E+03	3.06E+03	3.17E+04
Tc-98	3.79E-07	6.15E-07	8.75E-07	5.03E-07	1.08E-06	5.05E-07	3.96E-06
Tc-99	4.13E+00	3.38E+00	2.47E+00	1.41E+00	3.08E+00	1.43E+00	1.59E+01
Te-123	1.16E-14	3.63E-13	8.00E-13	4.60E-13	9.89E-13	4.61E-13	3.08E-12
Th-229	9.23E-08	8.28E-08	6.26E-08	3.60E-08	7.74E-08	3.61E-08	3.87E-07
Th-230	9.82E-04	5.39E-04	2.05E-05	3.86E-06	2.38E-04	1.65E-04	1.95E-03
Th-232	4.06E-10	2.92E-10	1.15E-10	6.58E-11	1.42E-10	6.61E-11	1.09E-09
U-232	7.79E-07	3.88E-05	3.26E-04	2.28E-04	3.20E-04	7.70E-05	9.91E-04
U-233	1.53E-06	1.04E-06	4.22E-06	1.03E-05	1.77E-05	4.17E-06	3.89E-05

Table 2-11. (continued).

Radionuclide	5.1-cm (2-in.) CSSF Residual Inventories[a]						
	CSSF 1 (Ci)	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)	Radionuclide Total (Ci)[b]
U-234	2.88E-02	2.94E-02	6.58E-03	5.95E-03	2.20E-02	1.69E-02	1.10E-01
U-235	2.00E-04	1.76E-04	6.49E-05	5.28E-05	2.84E-04	4.17E-04	1.19E-03
U-236	4.65E-04	4.45E-04	1.72E-04	1.44E-04	8.78E-04	9.30E-04	3.03E-03
U-238	1.13E-05	9.98E-06	1.14E-05	2.85E-05	1.55E-04	3.28E-04	5.45E-04
Zr-93	1.24E+00	1.50E+00	1.70E+00	9.79E-01	2.11E+00	9.82E-01	8.51E+00
Total (Ci)[e]	1.46E+04	1.41E+04	1.16E+04	6.93E+03	1.49E+04	6.71E+03	6.89E+04

a. Calculated as follows: $Post - waste\ retrieval\ Ci = Pre - waste\ retrieval\ Ci\ (Table\ 2-9) * \frac{Residual\ Percent\ Remaining\ (Table\ 2-10)}{100}$.

b. This is the sum of curies for all CSSFs for each radionuclide.

c. Stable isotope retained in table, but the inventory is listed as zero.

d. Noble gas retained in table, but the inventory is listed as not available.

e. This is the total sum of curies in each CSSF.

CSSF Calcined Solids Storage Facility
NA not available

Under current plans, CSSF 1 calcine will be transferred to CSSF 6 in the first phase of closure of the CSSF (DOE-ID 2022a). In the process, the volume and average concentration of calcine in CSSF 6 will change and the source term stated in EDF-11126 may require updating. However, the addition of the CSSF 1 calcine to CSSF 6 will have little impact on the concentrations in the residual calcine after closure of CSSF 6 because calcine will be removed from the bottom of the bins, creating a cone of depression (Bush, O'Connor, and Young 2017; EDF-11126). Once the cone of depression is created, the calcine near the top of the bins will flow to the bottom and be removed from the bin before the calcine that is lower in the bins is removed. For the purposes of the dose assessment calculations, the average concentration in CSSF 6 (as well as the other CSSFs) will continue to be used to define the concentrations in the residual calcine. The assumption is that the calcine retrieval plans will be optimized so that the residual calcine in each CSSF will result in lower curies than predicted, using the average inventory values. Visual inspection via video and monitoring of system pressure sensors will be a method used to determine that retrieval technology has been optimized and the deployed technology has reached its limit of application (no longer effectively removing waste) (Young 2019; ICP 2020, 2021b; Sandow 2021).

2.11.4 CSSF Closure Approach

The closure process includes removing calcine and then closing each CSSF in accordance with the State-approved closure plan. Additional details on the closure approach for the CSSF are described in the following subsections.

2.11.4.1 State-Approved Permit and Additional Information

An integrated closure approach applicable to each CSSF is being pursued in accordance with the current State-approved closure plan, consistent with the Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL (PER-114).

Following closure and in accordance with the PER-114 requirements, decisions and actions regarding capping (including capping remaining CSSF structures and equipment such as the storage vaults, bins, distributor pipes, cyclones, and transport lines), monitoring, and long-term maintenance of the closed facility will be coordinated with the CERCLA program. Under the CERCLA program, each CSSF will be closed as a non-time critical removal action under authority of the *Action Memorandum for General Decommissioning Activities under the Idaho Cleanup Project* (DOE-ID 2021a). That memorandum defines the process for performing general decommissioning activities at the INL Site as CERCLA non-time critical removal actions in accordance with the FFA/CO (DOE-ID 1991) and final comprehensive RODs pursuant to CERCLA for the INL Site.

Though general decommissioning of buildings and structures is not specifically addressed in the final comprehensive RODs for the INL Site, non-time critical removal actions under the Action Memorandum for General Decommissioning (DOE-ID 2021a) are consistent with the remedial action objectives established in the final comprehensive RODs and support the overall cleanup objectives identified in the FFA/CO. Addendums approved by the Idaho DEQ and EPA add facilities to the Action Memorandum for General Decommissioning. CSSF 1 was approved to be added to the Action Memorandum for General Decommissioning in 2017 after DOE submitted a request to the Idaho DEQ and EPA (Faulk 2017; Koch 2017; Whitham 2017).

CERCLA non-time critical removal action documentation will be developed as described below for each CSSF to evaluate the scope of decommissioning and to determine the end state configuration (including whether a barrier is needed to protect human health and the environment). Because each CSSF is considered a major facility, an engineering evaluation/cost assessment and action memorandum specific to the CSSF being closed will be prepared through the process defined in the Action Memorandum for General Decommissioning (DOE-ID 2021a). The removal action report will document the removal action, confirm remedial action objectives established in the OU 3-13 and OU 3-14 RODs (DOE-ID 1999a, 2007a) have been met, and ensure long-term stewardship functions, if required, are in place and functioning as intended.

For additional information, DOE must also comply with the DOE requirements in DOE M 435.1-1 Chg 3 concerning Tier 1 and 2 closure documentation and authorization for closure of the CSSF. As additional background information, DOE also may, in the future, decide whether it is appropriate to issue an amendment pursuant to the 2005 ROD (DOE 2005) concerning the CSSF.

2.11.4.2 Waste Removal

Calcine will be removed from the CSSF to the maximum extent practical for closure. Waste will be removed using a pneumatic retrieval system until removal of additional calcine is no longer practical. Retrieval operations will be performed in a manner that protects workers, public health and safety, and the environment. In preparation for the removal of actual calcine, the retrieval systems were tested in a full-scale mockup of a CSSF 1 nested bin, using $CaCO_3$ as simulated waste (ICP 2016; Sandow 2019). DOE decided to pursue this retrieval technology development and demonstration in the 2005 ROD (DOE 2005). DOE has evaluated several different retrieval concepts, and a pneumatic retrieval system was determined to be the most effective approach to remove calcine from the CSSF (ICP 2017). Previous studies and ongoing testing of the pneumatic retrieval and transfer system on the full-scale mockup have demonstrated the ability to remove simulated calcine sufficiently to meet or exceed the amount of waste volume assumed in the CSSF PA/CA (DOE-ID 2022a) (see Subsection 2.11.3 of this document). Remote visual monitoring within the bins, both during testing and actual operations, will provide an indication of waste removal effectiveness (Young 2019). Visual inspection via video will be a method used to confirm the effectiveness of waste removal.

Additional information on the retrieval technology and selection is provided in Subsection 5.2. The next subsection provides a summary of testing results performed on preliminary designs of the retrieval systems from 2017 to 2021. The designs and tests described below are specific to the retrieval system and equipment required to remove calcine from CSSF 1 and transfer it to CSSF 6; however, the results of testing have demonstrated that transfer is not limited to CSSF 6 and calcine can be transferred elsewhere, if needed.

2.11.4.2.1 Waste Retrieval Demonstration Results from 2017 to 2021—Since 2016, the Calcine Retrieval Project has completed numerous tests described in *Calcine Retrieval Project Conceptual Design for the Transfer of Calcined Solids from CSSF 1 to CSSF 6* (ICP 2016). Objectives of the tests were to mitigate safety hazards, eliminate design risk, and optimize design configurations for retrieval and transfer of calcine. Results from completed waste retrieval tests provide evidence that the retrieval technology used on the full-scale mockup effectively removes simulant material (i.e., $CaCO_3$) and will keep occupational exposure to radiation ALARA (ICP 2022).

In 2017, testing was successfully completed for a thermowell conversion (i.e., removing the thermocouple wires from the thermowell pipes, cutting off the bottom thermowell cap, and installing the retrieval system platform), distributor and fill line cleanout, vault core drilling and capture (i.e., capture of the concrete core from drilling through the vault, which will be captured and not allowed to fall into the vault/bin structure), retrieval vacuum line installation, access riser placement, and preliminary retrieval system designs. Testing of the preliminary designs of the retrieval systems was performed on a full-scale replica of a CSSF 1 nested bin. Testing results identified future work to refine designs and develop other components to optimize waste removal. Testing results are summarized in the *Calcine Retrieval Project—Test Report for 2017* (Burnett and Graham 2017).

In 2018, the CSSF 1 nested bin replica was transported to, and installed at, the former Fuel Reprocessing Restoration Facility (CPP-691) at INTEC (see Subsection 5.2.1.1 for additional details on the full-scale mockup). Operations at CPP-691 were originally intended to support INTEC SNF reprocessing; however, construction and installation of equipment ended in 1992 when DOE discontinued SNF reprocessing operations. The Calcine Retrieval Project rehabilitated the abandoned building by removing excess equipment and supplies, establishing permanent electrical power, and installing required life safety equipment in areas of the building used by the Calcine Retrieval Project. CPP-691 is currently being used as the demonstration area to test retrieval designs and operations. Construction and installation of the full-scale bin replica and transfer equipment were completed in early 2019.

Testing in 2018 primarily focused on the bin surface cleaning tool, residual cleanout systems, the access riser connection system, and the bin visual inspection system. Test results concluded that additional development and testing are needed before final implementation of these systems and tools. Test results are summarized in the *Calcine Retrieval Project FY 2018 Test Report* (Burnett and Graham 2018), and technology development and testing continued into 2019.

Testing and operations of the full-scale integrated mockup in 2019 continued advancing the designs of the different components and systems needed to remove calcine from the CSSF. Operations of the full-scale integrated mockup has provided valuable data that have improved the design of the retrieval and transfer system. Test results are summarized in the *Calcine Retrieval Project FY 2019 Test Report* (ICP 2020), which identifies the next stages of design development. Testing continued through 2021—documented in *Calcine Retrieval Project Status Report – Fiscal Years 2020 and 2021* (ICP 2021b)—and an independent assessment determined that designs for retrieval systems have been proven (Balls et al. 2021). Results of the retrieval demonstration in 2021 have proven the effectiveness of the retrieval system and the ability to meet the residual waste analyzed in the CSSF PA/CA (DOE-ID 2022a). While Calcine Retrieval Project testing has primarily focused on demonstrating the ability to retrieve and transfer calcine from CSSF 1 to CSSF 6, the pneumatic retrieval processes are applicable to each CSSF and calcine can be transferred elsewhere, if needed.

2.11.4.3 Residual Waste Stabilization and End State Configuration

The closure process includes removing calcine and then closing each CSSF. Calcine will be removed from the CSSF to the maximum extent practical, and remote visual inspections will be performed. Disposal of non-calcine waste (e.g., soil, concrete, and personnel protective equipment) generated during closure will be coordinated with the ICP Waste Generator Services organization in accordance with approved ICP procedures. Residual waste or waste transfer equipment (including the distributor lines or retrieval equipment such as the access risers or vacuum lines) that cannot be removed after retrieval operations are complete will be stabilized with grout within the bin(s) or storage vault at closure.

The closure configuration for the CSSF is anticipated to include the concrete storage vaults, stainless-steel bins, and void spaces (storage vaults, bins, and piping) filled with grout, which will serve to provide long-term structural stability, limit the amount of water infiltration into the bins to mitigate contaminant migration, and provide a barrier against intrusion by burrowing animals, roots, or a hypothetical human intruder.

The grout will be made from materials such as cement, fly ash, fine aggregate, and water to create a free-flowing material than will be used to fill the bins and vaults after waste retrieval operations are complete. The grout will harden in the bins and vault structures to stabilize and encapsulate the residual waste and provide structural stability for closure of the CSSF. As stated above, the grout will immobilize the residuals, minimize water infiltration, and discourage intrusion. DOE will tailor and finalize the specific formulation of the grout in the future before it is added to the bins and vaults.[36]

Preliminary studies have been conducted to evaluate possible end states for the CSSF (EDF-11201; EDF-11231). These studies were performed to support future closure options that may be considered and were used to develop a bounding closure configuration modeled in the base case for the PA (DOE-ID 2022a). Closure configurations considered in the preliminary studies were scenarios that have been evaluated and approved by the State for other facilities at the INL Site. However, further analysis of the end state alternatives will be conducted for each CSSF as they are closed. Involvement with regulatory authorities and the public will be required to determine the end state for the CSSF.

In addition, the grouted bins and remaining structures and equipment may be covered with a closure cap.[37] An engineered closure cap would provide additional physical stabilization of the closed site, limit infiltration and further reduce the impact of burrowing plants and animals, and limit human intrusion into the waste;[38] however, a specific closure cap design has not been determined for use at the CSSF at this time (see discussion in Subsection 2.11.4.2), and the CSSF PA/CA (DOE-ID 2022a) did not consider an engineered cap in the analysis to demonstrate compliance with the requirements of 10 CFR 61, Subpart C. EDF-11201, however, evaluated the extent of grading and volume of soil needed for different closure concept scenarios where an earthen barrier is placed over each CSSF. The

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36. Each CSSF will be grouted with materials such as cement, fine aggregate, and water to create a free-flowing material that will be used to fill the bins after waste retrieval is completed. The grout formula from the TFF (EDF-1464) will be used with some modification. The grout formulas may change, but the general performance objectives will remain the same.

37. This Draft CSSF 3116 Basis Document and its supporting references do not assume or take credit for a potential closure cap.

38. EDF-11201, "Conceptual Site Grading Plans for CSSF 1 to 6," evaluated the extent of grading and volume of soil needed for different closure concept scenarios where an earthen barrier is placed over each CSSF. EDF-11201 assumed the earthen barrier is not an engineered barrier that would reduce infiltration or prevent biointrusion. This is because the CSSF PA/CA (DOE-ID 2022a) will not drive any design requirements for a cap, and this is in part due to existing engineered features of the facility (e.g., concrete storage vault, stainless-steel bins, and depth of waste).

evaluation also determined that an earthen barrier would not connect to, or need to be integrated with, other nearby closure caps (planned and existing). As previously stated, further analysis and involvement with the State of Idaho will be required to determine the end state for the CSSF.[39]

Design features described above serve to impede the release of stabilized contaminants into the general environment. These features, along with removal of HRRs to the maximum extent practical, are consistent with the applicable DOE requirements for closure.

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39. Closure details for each CSSF will be provided in their respective State-approved closure plans under PER-114.

3. SECTION 3116 OF THE RONALD W. REAGAN NATIONAL DEFENSE AUTHORIZATION ACT FOR FISCAL YEAR 2005

Section 3116(a) of the NDAA provides that radioactive waste in Idaho or South Carolina that results from reprocessing SNF is not "high-level radioactive waste" that the Secretary of Energy determines, in consultation with the NRC, meets certain specified criteria.

Section 3116(a) of the NDAA states:

> In General—Notwithstanding the provisions of the Nuclear Waste Policy Act of 1982, the requirements of section 202 of the Energy Reorganization Act of 1974, and other laws that define classes of radioactive waste, with respect to material stored at a Department of Energy site at which activities are regulated by a covered State pursuant to approved closure plans or permits issued by the State, the term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy (in this section referred to as the "Secretary"), in consultation with the Nuclear Regulatory Commission (in this section referred to as the "Commission"), determines—
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical; and
>
> (3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, and will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of—
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations, and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and
>
> (iii) pursuant to plans developed by the Secretary in consultation with the Commission

As will be demonstrated in the next five sections of this Draft CSSF 3116 Basis Document, DOE has evaluated the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure against these criteria, and for the reasons presented, this Draft CSSF 3116 Basis Document shows that the closed CSSF meets the applicable Section 3116(a) criteria and may be disposed of (closed) in place as LLW.

4. WASTE DOES NOT REQUIRE PERMANENT ISOLATION IN A DEEP GEOLOGICAL REPOSITORY

Section 4 Purpose

This section provides information to support the conclusion that the CSSF waste does not require permanent isolation in a deep geological repository.

Section 4 Contents

This section describes the CSSF waste in terms of the waste not requiring permanent isolation in a deep geological repository.

Section 4 Key Points

- The stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet Criterion (1) of NDAA Section 3116(a).
- The stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet the performance objectives of 10 CFR 61, Subpart C, to provide protection of public health and the environment.
- The stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not require disposal in a deep geologic repository due to the risk to public health and safety. Furthermore, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not raise any unique considerations that, notwithstanding these demonstrations, require permanent isolation in a deep geologic repository.

This section provides information showing that stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein remaining within the CSSF, upon completion of waste retrieval and stabilization activities at the time of closure, will not require permanent isolation in a deep geological repository for HLW per Criterion (1) of NDAA Section 3116(a) (Public Law 108-375).

NDAA Section 3116(a) provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy…, in consultation with the Nuclear Regulatory Commission…, determines –
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste.

Under NDAA Section 3116(a), certain wastes from reprocessing are not "high-level radioactive waste" if the Secretary of Energy, in consultation with the NRC, determines that certain criteria have been met. NDAA Section 3116(a) sets out three criteria. Criterion (2), which is set forth in NDAA Section 3116(a)(2), requires removal of HRRs to the maximum extent practical. Criterion (3) generally mirrors the regulatory criteria that the NRC has established for determining whether waste qualifies for land disposal as LLW. That criterion provides that disposal of the waste must meet the performance objectives of 10 CFR 61, Subpart C, and that residual waste must not exceed concentration limits for

Class C waste in 10 CFR 61.55 or, if it exceeds Class C concentration limits, it must be disposed of pursuant to plans developed by the Secretary in consultation with NRC, and that disposal onsite must be pursuant to a State-approved closure plan or permit. Subsequent sections of this Draft CSSF 3116 Basis Document will demonstrate that Criteria (2) and (3) are satisfied.

Criterion (1), quoted above, is a broader criterion that allows the Secretary of Energy, in consultation with the NRC, to consider whether other factors warrant permanent isolation of the radioactive waste in a deep geologic repository. Generally, such factors would be unusual because waste that meets Criterion (3) would be disposed of in a manner that meets the 10 CFR 61, Subpart C, performance objectives and either falls within one of the classes set out in 10 CFR 61.55 that the NRC has specified are considered "generally acceptable for near-surface disposal" or regarding which the Secretary of Energy has consulted with NRC concerning DOE's disposal plans.[40] Normally, it follows that if disposal of a waste stream in a facility that is not a deep geologic repository will meet these objectives, in the ordinary case, that waste does not "require permanent isolation in a deep geologic repository" because non-repository disposal will be protective of public health and safety.

However, it is possible that in rare circumstances a waste stream that meets Criterion (3) might have some other unique radiological characteristic or may raise unique policy considerations that warrant its disposal in a deep geologic repository. Criterion (1) of NDAA Section 3116(a) is an acknowledgment by Congress of that possibility. For example, the waste stream could contain material that, while not presenting a health and safety danger if disposed of at the near- or intermediate-surface, nevertheless presents nonproliferation risks that the Secretary concludes cannot be adequately guarded against absent deep geologic disposal.[41] Criterion (1) allows the Secretary of Energy, in consultation with the NRC, to consider such factors in determining whether waste that meets the other two criteria may need permanent isolation in a deep geologic repository in light of these considerations.

This is not the case here. As demonstrated later in this Draft CSSF 3116 Basis Document, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of CSSF closure will meet the performance objectives of 10 CFR 61, Subpart C, so as to provide for the protection of the public health and the environment. Accordingly, the waste does not require disposal in a deep geologic repository to protect public health and safety. Furthermore, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not raise any unique considerations that, notwithstanding these demonstrations, require permanent isolation in a deep geologic repository. Accordingly, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein meet Criterion (1) of NDAA Section 3116(a).

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40. As the NRC explained in *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services)* (NRC 2005), the 10 CFR 61, Subpart C, performance objectives in turn "set forth the ultimate standards and radiation limits for (1) protection of the general population from releases of radioactivity; (2) protection of individuals from inadvertent intrusion; (3) protection of individuals during operations; and (4) stability of the disposal site after closure."

41. In NUREG-1854, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations* (NRC 2007), the NRC similarly explains: "In general, there is reasonable assurance that this criterion can be met if the two other NDAA criteria can be met. In other words, if highly radioactive radionuclides have been removed to the maximum extent practical and the waste will be disposed of in compliance with the performance objectives in 10 CFR Part 61, Subpart C (which are the same performance objectives NRC uses for disposal of low-level waste), then this supports a conclusion that the waste does not require disposal in a deep geologic repository. However, this criterion allows for the consideration that waste may require disposal in a geologic repository even though the two other NDAA criteria may be met. Those circumstances under which geologic disposal is warranted to protect public health and safety and the environment could be considered; for example, unique radiological characteristics of waste or nonproliferation concerns for particular types of material."

5. WASTE HAS HAD HIGHLY RADIOACTIVE RADIONUCLIDES REMOVED TO THE MAXIMUM EXTENT PRACTICAL

Section 5 Purpose

NDAA Section 3116(a) provides that certain waste resulting from reprocessing is not HLW if the Secretary of Energy, in consultation with the NRC, determines, among other things, that the waste has had HRRs removed "to the maximum extent practical." This section demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein, upon completion of waste retrieval activities at the time of closure, will have had HRRs removed to the maximum extent practical and meet this criterion.

Section 5 Contents

This section identifies the HRRs for the purpose of this Draft CSSF 3116 Basis Document; describes the retrieval processes used to remove HRRs to the maximum extent practical; and demonstrates that, at closure, the HRRs will have been removed to the maximum extent practical.

Section 5 Key Points

- The CSSF bins (including integral equipment), transport lines, and any residual calcine therein will have had HRRs that are contained in the calcine removed to the maximum extent practical at the time of CSSF closure, thereby meeting Criterion (2) of NDAA Section 3116(a). HRRs include all radionuclides important to meeting the performance objectives identified in 10 CFR 61, Subpart C, and all radionuclides that are listed in Tables 1 and 2 of 10 CFR 61.55 and present in the calcine.
- The list of HRRs identified for this Draft 3116 Basis Document are Tc-99 from the CSSF PA/CA groundwater analysis and Am-241, C-14, Co-60, Cs-135, Cs-137, H-3, I-129, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-242, and Sr-90 from Tables 1 and 2 of 10 CFR 61.55. Additionally, HRRs include Am-242m, Am-243, Cf-249, Cf-250, Cf-251, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, and Pu-244 for alpha-emitting transuranic nuclides with a half-life >5 years provided in Table 1 of 10 CFR 61.55. These radionuclides were selected using a risk-informed approach identifying those radionuclides that may contribute significantly to the radiological risk to workers, the public, and the environment, while also taking into account scientific principles, knowledge, and expertise.
- A pneumatic retrieval system has been determined to be the most cost-effective, reliable, and safest approach to remove calcine containing HRRs. Waste removal using a pneumatic retrieval system is expected to leave no more than a 5.1-cm-thick (2-in.-thick) residual depth of calcine in each CSSF.
- Waste removal to this depth removes approximately 99% or more of the waste volume and approximately 99% of the radioactivity attributable to HRRs from the CSSF. This is the depth of calcine at the time of closure that DOE studies and tests have shown will likely remain upon completion of waste retrieval using a pneumatic retrieval system.
- Retrieval technologies and methods will be optimized so that residual calcine in each CSSF will be minimized.
- Based on this analysis, further removal of CSSF residual waste after retrieval operations, which are expected to result in the 5.1-cm (2-in.) residual depth of calcine, is not practical.

This section demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein, upon completion of waste retrieval activities, will have had key radionuclides removed to the maximum extent practical in accordance with Criterion (2) of NDAA Section 3116(a) (Public Law 108-375).

NDAA Section 3116(a) provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines —
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical.

DOE views HRRs to be those radionuclides that, using a risk-informed approach, contribute most significantly to radiological risk to workers, the public, and the environment. The HRRs are those present in the waste and listed in Tables 1 and 2 of 10 CFR 61.55, as well as those that are important to satisfying the performance objectives of 10 CFR 61, Subpart C.

5.1 Highly Radioactive Radionuclides

The list of HRRs was developed beginning with the radionuclide inventory described in Subsection 2.11.3. DOE reviewed this initial radionuclide inventory and identified the radionuclides that were important in meeting performance objectives in 10 CFR 61, Subpart C, because they contribute to the dose to the workers, the public, and/or the inadvertent intruder (for one or more reasonable intruder scenarios) in the CSSF PA/CA base case and sensitivity and uncertainty analyses. In DOE's view, this approach results in a risk-informed list of HRRs that includes (1) short-lived radionuclides that may present risk because they produce radiation emissions that, without shielding or controls, may harm humans simply by proximity without inhalation or ingestion and (2) long-lived radionuclides that persist well into the future, may be mobile in the environment, or may pose a risk to humans if inhaled or ingested. The list of HRRs also includes all radionuclides that are listed in Tables 1 and 2 of 10 CFR 61.55 and present in the calcine.

5.1.1 Highly Radioactive Radionuclides Retained from the CSSF Performance Assessment

Radionuclides were identified for the CSSF PA/CA (DOE-ID 2022a)[42] using screening evaluations that considered radionuclide half-lives, ingrowth of constituents from chain decay (i.e., a series of radioactive decays of different radioactive decay products as a sequential series of transformations, such as in the uranium decay series), and activity levels. The following summarizes analyses performed for the CSSF PA/CA that show dose results by radionuclide to identify HRRs that contribute most significantly to radiological risk to the workers, members of the public, and hypothetical inadvertent human intruder.[43]

5.1.1.1 Groundwater All-Pathways Dose

Groundwater modeling for the CSSF PA/CA was used to determine an annual all-pathways ED[44] to a member of the public. Results of the groundwater annual all-pathways dose analysis are summarized in Section 6 of this Draft CSSF 3116 Basis Document. The deterministic base case analysis projected the peak annual all-pathways ED to a public receptor (i.e., an individual located 100 m [328 ft] downgradient from the CSSF) to be well below the 25-mrem annual dose performance objective.[45]

All-pathways doses for the CSSF receptors were considerably less than the 25-mrem annual performance objective during the 1,000-year post closure period[46] and beyond that period to peak dose. The all-pathways annual dose was essentially zero (6.79E-14 mrem) for the 1,000-year post-closure period after CSSF closure. All-pathways doses for the 1,000- to 10,000-year post-closure period were highest at the CSSF 1 receptor (3.45E-02 mrem) followed by the CSSF 2 receptor (3.02E-02 mrem) (see Table 5-1). The peak annual all-pathways dose of 1.9E-01 mrem (rounded to two significant digits) was projected to

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42. A PA and CA are required and maintained pursuant to DOE M 435.1-1 Chg 3. Generally, a PA is a multidiscipline assessment (e.g., geochemistry, hydrology, materials science, and health physics) that uses a variety of computational modeling codes to evaluate groundwater concentrations and doses at various points of the assessment over time. In doing this assessment, DOE evaluates the impact of natural features (e.g., hydrology, soil properties, groundwater infiltration) and engineered barriers (e.g., closure cap, fill grout, bin design) on the release of radionuclides to estimate, among other things, the potential dose to a hypothetical member of the public and a hypothetical inadvertent intruder. The results of the CSSF PA/CA, as reported here, should not be considered limits or thresholds. As required by DOE M 435.1-1 Chg 3, maintenance of the CSSF PA/CA will include future PA revisions or special analyses to incorporate new information, update model codes, and reflect the analysis of actual residual inventories.

43. DOE normally would view radionuclides as making an insignificant contribution if the contribution to dose from those radionuclides, in both the expected case and sensitivity analyses, does not exceed any of the following: (1) 10% of the 25-mrem/year all-pathways annual dose to the public performance objective, (2) 10% of the DOE 100-mrem annual dose limit to the intruder (under all reasonable intruder scenarios), (3) 10% of the DOE 500-mrem acute dose limit to the intruder (under all intruder scenarios), and (4) 10% of the annual worker dose in the relevant provisions of 10 CFR 20, "Standards for Protection Against Radiation." This methodology is based on NRC consultation and is intended to be consistent with the guidance and general approach in NUREG-1757 (NRC 2006), which explains that "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10% of the dose criteria to be insignificant contributors." DOE has previously used 5% of the 25-mrem/year all-pathways dose limit (i.e., 1.25 mrem/year) to ensure the selection of HRRs is conservative (DOE 2014); however, for this Draft CSSF 3116 Basis Document, 10% of the all-pathways dose limit will be used to be consistent with NRC guidance as further conservatism does not impact the selection of HRRs identified for the CSSF. The above-referenced NUREG, which applies to NRC licensees, is being used only as general guidance, and DOE's use of this NUREG as guidance should not be construed to suggest that it is a requirement under NDAA 3116 or that either the NUREG or 10 CFR 20, Subpart E, "Radiological Criteria for License Termination," is applicable to DOE.

44. In the CSSF PA/CA, dose coefficients used to evaluate the all-pathways dose, atmospheric doses, and intruder doses are taken from DOE-STD-1196-2011, "Derived Concentration Technical Standard." The term "effective dose equivalent" used in DOE-STD-1196-2011 is now referred to as the effective dose, or ED. ED is a weighted sum of the equivalent doses in the various organs and tissues of the body. In this Draft CSSF 3116 Basis Document, all doses are reported as ED. The CSSF PA/CA will be updated to reflect the updated dose coefficients in DOE-STD-1196-2021 under the PA maintenance program in the future.

45. See Section 1.11.1 of the CSSF PA/CA (DOE-ID 2022a) for details on the deterministic base case assumptions.

46. The 1,000-year post-closure period is considered the compliance period for a PA in DOE M 435.1-1 Chg 3. NRC NUREG-1854 recommends generally evaluating the doses out to 10,000 years after closure.

occur at 19,500 years after CSSF closure (see Table 5-1). This peak dose is well below the 25-mrem annual dose specified in the performance objective at 10 CFR 61.41, as discussed further in Section 6 of this Draft CSSF 3116 Basis Document.

All groundwater annual all-pathways doses at all evaluated time periods were considerably less than the 25-mrem annual performance objective and essentially zero for the first 1,000-year post-closure period. Tc-99 dominates the groundwater doses during the post-closure period. Therefore, Tc-99 was retained as an HRR from the CSSF PA/CA groundwater base case analysis for this Draft CSSF 3116 Basis Document even though the dose contribution is insignificant.

As shown in Table 5-2 and Figure 5-1, all-pathways annual doses are dominated by Tc-99 (1.91E-01 mrem) followed by Se-79 (2.19E-03 mrem) between 10,000 and 50,000 years and by Np-237 and progeny (2.03E-3 mrem) after 100,000 years. The actinides (Pu-239, Pu-240, and Pu-242) and Cs-135 contribute little to the all-pathways dose until >500,000 years after CSSF closure, because these radionuclides sorb strongly in the vadose zone and, consequently, have long unsaturated transit times.

Table 5-1. Maximum groundwater annual all-pathways effective dose at the receptor 100 m (328 ft) south of each Calcined Solids Storage Facility for various time windows.

Time Period (yr)	Maximum Groundwater Annual All-Pathways Effective Dose at the Receptor 100 m (328 ft) South of Each CSSF (mrem)[a]						
	CSSF 1	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6	Maximum
0–100	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
100–500	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
500–1,000	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
1,000–5,000	6.34E-04	5.55E-04	7.93E-05	1.02E-05	1.10E-05	5.82E-06	6.34E-04
5,000–10,000	3.45E-02	3.02E-02	4.20E-03	2.92E-06	3.16E-06	1.67E-06	3.45E-02
10,000–50,000	1.91E-01	1.67E-01	2.33E-02	1.78E-03	1.92E-03	1.02E-03	1.91E-01
50,000–100,000	4.10E-02	3.70E-02	2.59E-02	1.91E-02	2.07E-02	1.09E-02	4.10E-02
100,000–500,000	1.61E-02	1.46E-02	1.02E-02	2.02E-02	2.19E-02	1.16E-02	2.19E-02
>500,000	1.94E-03	1.95E-03	2.66E-03	9.33E-03	9.50E-03	3.32E-03	9.50E-03

a. Doses presented as less than 1E-10 are essentially zero.
CSSF Calcined Solids Storage Facility

Table 5-2. Maximum groundwater annual all-pathways effective dose by radionuclide at the receptor 100 m (328 ft) south of Calcined Solids Storage Facility 1. Progeny radionuclides are indented for clarity.

	Maximum Groundwater Annual All-Pathways Effective Dose by Radionuclide at the Receptor 100 m (328 ft) South of CSSF 1 (mrem)[a]								
Radionuclide	0–100 yr	100–500 yr	500–1,000 yr	1,000–5,000 yr	5,000–10,000 yr	10,000–50,000 yr	50,000–100,000 yr	100,000–500,000 yr	>500,000 yr
Cs-135	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	6.10E-09	1.13E-04	4.01E-04
H-3	<1E-10	<1E-10	<1E-10	<1E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129	<1E-10	<1E-10	<1E-10	<1E-10	7.90E-07	2.48E-04	6.96E-05	7.18E-05	<1E-10
Se-79	<1E-10	<1E-10	<1E-10	<1E-10	8.76E-07	2.19E-03	1.07E-03	5.48E-04	<1E-10
Tc-99	<1E-10	<1E-10	<1E-10	6.34E-04	3.45E-02	1.91E-01	4.07E-02	1.56E-02	0.00E+00
Pu-239[b]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	6.08E-09	2.36E-07	4.67E-06	1.05E-05
Pu-240[b]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	4.96E-09	1.50E-08	8.05E-06	1.86E-05
Np-237[b]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	3.93E-08	2.51E-05	2.03E-03	1.52E-03
Pu-242[b]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	4.26E-10	1.57E-07	2.84E-04	3.82E-04
Total Dose[c]	<1E-10	<1E-10	<1E-10	6.34E-04	3.45E-02	1.91E-01	4.10E-02	1.61E-02	1.94E-03
Individual Decay Chain Members									
Pu-239	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.96E-10	<1E-10
U-235[d]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	5.79E-09	2.01E-07	3.33E-06	7.03E-06
Pa-231	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.13E-10	1.30E-08	4.85E-07	1.28E-06
Ac-227	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.85E-10	2.26E-08	8.52E-07	2.25E-06
Pu-240	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
U-236[e]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	4.96E-09	1.50E-08	8.05E-06	1.86E-05
Th-232	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
Ra-228	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	4.76E-10
Th-228	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10

Table 5-2. (continued).

Radionuclide	Maximum Groundwater Annual All-Pathways Effective Dose by Radionuclide at the Receptor 100 m (328 ft) South of CSSF 1 (mrem)[a]								
	0–100 yr	100–500 yr	500–1,000 yr	1,000–5,000 yr	5,000–10,000 yr	10,000–50,000 yr	50,000–100,000 yr	100,000–500,000 yr	>500,000 yr
Np-237	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	3.30E-08	2.18E-05	1.72E-03	1.20E-03
U-233	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	5.68E-09	2.80E-06	2.56E-04	2.44E-04
Th-229	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	5.38E-10	4.74E-07	7.62E-05	7.32E-05
Pu-242	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.46E-08	2.05E-06
U-238[f]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.37E-10	1.85E-07	3.88E-07
U-234[f]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	4.05E-10	1.36E-07	1.27E-04	1.44E-04
Th-230[f]	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.22E-09	7.48E-06	1.15E-05
Ra-226	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.80E-09	1.35E-05	2.10E-05
Pb-210	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	1.81E-08	1.36E-04	2.12E-04

a. Doses presented as less than 1E-10 are essentially zero.

b. Total decay chain dose.

c. The total is the maximum total dose during that time period and not the sum of the maximum doses shown for each radionuclide. The maximums for each radionuclide can be at different times during the time period. The total is the maximum dose during that time period, including the time-dependent contributions from each radionuclide.

d. U-235 has an initial inventory.

e. U-236 has an initial inventory.

f. U-238, U-234, and Th-230 have an initial inventory.

CSSF Calcined Solids Storage Facility

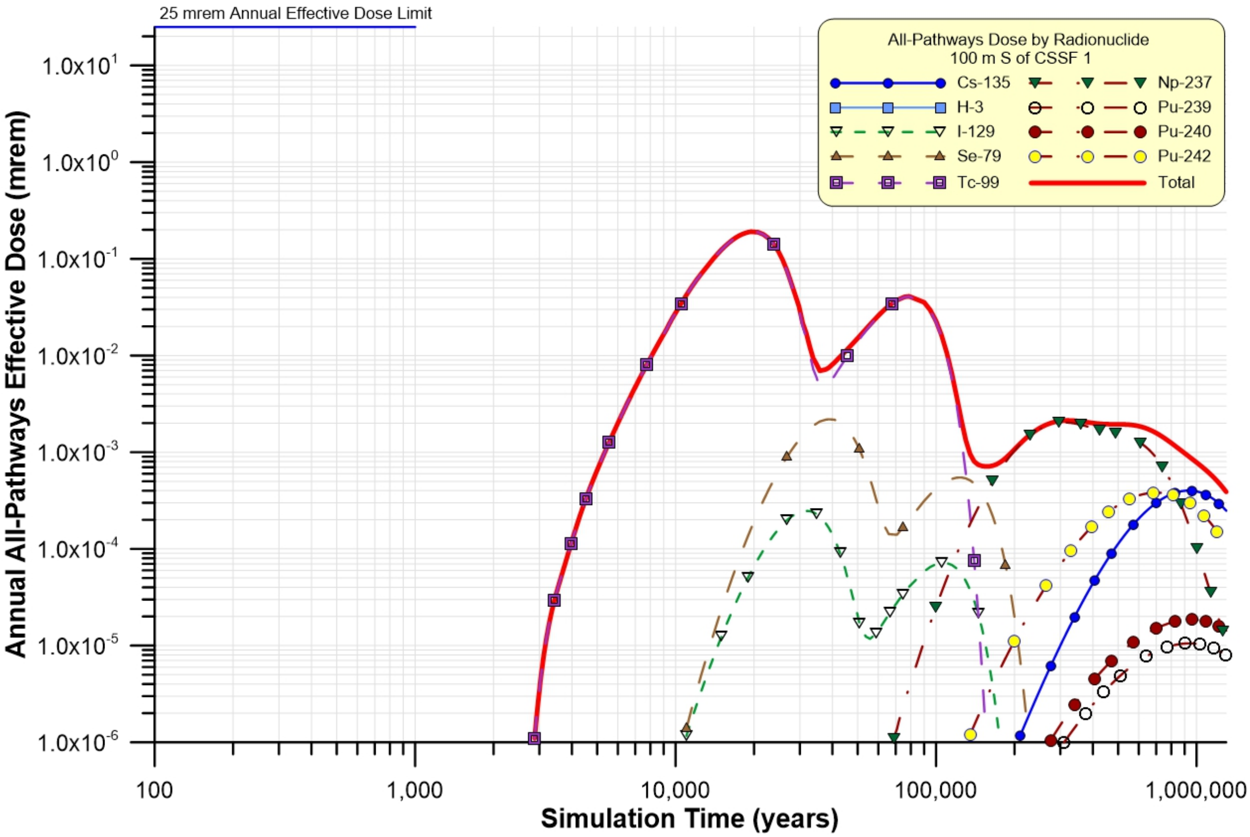

Figure 5-1. Log-log plot of the groundwater annual all-pathways effective dose by radionuclide at the receptor 100 m (328 ft) south of Calcined Solids Storage Facility 1.

5.1.1.1.1 Groundwater All-Pathways Sensitivity and Uncertainty Analysis—This subsection presents results of the groundwater all-pathways sensitivity and uncertainty analysis. Details of the groundwater all-pathways sensitivity and uncertainty analysis are presented in the CSSF PA/CA (DOE-ID 2022a). The groundwater all-pathways sensitivity and uncertainty analysis includes six one-factor-at-a-time (OFAT) cases. These OFAT cases examined the sensitivity of the output variable (all-pathways dose) to various assumptions and parameters. Results of the six OFAT sensitivity analysis cases are as follows:

- ***OFAT Case 1: Failure of the Containment Structure Due to a Seismic Event***—This OFAT case addresses the significance of early failure of the grout and stainless-steel containment system due to a seismic event. The groundwater modeling domain for each CSSF does not explicitly include the grouted vault, but only the grouted stainless-steel bin. The grouted vault controls the water flux that may potentially enter the stainless-steel bin and the water flux through the concrete base while the grout is intact. This process is modeled by assigning a time-dependent water flux through the grout and concrete base. This water flux determines the maximum amount of water that may enter a failed stainless-steel bin. In addition, the groundwater model domain does not explicitly include the stainless-steel bins. The stainless-steel bins control the timing, due to corrosion, when water flowing through the grout may potentially enter the grouted stainless-steel bins.

  A seismic event is pessimistically assumed to occur during the 1,000-year post-closure period at 500 years. An evaluation of the structural integrity of the CSSF after 500 years indicated that the bins could still withstand a design-basis earthquake[47] (DOE-ID 2021b). Therefore, the assumption of failure

47. Design basis earthquake criteria for the CSSF can be found in Table 2-2 of SAR-105, "Safety Analysis Report for the Calcined Solids Storage Facilities."

of the containment structure from an earthquake in 500 years is not expected to occur but was evaluated to address potential underestimates in the longevity of the containment.

This OFAT case assumes the grout surrounding and filling the bins, including the reinforced-concrete vault, is damaged during the earthquake and fails completely over the next 100 years. As a result, by simulation year 600, the infiltration rate through all the concrete and grout is the background rate of 1 cm/year. In addition, the stainless-steel bins are damaged, and the stainless-steel surface is exposed to additional water and air, resulting in enhanced corrosion. So, over the next 100 years, 97.2% of the stainless steel fails, and complete failure occurs by simulation year 700. Early failure of the stainless steel was modeled by setting the geometric mean failure time at 500 years and the geometric standard deviation to 1.1 for each CSSF. Therefore, by simulation year 600, most of the grouted residual calcine in the bins will be available for transport to the subsurface, and the infiltration rate will have reached the background infiltration rate of 1 cm/year. The Structural Integrity Program for the CSSF (DOE-ID 2021b) calculated expected service lifetimes for the bins based on average corrosion rates for stainless steel in calcine and a corrosion allowance of 16 mils. Expected service lifetimes, which include the calcine remaining in the bins and withstanding a seismic event, ranged from 1,720 to 1,840 years. Thus, assuming initial failure of the bins' containment in 500 years and almost total failure by simulation year 600 is extremely pessimistic.

The results of this OFAT case are illustrated in Figure 5-2, which shows the groundwater annual all-pathways dose as a function of time. The peak dose of 6.82E+00 mrem occurs much earlier than in the base case, and all-pathways doses are much higher during the 1,000-year post-closure period (1.78E-01-mrem peak annual dose at 1,000 years) compared to the base case. However, doses for all times post-closure for this extremely pessimistic case are still well below the 25-mrem annual dose performance objective. Results of this OFAT case indicate that uncertainty in the time of containment of the residual calcine will not result in all-pathways doses greater than a 25-mrem annual dose. Tc-99 was retained as an HRR from the CSSF PA/CA OFAT Case 1 analysis for this Draft CSSF 3116 Basis Document. All radionuclides shown in Figure 5-2 for OFAT Case 1 were included as HRRs (see Subsection 5.1.3).

- ***OFAT Case 2: No Containment Provided by the Stainless-Steel Bins—***This OFAT case evaluates the significance of the stainless-steel bins in containing the grouted residual calcine. Thus, in this case, no containment from the stainless-steel bins was assumed. This OFAT case addresses possible underestimates in the measured long-term corrosion rates for stainless steel.

  For this extreme case, the only barrier was the grout, which was assumed to degrade at the base-case rates. The groundwater modeling domain for each CSSF does not explicitly include the grouted vault, but only the grouted stainless-steel bin. The grouted vault controls the water flux that may potentially enter the stainless-steel bin and the water flux through the concrete base while the grout is intact. This process is modeled by assigning a time-dependent water flux through the grout and concrete base, and this water flux determines the maximum amount of water that may enter a failed stainless-steel bin. In addition, the groundwater model domain does not explicitly include the stainless-steel bins. The stainless-steel bins control the timing, due to corrosion, when water flowing through the grout may potentially enter the grouted stainless-steel bins.

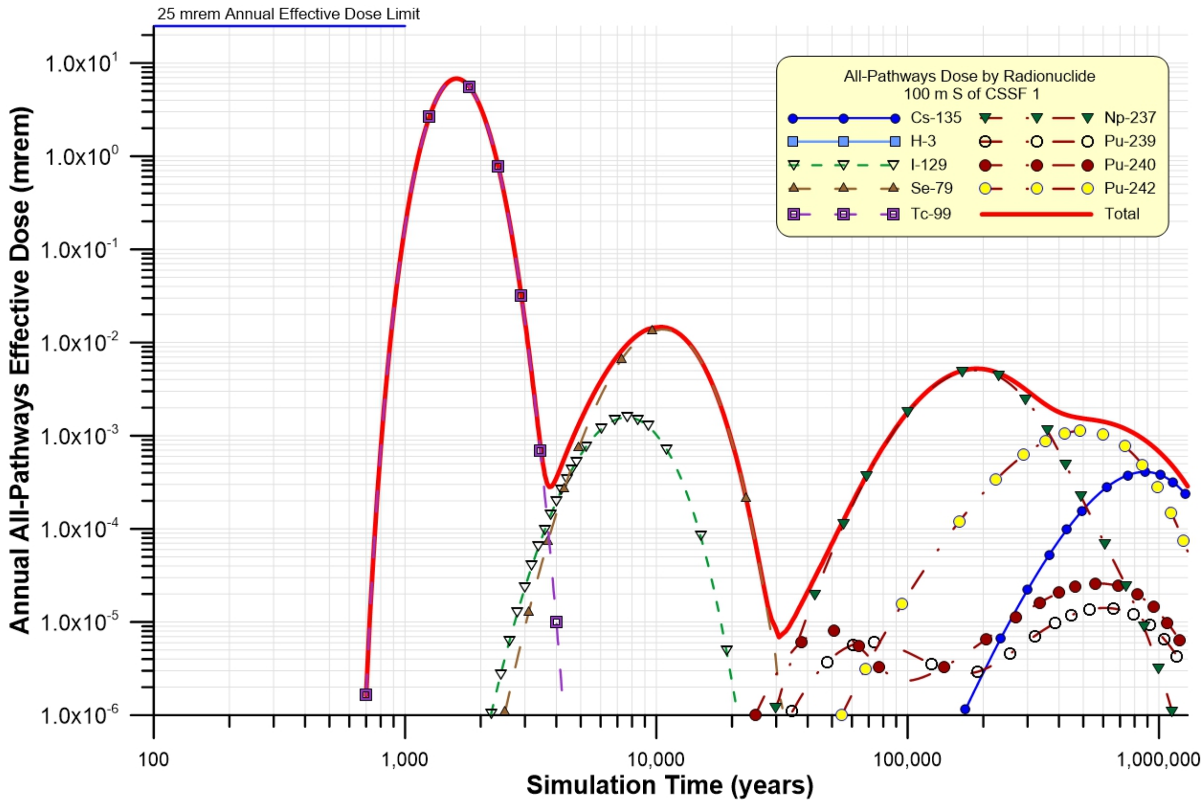

Figure 5-2. Groundwater annual all-pathways effective dose as a function of time for OFAT Case 1: Failure of the containment structure due to a seismic event.

Results are shown in Figure 5-3. The peak annual all-pathways dose during the 1,000-year post-closure period was the same as the base-case value of 3.40E-15 mrem (less than 1E-10 and thus essentially zero) because the grout limits infiltration during this time. During the post-closure period of 1,000 to 10,000 years and after the grout has failed, the peak dose was 5.96E+00 mrem in simulation year 3,675. Even for this unrealistic case where there is no credit taken for the barrier provided by the stainless-steel bins, the peak dose is still less than the 25-mrem annual dose performance objective. This OFAT case demonstrates that even when no credit is taken for the stainless-steel bins—corresponding to 100% failure of the stainless steel for each CSSF in simulation year 100—performance objectives are still met for the CSSF closure. Tc-99 was retained as an HRR from the OFAT Case 2 analysis for this Draft CSSF 3116 Basis Document, even though the dose contribution is insignificant. All radionuclides shown in Figure 5-3 for OFAT Case 2 were included as HRRs (see Subsection 5.1.3).

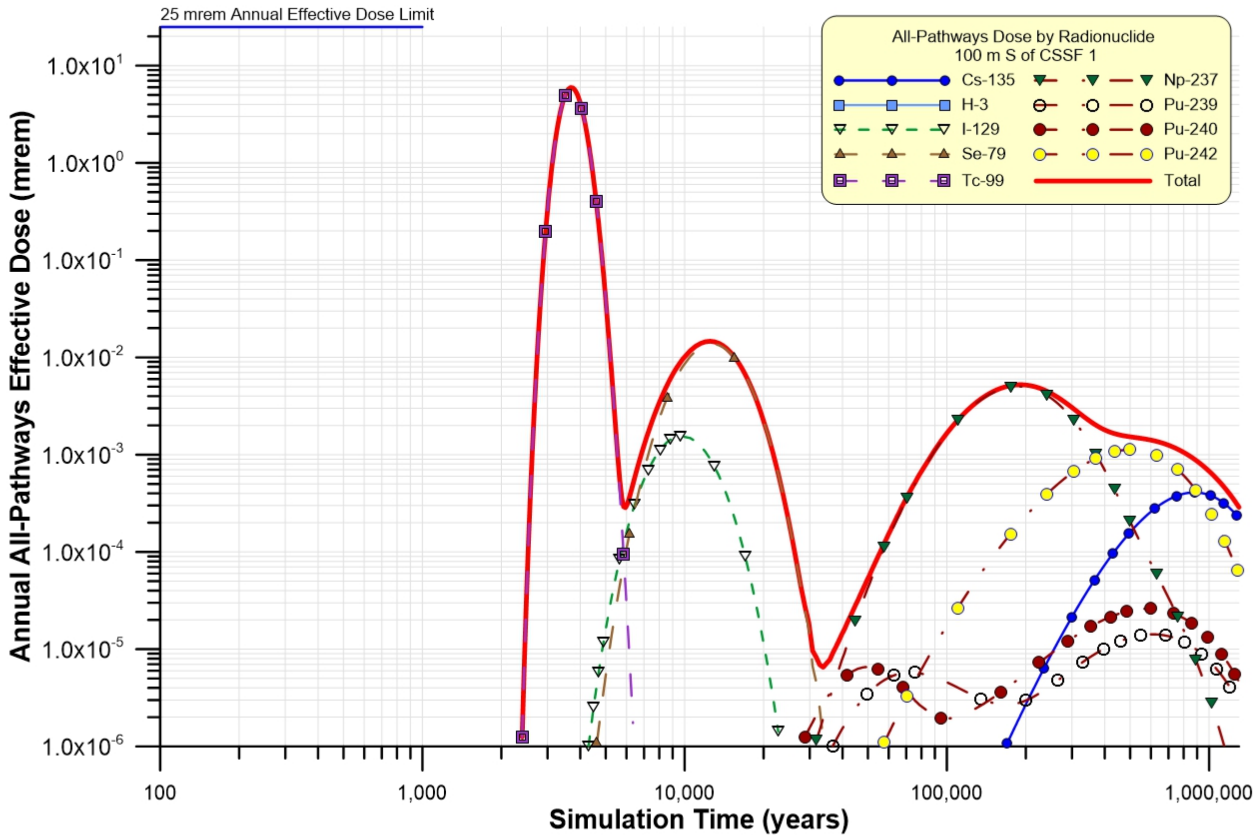

Figure 5-3. All-pathways effective dose as a function of time for OFAT Case 2: No containment provided by the stainless-steel bins.

- ***OFAT Case 3: 25.4 cm (10 in.) of Calcine Remains in Bins***—In this OFAT case, the amount of calcine remaining in the bins is assumed to be 25.4 cm (10 in.) instead of the 5.1 cm (2 in.) that is assumed for the base case. This increases the radionuclide inventory by a factor of 5. However, the grout in which the calcine is mixed also increases from 30.48 to 60.96 cm (1 to 2 ft) and represents two Mixing Cell Model cells.[48] The peak all-pathways dose increased from 1.91E-01 to 9.19E-01 mrem (Figure 5-4). The increase in dose is not a factor of 5 because the additional inventory is mixed in a larger volume. The leach rate is inversely proportional to the thickness of cells containing the inventory; thus, the overall increase in dose can be *approximated* by the ratio of the inventory increase (5) to cell thickness increase (2) or 5/2 = 2.5. The actual increase in dose is slightly more (~3.4).

  Leaving more calcine in the bins will increase the base-case dose, but even if significantly more calcine were left in the bins, the closed CSSF would still meet the 25-mrem annual dose performance objective. Assuming a linear response to the amount of residual calcine left behind, the base-case calcine residual could be increased from 5.1 cm (2 in.) (which results in a dose of 1.91E-01 mrem in a year) to 131 times as much calcine (approximately 6.68 m [22 ft] of calcine) in order for the base-case dose to be 25 mrem in a year.

  Related to this OFAT case is the sensitivity of the dose results to the average concentration in the calcine that was used to compute the residual inventory in the bins. This does not require a model simulation because the doses are directly proportional to the radionuclide concentrations in the calcine. Thus, if the calcine concentration is 2 times the base values for all radionuclides, then the resulting

48. The Mixing Cell Model is used to support exposure assessments and dose calculations for waste disposal or contaminated sites at the INL Site. Software used for the groundwater model is described in the CSSF PA/CA (DOE-ID 2022a).

doses are the base-case doses times 2. Staiger and Swenson (2021) reported Tc-99 concentrations in aluminum calcine (disposed of in CSSF 1) range from 1.5 to 2.5 Ci/m^3, whereas zirconium calcine (disposed of in CSSFs 3 through 5) range from 0.5 to 1.5 Ci/m^3. Thus, at the worst, Tc-99 concentrations would be a factor of 3 times higher than the base-case average in the residual calcine. If such were the case, then the doses would still be well below the 25-mrem annual all-pathways dose performance objective. Tc-99 is retained as an HRR as the most significant radionuclide as shown in Figure 5-4 for OFAT Case 3, even though the dose contribution from Tc-99 is insignificant.

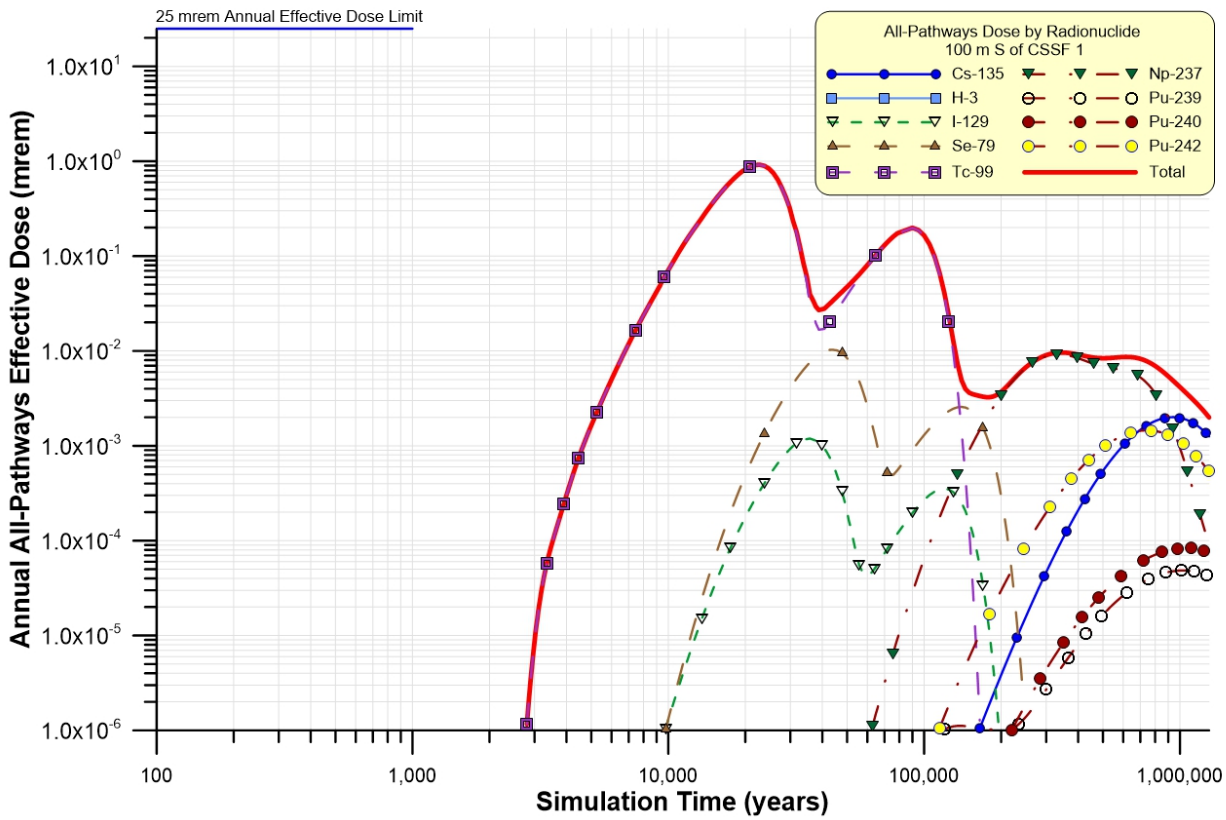

Figure 5-4. All-pathways effective dose as a function of time for OFAT Case 3: 25.4 cm (10 in.) of calcine remains in the bins.

- ***OFAT Case 4: Higher Infiltration as a Result of Climate Change***—One possible effect of climate change that would have a negative impact on the performance of the closed CSSF is higher precipitation. Higher precipitation does not translate directly to higher infiltration because higher precipitation will result in greater vegetative cover, resulting in more evapotranspiration. It is also likely that climate change will result in less precipitation and, combined with a warmer climate, will result in little downward water flux. Thus, this OFAT case is designed to illustrate the impact of an unlikely scenario with a significant increase in infiltration and does not consider the potential that precipitation could decrease as a result of climate change.

  For this OFAT case, it was assumed that the precipitation rate is much higher than the current rate, resulting in an increase of the natural infiltration from the base-case value of 1 to 10 cm/year (0.39 to 3.9 in./year). For this scenario to occur, a substantial increase in precipitation above the natural value of approximately 22 cm/year (8.7 in./year) would be required, because the increase in vegetation resulting from the higher precipitation will also lead to higher evapotranspiration rates (Yu et al. 2015). Based on paleo climatic data, droughts in the western United States were decreasing from 1,400 years before the present until recently when droughts have increased in the past 200 years (Marlon et al. 2011). Droughts were correlated with a rise in ambient temperature, and if the trend continues, it is unlikely in the next 1,000 years that higher precipitation will occur. Thus, an increase in the infiltration rate by

1 order of magnitude is extreme and is only intended to observe how the model will respond to a large increase in this parameter.

The peak annual all-pathways dose increased to 3.77E-01 mrem from the base case of 1.91E-01 mrem, and the peak time occurs at ~11,500 years instead of ~19,500 for the base case (Figure 5-5). The earlier peak times are a function of earlier peak fluxes from the concrete base (~19,000 for the base case and ~11,250 years for this OFAT for Tc-99) and shorter vadose zone transit times (~730 years for the base case and ~210 years for this OFAT for Tc-99).

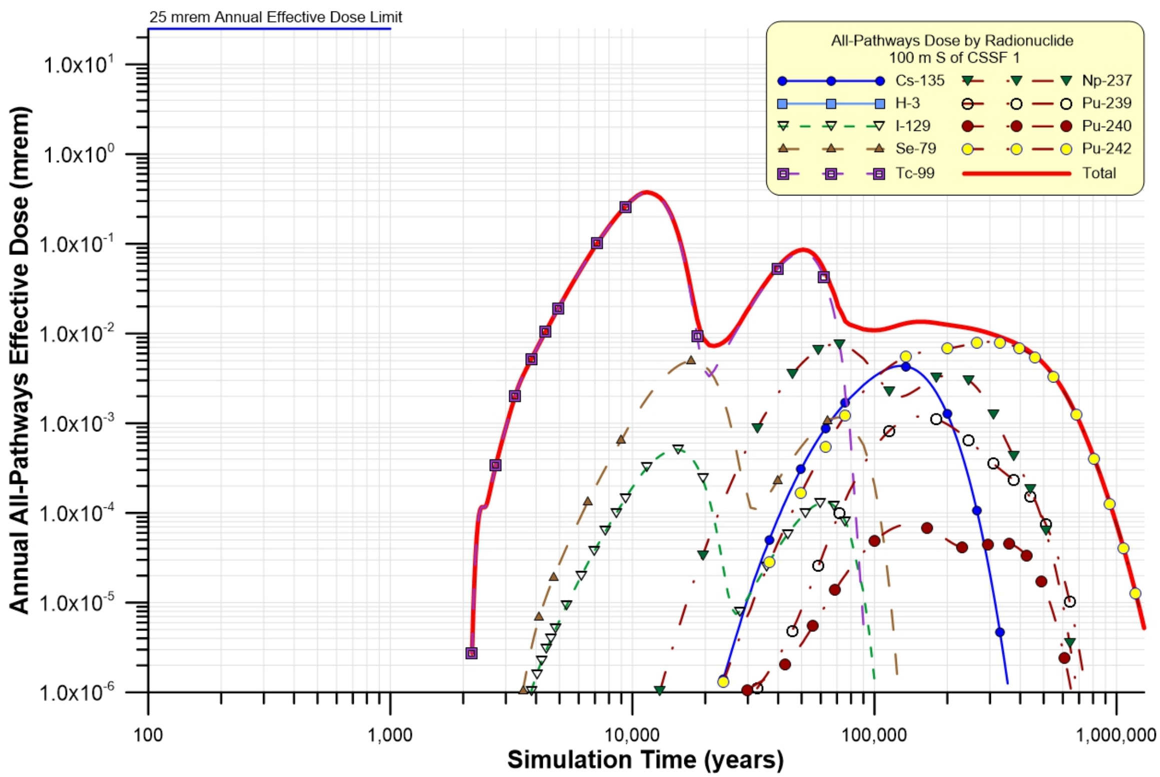

Figure 5-5. All-pathways effective dose as a function of time for OFAT Case 4: Higher infiltration as a result of climate change.

It could be expected that the higher infiltration would result in a proportional increase in the leaching of radionuclides and thus a proportional increase in the all-pathways dose. However, the grout and stainless-steel integrity are also factors to be considered. The grout is assumed to remain intact for 2,000 years and restricts water flux to that of the hydraulic conductivity of the intact grout. Thus, additional water infiltration does not increase the leaching of radionuclides while the grout remains intact. After the grout begins to fail and the stainless-steel bin starts to corrode (allowing more water to flow through the grouted waste layer), the radionuclide fluxes increase and more radionuclides are leached from the grouted waste layer. Because the stainless-steel and grout failure operate independently with respect to the water flux, the increase in infiltration does not directly translate to an increase in the all-pathways dose. If the water flux through the concrete base and stainless-steel bin were dependent on one another, then the all-pathways dose would be roughly proportional to the infiltration rate.

The small jog in the curve for Tc-99 around the 2,500-year mark reflects the time when the grout has fully failed and is subject to higher water fluxes, which flush any activity out of the concrete base. However, the stainless-steel bin has not sufficiently failed to replenish Tc-99 to the concrete base, and thus releases from the concrete base do not plateau until more failure occurs in the bins. A similar situation occurs in OFAT Case 5 (described below).

Even if the infiltration rate is increased to 10 cm/year (3.9 in./year), which is more than 40% of the average annual precipitation at the INL Site, the predicted peak dose is still 66 times smaller than the 25-mrem annual dose performance objective and occurs during the 1,000- to 10,000-year post-closure period. Tc-99 was retained as an HRR as the most significant radionuclide, as shown in Figure 5-5 for OFAT Case 4, even though the dose contribution is insignificant. All radionuclides shown in Figure 5-5 for OFAT Case 4 were included as HRRs (see Subsection 5.1.3).

- ***OFAT Case 5: No Credit for Beneficial Geochemical Conditions of Grout***—Generally, but not in all cases, grout provides a geochemical environment that results in higher soil-to-water partition coefficients (K_ds). This OFAT case provides a bounding estimate of the doses that does not account for the beneficial properties of the grout. For this OFAT case, if the grout K_d was greater than the vadose zone (sediment) K_d, then the K_d for the sediment was substituted. If the sediment K_d was smaller, then the sediment K_d was also applied to the concrete base.

  Results of this OFAT analysis showed that the peak annual dose was 3.38E-01 mrem and occurs ~13,000 years after CSSF closure compared to a base-case annual dose of 1.91E-01 mrem occurring at ~19,500 years (see Figure 5-6). Like the base case, Tc-99 is the dominant dose contributor, and the difference between the grout K_d and sediment K_d was 1 order of magnitude. The difference in the peak time relates to the time of peak flux from the concrete base. The Tc-99 peak flux from the concrete base for this OFAT occurred at ~12,000 years, whereas the peak flux for the base case occurred in ~19,000 years. The base-case vadose zone transit time for Tc-99, which would apply to this OFAT case, was ~730 years.

  Like OFAT Case 4, which increased the infiltration rate 1 order of magnitude higher than the base case, it might be expected that the peak doses would go up 1 order of magnitude because the technetium K_d decreased 1 order of magnitude. However, this was not the case, and peak doses were roughly the same as OFAT Case 4 and exhibit the same jog in the dose curve but at ~3,000 years instead of the ~2,500 years in OFAT Case 4. This is because water fluxes through the stainless-steel bins and the concrete base are independent of one another. Tc-99 was retained as an HRR as the most significant radionuclide, as shown in Figure 5-6 for OFAT Case 5, even though the dose contribution is insignificant. All radionuclides shown in Figure 5-6 for OFAT Case 5 were included as HRRs (see Subsection 5.1.3).

- ***OFAT Case 6: Sensitivity of the Grouted Waste Layer Thickness***—The base case assumed residual calcine was mixed in 0.3048 m (1 ft) of grout (i.e., the grouted waste layer). In this OFAT case, the grouted waste layer is pessimistically assumed to be only 0.1016 m (4 in.) thick. Under steady-state flow conditions, a thinner grouted waste layer will result in proportionally higher radionuclide fluxes from the grouted waste layer because the flux is inversely proportional to the thickness of the grouted waste layer. However, the bin set model is not a steady-state flow condition because the integrity of the stainless-steel bins and grout will restrict water flow through the grouted waste layer for at least 2,000 years. After that time, water fluxes slowly increase over thousands of years. During the time when water fluxes are very low, radionuclides diffuse in the pore water though the clean grout lying on top of the grouted waste layer, resulting in a spreading of the activity throughout the grout. Additionally, there is a slow loss of inventory from the grouted waste layer, regardless of its thickness, when the water flux is low and the corresponding radionuclide fluxes are low. Consequently, there is no proportional increase in the *peak* radionuclide flux from the grouted waste layer.

  To test the sensitivity of the model to this parameter, the grout in CSSF 1 was discretized into fifteen 0.1016-m (4-in.) layers, and the entire Tc-99 inventory was assigned to the lowest layer. The peak flux was 4.2E-04 Ci/year, whereas the base-case peak flux was 3.1E-04 Ci/year. Thus, a decrease in the grouted waste layer thickness by a factor of 3 resulted in an increase in the peak flux by a factor of 1.35. For the reasons stated earlier, it is unlikely that the entire inventory of residual calcine will remain on the bottom of the bins. Nevertheless, if such a condition does occur, it will not result in a drastic increase of peak fluxes and corresponding peak doses. Thus, annual all-pathways doses are not very sensitive to the reduction in waste layer thickness. No additional HRR was identified based on OFAT Case 6. All radionuclides shown in Figure 5-6 for OFAT Case 6 were included as HRRs (see Subsection 5.1.3).

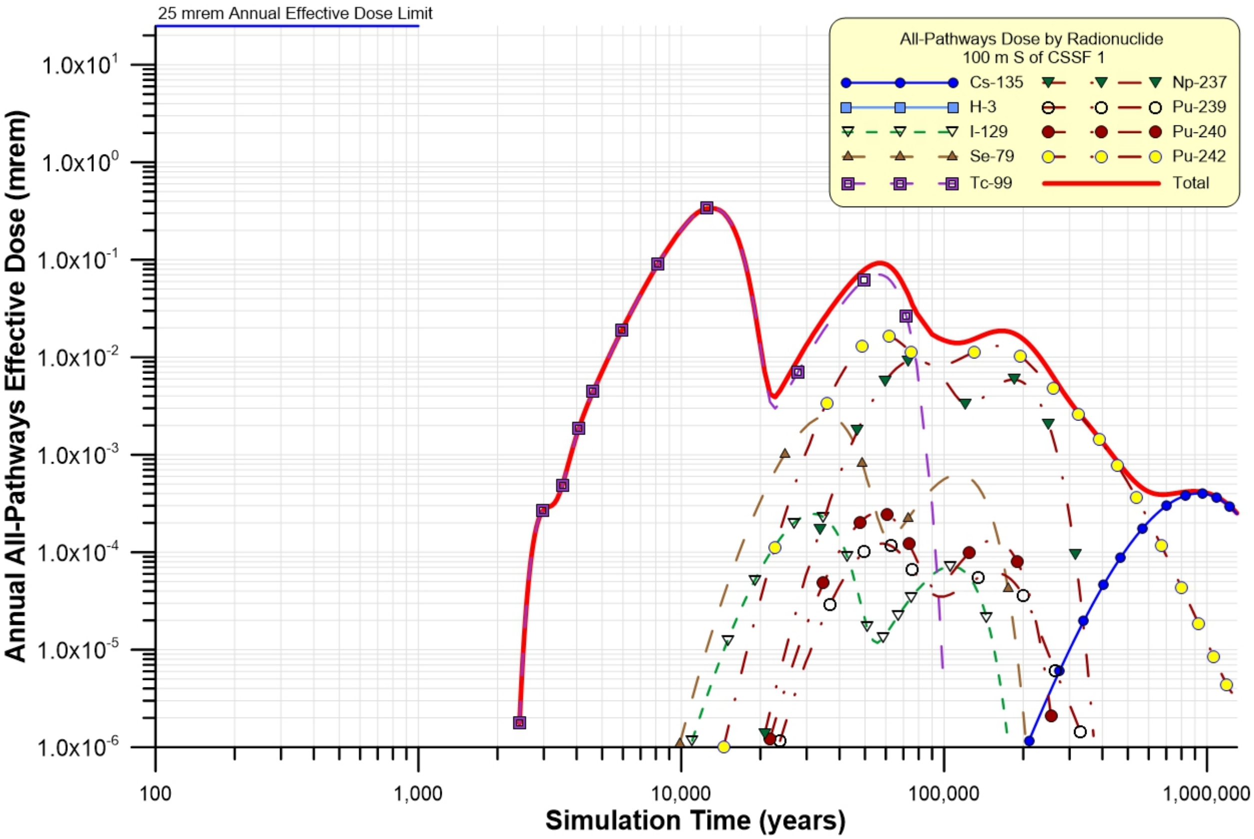

Figure 5-6. All-pathways effective dose as a function of time for OFAT Case 5: No credit for beneficial geochemical conditions of grout.

5.1.1.2 Air All-Pathways Dose

A summary of the results for air all-pathways dose analysis is presented in Section 7.1.4. Details of the air all-pathways dose analysis are provided in the CSSF PA/CA (DOE-ID 2022a). Annual all-pathways EDs from atmospheric releases from the CSSF were all substantially below the 10-mrem performance objective in DOE M 435.1-1 Chg 3 (Table 5-3) and are insignificant contributors to dose. The annual ED from the air pathway for an MEI located off the INL Site during the 100-year period of assumed institutional control[49] was 1.35E-04 mrem. After the 100-year institutional control period and during the 1,000-year post-closure period, the maximum dose was 6.66E-06 mrem at the MEI located 100 m (328 ft) from the CSSF (referred to hereinafter as the 100-m receptor), which is substantially less than the 10-mrem performance objective.

49. In the CSSF PA/CA, the analysis assumed a 100-year institutional control period. See Footnote 34 for details.

The atmospheric pathway annual all-pathways ED did not consider the contribution from a biotic intruder. Biointrusion into the CSSF is not expected due to the depth of the residual waste. The residual waste in the bins is a minimum of 13.7 m (45 ft) below the ground surface, which is much greater than the depth of roots and animal burrows at the INL Site (i.e., a maximum of 2.25 and 2.70 m [89 and 106 in.], respectively). Details of the CSSF biointrusion pathway screening analysis are documented in EDF-11173, "Biotic Pathway Screening Analysis for the CSSF Performance Assessment."

Because the CSSF PA/CA atmospheric pathway dose analysis projected insignificant doses, no HRR was identified from the air pathway dose analysis.

Table 5-3. Maximum annual all-pathways effective dose for a maximally exposed individual located off the Idaho National Laboratory Site and a 100-m (328-ft) receptor for atmospheric releases from the Calcined Solids Storage Facility.

Radionuclide	Maximum Air Annual All-Pathways Effective Dose during the 100-yr Institutional Control Period for an MEI Located Off the INL Site (mrem)[a]	Maximum Air Annual All-Pathways Effective Dose during the First 1,000 Years of the Post-Closure Period at 100-m Receptor (mrem)
H-3	1.35E-04	3.10E-07
C-14	<1E-10	4.79E-08
I-129	1.41E-09	6.30E-06
Total	1.35E-04	6.66E-06

a. Doses presented as less than 1E-10 are essentially zero.

INL Idaho National Laboratory
MEI maximally exposed individual

5.1.1.3 Hypothetical Inadvertent Intruder Pathway Dose

Results of the CSSF PA/CA inadvertent intruder pathway dose analysis are summarized in Section 6 of this Draft CSSF 3116 Basis Document. Details of the inadvertent intruder pathway analysis are provided in the CSSF PA/CA (DOE-ID 2022a) and EDF-11132. The CSSF PA/CA hypothetical inadvertent intruder pathway analysis assumed two stylized scenarios at the INL Site: an acute intruder drilling scenario and a chronic intruder post-drilling agricultural scenario.

The acute intruder drilling scenario yielded a peak total ED of 7.1E+00 mrem 500 years post-closure for the CSSF bins. The acute intruder drilling scenario yielded a peak total ED of 5.6E-02 mrem 500 years post-closure for the CSSF transport lines. Am-241, Pu-239, Pu-240, Pu-238, Nb-94, Pu-241, Sn-126, and Np-237 provide most of the intruder doses for the bin sets and transport lines at 500 years post-closure for the CSSF 4 residual waste profile (see Tables 5-4 and 5-5, respectively).

The chronic intruder post-drilling agriculture scenario yielded a peak annual dose of 3.6E+00 mrem 500 years post-closure for the CSSF bins. The chronic intruder post-drilling agriculture scenario yielded a peak annual dose of 6.2E-02 mrem 500 years post-closure for the CSSF transport lines. Tc-99, Sn-126, Cs-137, Sr-90, Am-241, Pu-239, Pu-240, and Np-237 provide most of the contributions to the intruder doses for the bin sets, based on the CSSF 1 waste profile (see Table 5-6). Tc-99, Nb-94, Sn-126, Am-241, Pu-239, Pu-240, Pu-238, Cs-137, Np-237, Sr-90, and Pu-241 provide most of the contributions to the intruder doses for the transport lines based on the CSSF 5 waste profile (see Table 5-7). It is noted that Cs-137 and Sr-90 continue to provide significant doses in the chronic intruder scenarios even after 500 years. These doses are due to the initial inventories for Cs-137 (7.6E+03 Ci) and Sr-90 (7.0E+03 Ci) being the largest at the CSSF such that after 500 years of decay, they provide sufficient inventories (8.0E-02 Ci for Cs-137 and 3.6E-02 for Sr-90) to result in chronic intruder dose contributions.

No radionuclides listed in Tables 5-4 through 5-7 for the intruder analysis were included in the list of HRRs (see Subsection 5.1.3) because their contribution to dose is insignificant.

Table 5-4. Radionuclide total effective dose results for the acute intruder drilling scenario at the Calcined Solids Storage Facility bins.

Radionuclide	Total Effective Dose[a] (mrem)
Am-241	2.1E+00
Pu-239	1.6E+00
Pu-238	1.0E+00
Pu-240	9.9E-01
Nb-94	5.2E-01
Pu-241	3.7E-01
Sn-126	2.8E-01
Np-237	8.3E-02
Cs-137	7.4E-02
Total (all nuclides)	**7.1E+00**

a. Doses reported for the Calcined Solids Storage Facility 4 waste profile, which provides the maximum intruder doses for this scenario.

Table 5-5. Radionuclide total effective dose results for the acute intruder drilling scenario at the Calcined Solids Storage Facility transport lines.

Radionuclide	Total Effective Dose[a] (mrem)
Am-241	1.7E-02
Pu-239	1.3E-02
Pu-238	8.1E-03
Pu-240	7.9E-03
Nb-94	4.2E-03
Pu-241	3.0E-03
Sn-126	2.2E-03
Np-237	6.6E-04
Cs-137	5.9E-04
Total (all nuclides)	**5.6E-02**

a. Doses reported for the CSSF 4 waste profile, which provides the maximum intruder doses for this scenario.

Table 5-6. Radionuclide annual effective dose results for the chronic intruder post-drilling agriculture scenario at the Calcined Solids Storage Facility bins.

Radionuclide	Total Effective Dose[a] (mrem)
Tc-99	2.8E+00
Sn-126	4.8E-01
Cs-137	1.1E-01
Sr-90	7.0E-02
Am-241	4.0E-02
Pu-239	2.9E-02
Pu-240	1.1E-02
Np-237	9.0E-03
Pu-238	3.8E-03
Total (all nuclides)	**3.6E+00**

a. Doses reported for the Calcined Solids Storage Facility 1 waste profile, which provides the maximum intruder doses for this scenario.

Table 5-7. Radionuclide annual effective dose results for the chronic intruder post-drilling agriculture scenario at the Calcined Solids Storage Facility transport lines.

Radionuclide	Total Effective Dose[a] (mrem)
Tc-99	3.0E-02
Nb-94	1.0E-02
Sn-126	5.4E-03
Am-241	4.5E-03
Pu-239	3.0E-03
Pu-240	2.0E-03
Pu-238	1.9E-03
Cs-137	1.5E-03
Np-237	1.5E-03
Sr-90	1.1E-03
Pu-241	9.2E-04
Total (all nuclides)	**6.2E-02**

a. Doses reported for the CSSF 5 waste profile, which provides the maximum intruder doses for this scenario.

5.1.2 Highly Radioactive Radionuclides Retained from 10 CFR 61.55

Radionuclides and their associated concentration limits are specified in two separate tables within 10 CFR 61.55. Those tables are reproduced here as Tables 5-8 and 5-9. All radionuclides identified in Tables 1 and 2 of 10 CFR 61.55 that are present in the calcine are included as HRRs.

Table 5-8. Waste classification concentration limits from Table 1 of 10 CFR 61.55.

Radionuclides (long lived)[a]	Concentration
C-14	8 Ci/m^3
C-14 in activated metal	80 Ci/m^3
Ni-59 in activated metal	220 Ci/m^3
Nb-94 in activated metal	0.2 Ci/m^3
Tc-99	3 Ci/m^3
I-129	0.08 Ci/m^3
Alpha-emitting transuranic nuclides with half-life >5 yr	100 nCi/g
Pu-241	3,500 nCi/g
Cm-242	20,000 nCi/g

a. Classification is determined by long-lived radionuclides. If radioactive waste contains only radionuclides listed in this table, then classification shall be determined based on the concentrations shown in Table 5-9 as follows:
(i) If the concentration does not exceed 0.1 times the value in this table, then the waste is Class A.
(ii) If the concentration exceeds 0.1 times the value but does not exceed the value in this table, then the waste is Class C.
(iii) If the concentration exceeds the value in this table, the waste is not generally acceptable for near-surface disposal.

Table 5-9. Waste classification concentration limits from Table 2 of 10 CFR 61.55.

Radionuclides (short lived)	Concentration (Ci/m^3)		
	Column 1 (Class A)	Column 2 (Class B)	Column 3 (Class C)
Total of all nuclides with <5-yr half-life	700	a	a
H-3	40	a	a
Co-60	700	a	a
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7,000
Sr-90	0.04	150	7,000
Cs-137	1	44	4,600

a. There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in the table determine the waste to be Class C independent of these nuclides.

5.1.3 Summary of Highly Radioactive Radionuclides Identified for the CSSF

HRRs were identified from the CSSF PA/CA (DOE-ID 2022) considering results for the all-pathways dose analysis (i.e., groundwater and atmospheric pathways) and the hypothetical inadvertent intruder analyses (acute drilling scenario and chronic post-drilling agricultural scenario). HRRs were also identified from Tables 1 and 2 of 10 CFR 61.55. The HRRs for the NDAA Section 3116(a) analysis are provided in Table 5-10.[50] As discussed in the next subsection, approximately 99% or more of the total radionuclide inventory (volume) and approximately 99% of the attributable to HRRs, will be removed from the CSSF.

5.1 Removal of Highly Radioactive Radionuclides to the Maximum Extent Practical

NDAA Section 3116(a) provides that certain waste resulting from SNF reprocessing is not HLW if the Secretary of Energy, in consultation with the NRC, determines, among other things, that the waste has had HRRs removed "to the maximum extent practical."[51]

Removal to the maximum extent practical is not removal to the extent "theoretically possible." Rather, a "practical" approach to removal is one that is "adapted to actual conditions" (Fowler 1930); "adapted or designed for actual use" (Dictionary.com 2021); "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" (Dictionary.com 2021); fitted to "the needs of a particular situation in a helpful way...effective or suitable" (Cambridge University Press 2018). Therefore, the determination as to whether a specific HRR will be removed to the maximum extent practical will vary from situation to situation, based not only on available technologies but also on the overall costs and benefits[52] of deploying a technology for a particular waste stream. The "maximum extent practical" standard contemplates, among other things: consideration of expert judgment and opinion; environmental, health, timing, or other exigencies; the risks and benefits to public health, safety, and the environment arising from further radionuclide removal as compared with countervailing considerations that may ensue from not removing or delaying removal; life-cycle costs; net social value; the cost (monetary as well as environmental and human health and safety costs) per curie removed; radiological removal efficiency; the point at which removal costs increase significantly in relationship to removal efficiency; the service life of equipment; the reasonable availability of proven technologies; the limitations of such technologies; the usefulness of such technologies; project schedule or funding constraints; and the sensibleness of using such technologies. What may be removal to the maximum extent practical in one situation or at one point in time may not be that which, on balance, is practical, feasible, or sensible in another situation or at a prior or later point in time.

50. Some of the radionuclides listed as HRRs in this Draft CSSF 3116 Basis Document may not be listed in other NDAA Section 3116 basis documents if such radionuclides are not present in the waste or do not contribute significantly to dose to the worker, the public, or the inadvertent intruder.

51. NDAA Section 3116 does not specify "remedial goals" or other numerical objectives and does not require DOE to develop any such removal goals or objectives.

52. While DOE M 435.1-1 Chg 3 requires removal "to the maximum extent technically and economically practical," NDAA Section 3116 omits these adverbs, thereby suggesting that a broad range of considerations, including but not limited to technical and economic practicalities, may appropriately be taken into account in determining the extent of removal that is practical.

Table 5-10. Calcined Solids Storage Facility highly radioactive radionuclides for the NDAA Section 3116(a) analysis.[53]

Radionuclide	Performance Assessment Pathway[a]	10 CFR 61.55 Table
Am-241	Inadvertent intruder	Table 1
C-14	Insignificant to dose from air	Table 1
Co-60	Screened from further analysis in the CSSF PA/CA	Table 2
Cs-137	Insignificant to dose from inadvertent intruder	Table 2
H-3	Insignificant to dose from air	Table 2
I-129	Insignificant to dose from air	Table 1
Ni-63	Screened from further analysis in the CSSF PA/CA	Table 2
Np-237	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-238	Insignificant to dose for inadvertent intruder	Table 1
Pu-239	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-240	Insignificant to dose from groundwater and inadvertent intruder	Table 1
Pu-241	Insignificant to dose for inadvertent intruder	Table 1
Pu-242	Insignificant to dose from groundwater	Table 1
Sr-90	Insignificant to dose for inadvertent intruder	Table 2
Tc-99	Identified as an HRR from groundwater; insignificant dose for inadvertent intruder	Table 1
Alpha-emitting transuranic nuclides with half-life >5 yr[b]	Radionuclides not listed above were screened from further analysis in the CSSF PA/CA and are listed in Footnote b.	Table 1

a. Tc-99 is the only radionuclide that potentially may be a significant contributor to the very low doses to a member of the public or the hypothetical human intruder. All other HRRs identified under this column are identified as HRRs from Table 1 or 2 of 10 CFR 61.55.

b. Additional nuclides not already identified in the PA analysis as HRRs include Am-242m, Am-243, Cf-249, Cf-250, Cf-251, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, and Pu-244.

— Radionuclide is not included in either Table 1 or 2 of 10 CFR 61.55.
CA composite analysis
CSSF Calcined Solids Storage Facility
HRR highly radioactive radionuclide
PA performance analysis

53. Radionuclides listed as insignificant to dose are based on the 10% of the dose criteria as described in detail in Footnote 43.

Additionally, it may not be practical to undertake further removal of certain radionuclides because further removal is not sensible or useful when considering costs and risk compared to the overall benefit to human health or the environment. As a general matter, such a situation may arise if certain radionuclides are present in such extremely low quantities that they make an insignificant contribution[54] to potential doses to workers, the public, and the hypothetical human intruder.

Removal of HRRs to the maximum extent practical at the CSSF will occur through the use of a pneumatic retrieval system, including the use of an air lance to move residual waste to the vacuum system, as described in the following subsections. These technologies have been proven to be effective in removing calcine surrogate to a point where further removal of HRRs is not sensible or useful in light of the overall benefit to human health, safety, and the environment.

5.1.1 Selection of Waste Retrieval Technology

DOE has evaluated several different retrieval concepts to identify the best approach and available technologies to remove calcine from the CSSF. Waste retrieval operations will require modifications to the CSSF and installation of equipment to remove calcine to the maximum extent practical because retrieval mechanisms, other than access risers in CSSFs 2 through 6, were not incorporated into the facility designs. To accomplish waste removal, a pneumatic retrieval and transfer system will be used. A pneumatic retrieval and transfer system has served as the basis for most DOE studies and tests regarding calcine removal from the CSSF, and DOE has concluded that this is the most viable approach to remove calcine (and the HRRs therein) to the maximum extent practical (ICP 2017). The *Calcine Retrieval Project—Waste Removal Technology Selection Report* (ICP 2017) summarizes the different studies and technologies that have been evaluated by DOE over the past 28 years and explains the reasons for selecting a pneumatic retrieval and transfer system as the preferred method.

DOE determined that a pneumatic retrieval and transfer system is the most cost-effective, reliable, and safest approach for removal of calcine from the CSSF (ICP 2017). Use of a pneumatic system is well understood after years of WCF and NWCF operations, which successfully used a pneumatic system to transfer calcine to the CSSF. A pneumatic system is also well suited for calcine removal due to the following: (1) the calcine is a unique waste form (i.e., a highly radioactive, dry, granular solid); (2) the CSSF is not designed to receive, contain, or be decontaminated with liquid materials; and (3) the CSSF bins are mostly filled to capacity, limiting in situ treatment or fluid dissolution to remove calcine as a liquid slurry because there is not enough void space (ICP 2017). DOE also considered methods such as sluicing, mixing, chemical cleaning, mechanical manipulators, and robotic vehicles, but most of these methods were eliminated for calcine removal due to safety risks, equipment reliability, or increased waste volume (ICP 2017). Mechanical equipment, such as an auger system, for example, was eliminated because such methods generally include numerous moving parts that introduce reliability issues and difficulties with ensuring confinement of the calcine. A sluicing or mixing approach was eliminated

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54. DOE normally would view radionuclides as making an insignificant contribution if the contribution to dose from those radionuclides, in both the expected case and considering sensitivity analyses, does not exceed any of the following: (1) 10% of the 25-mrem annual all-pathways dose to the public, (2) 10% of the DOE 100-mrem annual dose limit to the intruder (under all reasonable intruder scenarios), (3) 10% of the DOE 500-mrem acute dose limit to the intruder (under all intruder scenarios), and (4) 10% of the annual worker dose in the relevant provisions of 10 CFR 20, "Standards for Protection Against Radiation." This methodology is based on NRC consultation and is intended to be consistent with the guidance and general approach in NUREG-1757, Volume 2, *Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria Final Report* (NRC 2006), which explains that "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10% of the dose criteria to be insignificant contributors." DOE has previously used 5% of the 25-mrem/year all-pathways dose limit (i.e., 1.25 mrem/year) to ensure the selection of HRRs is conservative (DOE 2014); however, for this Draft 3116 Basis Document, 10% of the all-pathways dose limit will be used to be consistent with NRC guidance as further conservatism does not impact the selection of HRRs identified for the CSSF. NUREG-1757, which applies to NRC licensees, is being used only as general guidance, and DOE's use of the document as guidance should not be construed to suggest that it is a requirement under NDAA Section 3116 or that either NUREG-1757 or 10 CFR 20, Subpart E is applicable to DOE.

because it would generate large volumes of waste that is not as stable as calcine and would require storage, treatment, and disposal (ICP 2017). Thus, based on the above considerations, DOE has proceeded with developing and testing a pneumatic retrieval and transfer system.

As described above, a pneumatic system has served as the basis for most historical studies and tests, and vacuum extraction and pneumatic transfer make up the preferred method that will meet the criteria to remove calcine (and the HRRs therein) to the maximum extent practical. After DOE completed its AoA Summary Report (DOE-EM 2016), work proceeded to develop and test a full-scale retrieval system. The full-scale retrieval system has been built, and DOE has completed retrieval and transfer designs originally described in the *Calcine Retrieval Project Conceptual Design for the Transfer of Calcined Solids from CSSF 1 to CSSF 6* (ICP 2016). Testing objectives were to eliminate risks, optimize final design configurations, and verify the efficacy of calcine removal using a pneumatic retrieval system for bulk and residual cleanout. This testing was critical, not only to show that retrieval operations will meet the maximum-extent-practical criteria in NDAA Section 3116(a), but also to (1) establish criteria for ending calcine retrieval activities (i.e., determine when the maximum extent practical has been achieved), (2) evaluate cost and benefits associated with performing additional retrieval, and (3) ensure standards under other requirements, such as HWMA/RCRA closure performance standards and CERCLA remedial action objectives, are met. Information collected during the testing phase will also be used to support environmental closure.

Three primary components of the retrieval system have been designed, constructed, and tested:

- A full-scale mockup that proves the retrieval and transfer system design capabilities, verify operating parameters, and verifies calcine retrieval efficacy.
- Retrieval systems that will be used to remove calcine from the CSSF.
- A video monitoring system that will be used to monitor, evaluate, and document design testing. This system also will be used as a remote visual aid during actual retrieval operations to optimize technology deployment, maximize retrieval results, identify lessons for future deployments, and verify waste removal at the end of retrieval operations.

The following subsections describe three primary components of the calcine retrieval and transfer. Though the equipment described in the following subsections is specific to the transfer of calcine from CSSF 1 to CSSF 6, the designs and operating parameters are applicable to future retrieval operations to remove calcine from CSSFs 2 through 6.

5.1.1.1 Full-Scale Mockup

A full-scale mockup has been constructed to simulate pneumatic retrieval and transfer of CSSF calcine. $CaCO_3$, which has been determined to be the closest material to calcine in bulk density and particle size and distribution, has been used as a surrogate for the testing (Sandow 2019; Lower 2016). The full-scale mockup simulates transfer of calcine from CSSF 1 to CSSF 6 to demonstrate the capability to safely retrieve and transfer the stored calcine.[55] The full-scale mockup includes a replica of a CSSF 1 nested bin, transport piping, blind tees and crosses, cyclone separator, pre-filter, high-efficiency particulate air filter, off-gas blower/compressor, compressed air supply system, air compressor, compressor air dryer, air receiver storage vessel, and all other appurtenances needed for operation of the mockup. After initial retrieval testing at a commercial location, the bin replica was transported to, and installed at, the former Fuel Processing Restoration Facility (CPP-691) at INTEC (see Figure 5-7). Full system testing began in 2019 and has resulted in upgrades and modifications to the system to improve its performance, as described in Sandow (2021). Figure 5-8 shows the equipment that will be installed on the storage vault roof of CSSF 1 (after removal of the instrument room [CPP-639] and cyclone vault [CPP-729]). The full-

55. While testing was focused on retrieval and transfer of calcine from CSSF 1 to CSSF 6, the pneumatic retrieval processes are applicable to each CSSF and calcine can be transferred elsewhere, if needed.

scale mockup simulates the actual system that will be used to retrieve waste from CSSF 1 and transfer it to CSSF 6 or, if needed, another location.

5.1.1.2 Bulk Retrieval System

The bulk retrieval system uses a pipe-in-pipe vacuum and compressed air system to retrieve calcine from the bottom of a bin (ICP 2020, 2021b). A bottom-up retrieval system removes the need to add pipe sections during operation or have a hose-management system. Additionally, the pipe-in-pipe design reduces the area for the vacuum portion of the vertical line, which increases velocity and aids in the transport of solids (ICP 2017). The main control system for the bottom-up bulk retrieval system works by adjusting the fluidizing air tube up and down with a linear slide, which in turn moves the nozzle that delivers the fluidizing air to the bottom of the vacuum (ICP 2020, 2021b). In addition to the pipe-in-pipe vacuum system, an air lance to push residual calcine off the internal surface has been designed and tested (ICP 2020, 2021b). The purpose of the air lance is to create a circular wind to help agitate material off the surface and direct it toward the bulk retrieval system. Figure 5-9 shows the full-scale mockup of the bottom-up retrieval unit and transport manifold manufactured and installed at CPP-691. The equipment shown in Figures 5-8 and 5-9 will be installed on the storage vault roof of CSSF 1 (after removal of the instrument room [CPP-639] and cyclone vault[56] [CPP-729] [see Figure 2-28]). Historical retrieval studies and tests have demonstrated that less than 1% of the total calcine is estimated to remain after bulk retrieval using a vacuum system (ENICO 1981; Westra 1982; Griffith 1996; AEA Technology 2006) (see Subsection 2.11.3.3). Initial testing of the pipe-in-pipe bulk retrieval system and air lance indicates they will perform equally as well (ICP 2020, 2021b; Sandow 2021; M-1574).

5.1.1.3 Residual Cleanout System

It is expected that the bulk retrieval system will remove most of the calcine (~99% by volume and curies, see Subsection 2.11.3.3) (ENICO 1981; Westra 1982; Griffith 1996; AEA Technology 2006; Sandow 2021). A small amount (~1% by volume and curies) of calcine may remain on the stiffening rings, floor, and other areas of the bins. DOE has developed technologies to further remove remaining calcine using a residual cleanout system. These additional technologies will only be deployed if necessary to meet reduced residual waste volumes.

5.1.1.4 Video Monitoring System

The video monitoring system has been used extensively during testing and will be used during actual operations to monitor, evaluate, and document retrieval activities. Data collected will allow engineers to: (1) easily identify problem areas and system improvements during testing and actual retrieval operations, (2) reduce the need for individuals to access high-radiation fields during operations, and (3) provide verification of calcine retrieval effectiveness for environmental closure. The video monitoring system is network based and will use a combination of pan-tilt-zoom and fixed-position cameras. Ethernet cables connect system components and provide camera power, camera control, and image data transfer per camera location. The system is easily expanded, and any laptop computer running a common operating system can connect to the local area network and serve as a camera-control and viewing station (Young 2019; ICP 2020, 2021b). Figure 5-10 shows still images collected from inside the CSSF 1 bin replica during retrieval testing.

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56. Additional details on the strategy to remove the above grade structures and install the retrieval equipment are provided in PLN-6657, "CSSF 1 Deactivation and Decommissioning Strategy," and will be detailed in DOE M 435.1-1 Chg 3 Tier 2 closure documentation.

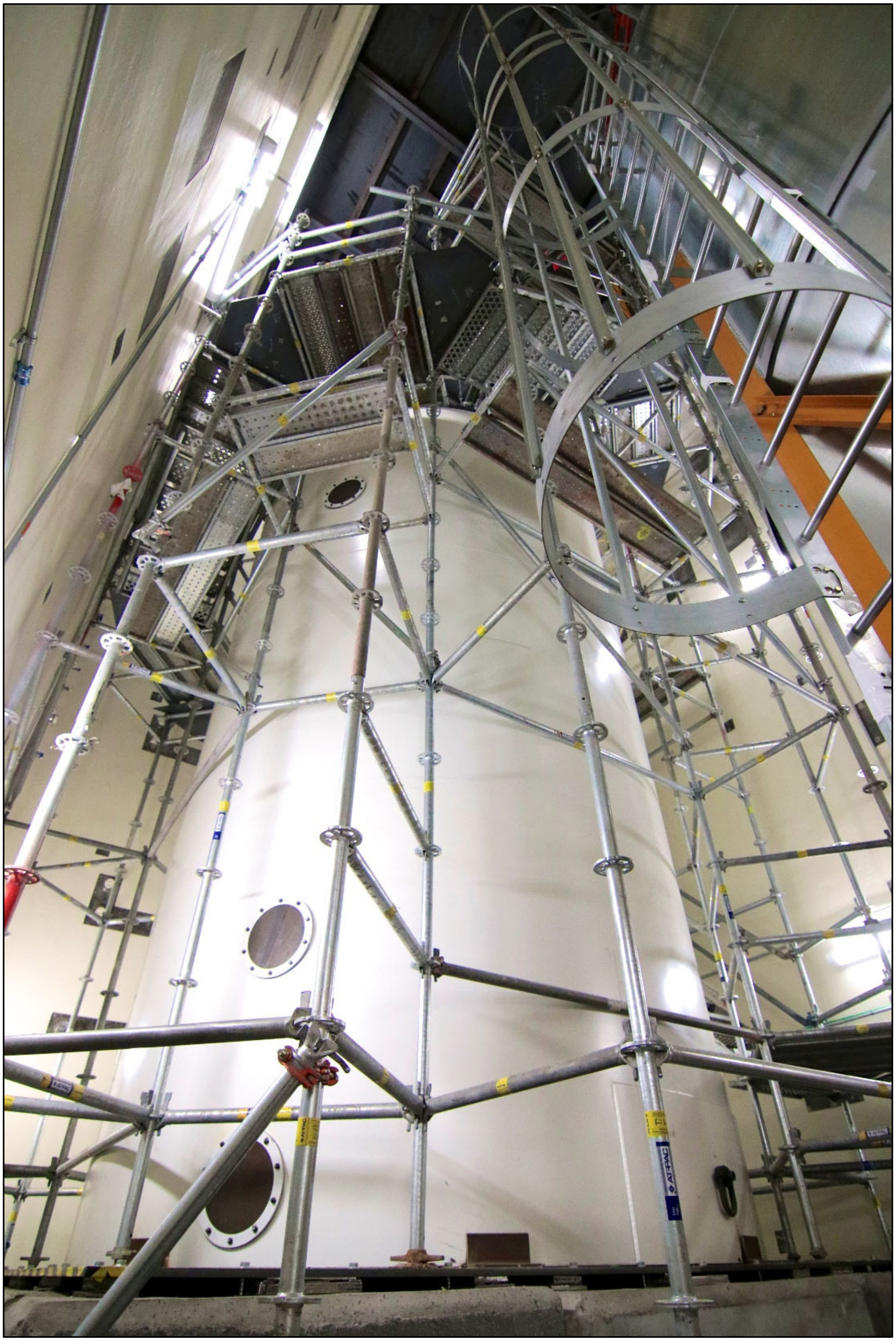

Figure 5-7. Placement of the Calcined Solids Storage Facility 1 nested bin replica in a belowgrade cell at the former Fuel Processing Restoration Facility (CPP-691).

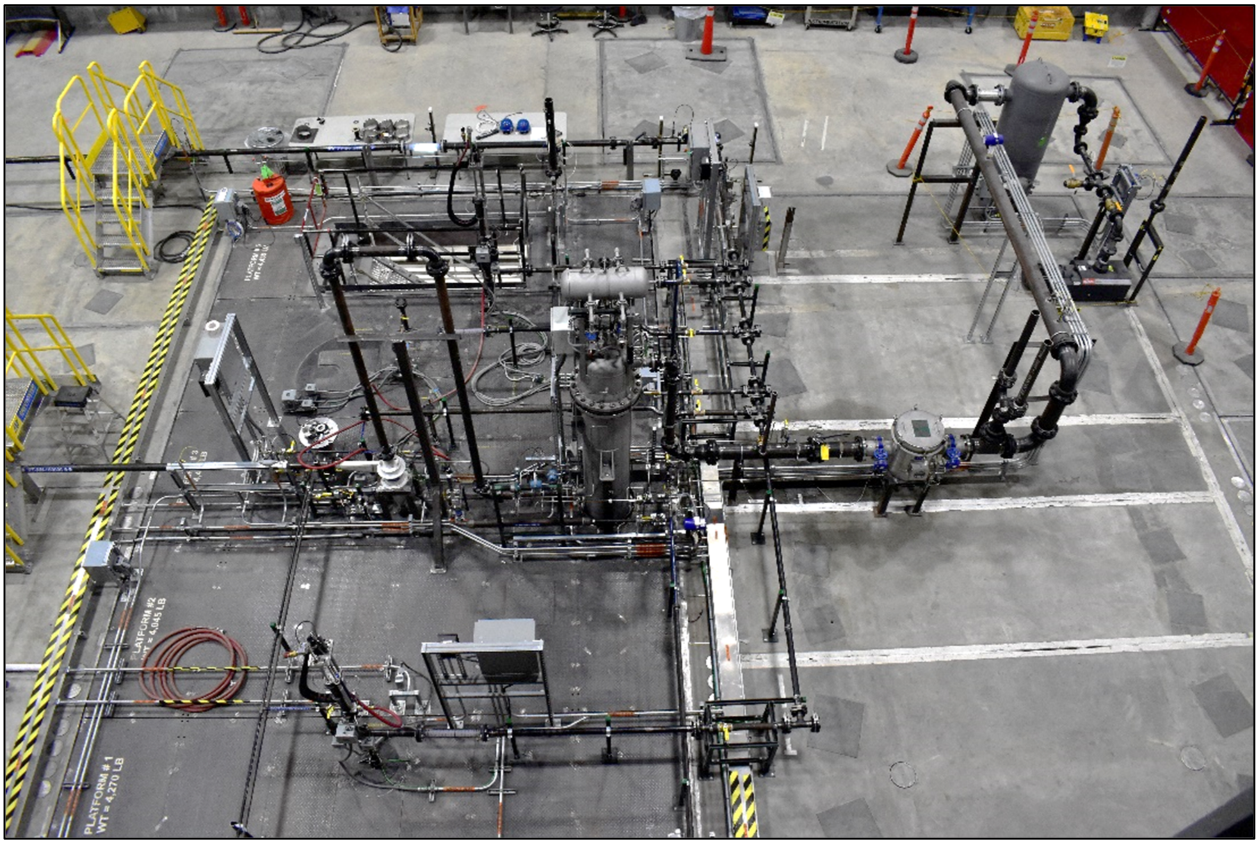

Figure 5-8. Configuration of full-scale mockup retrieval and transport system at the former Fuel Processing Restoration Facility (CPP-691) in 2021.

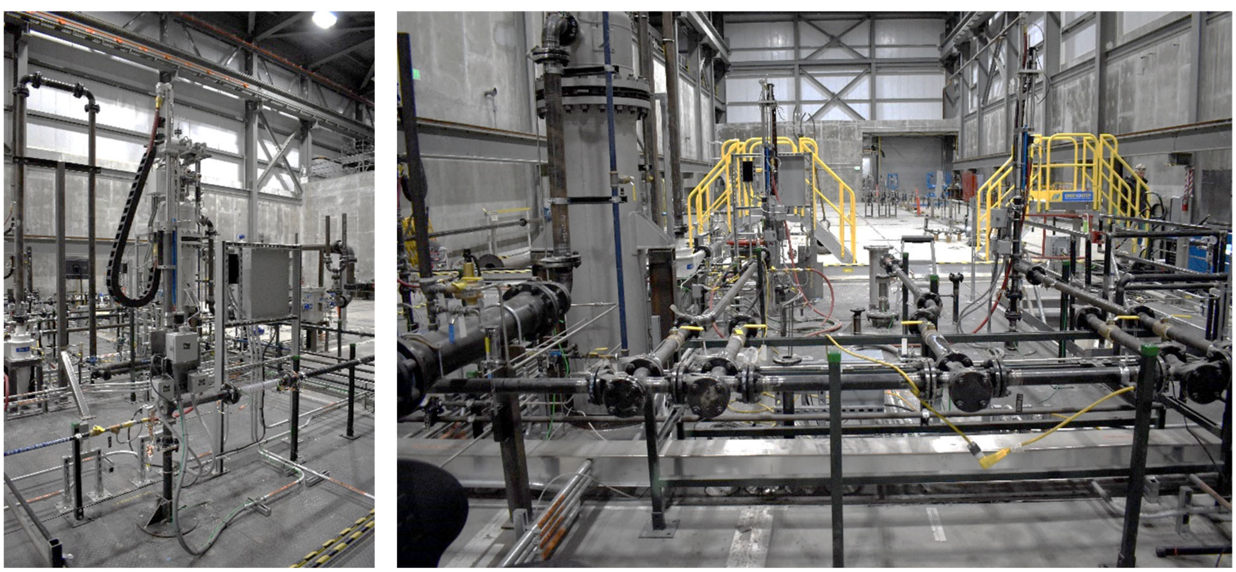

Figure 5-9. Full-scale mockup bulk retrieval unit (left) and transport manifold (right) at the former Fuel Processing Restoration Facility (CPP-691).

Figure 5-10. Video monitoring system images from inside Calcined Solids Storage Facility 1 during technology testing.

5.1.2 Deployment and Operation of the Waste Retrieval System

Deployment and operation of the pneumatic retrieval and transfer system have been tested and proven by the retrieval demonstration (Sandow 2021). Tests of the bulk retrieval system and air lance in the full-scale integrated mockup have shown that the bulk retrieval system is effective and will remove ~99% of the waste (Sandow 2021; M-1574).

Operation of the pneumatic retrieval and transfer system continues to be optimized by the retrieval demonstration. For example, testing data have been used to determine the number of access risers required to achieve optimum bulk retrieval efficiency (Sandow 2021; M-1574). In addition, testing data from the air lance have been used to inform the most efficient use of this device to move calcine from the stiffening rings and bin wall to the bin floor for removal by the retrieval and transfer system.

During retrieval operations, calcine will be removed from the CSSF to the maximum extent practical for closure. Waste will be removed using a pneumatic retrieval and transfer system until retrieval of additional calcine is no longer practical. Waste retrieval operations will be performed in a manner that protects workers, public health and safety, and the environment. Results of the retrieval demonstration have proven the effectiveness of the pneumatic retrieval and transfer system and the ability to meet the residual waste limits analyzed in the CSSF PA/CA (DOE-ID 2022a) (see discussion in Subsection 2.11.4.2).

5.1.3 Optimization of Waste Retrieval Technologies

DOE will continue to refine retrieval strategies as necessary to optimize calcine removal. Although pneumatic retrieval is the selected technology, DOE will also continue to participate in technology exchanges and evaluate new retrieval technologies that may address known challenges or improve technologies or processes that have already been selected.

The ability of pneumatic retrieval to remove HRRs to the maximum extent practical has been proven by previous and ongoing retrieval demonstrations (described in Subsections 2.11.3.3 and 5.2.1) and will be achieved by optimizing retrieval designs and operations (as described in Subsection 5.2.2). These demonstrations and design improvements will ensure that calcine removal is achieved to the maximum extent practical. Based on completed tests and demonstrations, full-scale mockup testing prior to deployment, and deployment of the retrieval and transfer system as previously described, retrieval operations using the pneumatic retrieval system will result in significant removal of the calcine, including its HRRs.[57]

5.2 Removal to the Maximum Extent Practical

As stated above, NDAA Section 3116(a) provides that certain waste resulting from SNF reprocessing is not HLW if the Secretary of Energy, in consultation with the NRC, determines, among other things, that the waste has had HRRs removed "to the maximum extent practical." This subsection summarizes the costs and benefits of further removal of calcine (and the HRRs therein) from the CSSF following guidance in NRC's NUREG-1854 (NRC 2007).

As discussed previously, engineering retrieval studies show that approximately 99% of the calcine (including the HRRs) would be removed from the bins during the retrieval process (see Subsection 2.11.3 and Table 2-11). This has been confirmed by previous studies referenced above and the current demonstration with the full-scale integrated mockup.[58]

As discussed in the following subsections, further removal of residual calcine waste after completion of retrieval operations (including any residual waste cleanout system, if necessary) would be impractical, increase costs, add schedule delay, increase the potential risk to workers, and result in an insignificant reduction in the very low potential doses to the public and the hypothetical human intruder. DOE has evaluated and selected a waste retrieval system that will optimize and successfully remove HRRs to the maximum extent practical. Various retrieval technologies were considered for effectively removing residuals from the CSSF, with the goal of protecting the public, the hypothetical human intruder, and the occupational health and safety of workers.

5.2.1 Estimated Cost of Waste Removal, Treatment Operations, and Facility Disposition

This subsection presents the most recent cost estimates for conducting a full-scale retrieval demonstration, retrieving and transferring calcine from CSSF 1 to CSSF 6, and placing CSSF 1 in a safe configuration. The information presented here is for comparison and to provide a basis for the cost per annual dose reduction estimate, which is presented in Subsection 5.3.2.

To support the added contract scope under the Calcine Retrieval Project, the ICP contractor provided DOE with a 5-year rough-order of magnitude (ROM) estimate (Williams 2016). This 5-year ROM estimate covered activities recommended in the AoA Summary Report (i.e., conduct a full-scale demonstration, retrieve and transfer calcine from CSSF 1 to CSSF 6, and place CSSF 1 in a safe configuration). The total ROM estimated cost, which included a 40% contingency due to project uncertainty and risk, is $52.1 million. As the project progresses, project scope and estimates will become

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57. NDAA Section 3116(a) does not specify "remedial goals" or other numerical objectives and does not require DOE to develop any such removal goals or objectives. Although the cleaning methodologies are expected to collectively remove 99% or more of the HRRs, waste removal from the individual bin sets may not achieve this level of HRR removal on an individual basis. Demonstration that waste removal within a particular bin has achieved 99% removal of HRRs is not, by itself, a justification for stopping HRR removal activities. In addition, demonstration that the residual radionuclide inventory of a given bin is below that assumed in the CSSF PA/CA (DOE-ID 2022a) is not the sole justification to conclude cleaning activities on an individual bin.

58. Information gathered from the full-scale integrated retrieval demonstration will be used as part of the iterative process of maintenance of the CSSF PA/CA (DOE-ID 2022a).

more definitive. Future cost estimates will continue to follow DOE cost estimating guidance (DOE G 413.3-21A) and ICP guidelines and standards.

5.2.2 Public Dose Reduction from Further Waste Removal

The CSSF PA/CA (DOE-ID 2022a) evaluated potential future doses to members of the public based on a residual waste thickness of 5.1 cm (2 in.) in the CSSF. The CSSF PA/CA dose results are presented in Sections 4 and 7 of this Draft 3116 Basis Document. Results of the PA dose assessment for the total annual all-pathways ED and the inadvertent intruder are summarized in Table 5-11. Based on the base case doses to a member of the public, removal of calcine beyond the estimated 5.1-cm (2-in.) residual depth assumed in the CSSF PA/CA would only further reduce the very low dose (3.45E-02 mrem) to a member of the public during the 10,000-year post-closure period. The maximum inadvertent intruder doses for the bins during the 10,000-year post-closure period were 7.1E+00 mrem ED for the acute intruder scenario and 3.6E+00 mrem annual ED for the chronic intruder scenario, both occurring 500 years post closure. As demonstrated in Subsection 5.3.3, future reductions in the calcine beyond the 5.1-cm (2-in.) residual depth would not provide significant reductions in the projected PA doses to the public or the inadvertent intruder.

5.2.3 Costs of Developing an Additional Retrieval Technology

In an effort to analyze the costs associated with removing additional residual waste, a ROM cost estimate is provided below. The cost is provided for retrieval technologies that have been designed and tested by the Calcine Retrieval Project for use in the CSSF (see Subsection 5.2.1.3). It is expected the bulk retrieval system will remove most of the calcine, achieving the 5.1 cm (2 in.) depth as assumed in the PA/CA base-case analysis. DOE has designed additional waste retrieval technologies, specifically, two custom-designed systems: the robotic vacuum crawler and the articulating arm. Generally, these systems will be able to remotely maneuver through, or to reach, most of the bin interior to remove residual calcine, and the systems will be able to deploy different tools, such as an air nozzle, a vacuum tool, and visual equipment. The ROM estimated costs for the fabrication, installation, and operation of these two systems (robotic vacuum crawler and articulating arm) at the CSSF are provided in Tables 5-12 and 5-13.

ROM estimated costs to use the two residual cleanout technologies are based on actual fabrication costs incurred to build the conceptual residual cleanout system prototypes and best-estimate ROM costs provided by the Calcine Retrieval Project engineering staff to install and operate the systems.

Table 5-14 shows incremental efficiencies of a hypothetical residual cleaning technology up to a maximum of 100%. This table shows projected annual doses (mrem in a year) from the CSSF PA/CA total annual all-pathways ED of 3.45E-02 mrem to a member of the public and associated averted annual dose for each increment of technology efficiency. Table 5-14 also shows the cost per mrem of deploying an additional removal technology. Cost per averted annual dose was estimated by using the total cost of $31 million for residual cleanout of the CSSF using an articulating arm, the least expensive of the two residual cleanout systems listed in Table 5-13. Table 5-13 costs show that additional radionuclide removal with less efficient technology costs more per averted dose, and, conversely, it costs less per averted dose when the technology is more efficient. Regardless of cost or efficiency, any increment of averted annual dose would not significantly reduce the risks because the 3.45E-02-mrem total annual all-pathways ED is 3,000,000 times less than the 25-mrem annual performance objective.

Table 5-11. Comparison of the performance objectives and the Calcined Solids Storage Facility base case performance assessment results for the institutional control and post-closure periods.

Performance Objective	Standard	Performance Assessment Results[a]			
		100-yr Institutional Control Period			
		(100 m [328 ft] Downgradient of CSSF)	(12.7 km [7.9 mi] Downgradient at Southern INL Site Boundary)	100- to 1,000-yr Post-Closure Period (100 m [328 ft] Downgradient of CSSF)	1,000- to 10,000-yr Post-Closure Period (100 m [328 ft] Downgradient of CSSF)
Total all pathways[b] (DOE O 435.1 Chg 2)	25-mrem annual ED	NA	1.35E-04 mrem	6.66E-06 mrem	3.45E-02
Groundwater all-pathways		NA	<1E-10 mrem[c]	<1E-10 mrem[c]	3.45E-02
Atmospheric all-pathways		NA	1.35E-04 mrem	6.66E-06 mrem	<6.66E-06
Acute inadvertent intruder (DOE O 435.1 Chg 2)	500-mrem total ED	NA	NA	7.1E+00 mrem[d] 5.6E-02 mrem[e]	4.5E+00 mrem 3.6E-02 mrem
Chronic inadvertent intruder (DOE O 435.1 Chg 2)	100-mrem annual ED	NA	NA	3.6E+00 mrem[f] 6.2E-02 mrem[g]	3.4E+00 mrem 5.4E-02 mrem

a. Doses presented as less than 1E-10 are essentially zero.

b. Total all-pathways annual ED is equal to the sum of the groundwater and atmospheric all-pathways ED.

c. A peak dose of 6.79E-14 mrem was projected during the 1,000-yr post-closure period. This is less than 1E-10 and thus essentially zero. It was also calculated at 100-m (328 ft) downgradient of CSSF 1, not the INL Site boundary; thus, the annual ED during the 100-yr institutional control period at the INL Site boundary would be less than 6.79E-14 mrem.

d. Maximum total ED for an acute exposure at the CSSF bins 500 yr after closure.

e. Maximum total ED for an acute exposure at the CSSF transport lines 500 yr after closure.

f. Maximum annual ED for a chronic exposure at the CSSF bins 500 yr after closure.

g. Maximum annual ED for a chronic exposure at the CSSF transport lines 500 yr after closure.

CSSF Calcined Solids Storage Facility
ED effective dose
INL Idaho National Laboratory
NA not applicable

Table 5-12. Rough order of magnitude cost estimate to fabricate, install, and operate residual cleanout technology at Calcined Solids Storage Facility 1.

Description	Articulating Arm	Robotic Vacuum Wall Crawler
Materials and fabrication	\$100,000	\$100,000
Installation	\$100,000	\$100,000
Operation	\$300,000	\$600,000
\$/bin	\$500,000	\$800,000
CSSF 1 Total 5.1 cm (2 in.) Residual Calcine (m^3)[a]	2.14	2.14
CSSF 1 (\$/m^3/bin)	**\$234,183**	**\$374,692**

a. This is the expected estimated residual waste volume remaining following bulk retrieval in CSSF 1.

CSSF Calcined Solids Storage Facility

Table 5-13. Rough-order of magnitude cost estimate to fabricate, install, and operate residual cleanout technology at each Calcined Solids Storage Facility based on residual cleanout cost of CSSF 1.

			ROM Cost Estimate to Deploy Residual Cleanout System[a]	
CSSF	Total Residual Calcine (m^3)	Number of Bins	Articulating Arm	Robotic Vacuum Crawler
1	2.14	12[b]	\$6,013,819	\$9,622,091
2	3.73	7	\$6,114,518	\$9,783,208
3	3.73	7	\$6,114,518	\$9,783,208
4	1.60	3	\$1,124,078	\$1,798,522
5	3.32	7	\$5,444,413	\$8,707,842
6	4.08	7	\$6,688,266	\$10,701,204
Total residual cleanout cost estimate			**\$31M**	**\$50M**

a. ROM cost estimate to deploy (fabricate, install, and operate) the residual cleanout systems is calculated by multiplying the residual cleanout cost of one CSSF 1 bin (\$234,183 for the articulating arm and \$374,692 for the robotic vacuum crawler [Table 5-12]) by the total residual calcine (m^3) and then multiplying by the number of CSSF bins.

b. Each of the four CSSF 1 bins consists of three concentric sub-bins, which are collectively referred to as a bin group (see Subsection 2.11.1.1). For the purposes of this ROM cost estimate, each sub-bin was counted individually; thus, the total is 12 CSSF 1 bins.

CSSF Calcined Solids Storage Facility
ROM rough order of magnitude

Table 5-14. Cost per averted total annual all-pathways effective dose for the Calcined Solids Storage Facility.

	Efficiency of Technology			
	25%	50%	75%	100%
Projected total annual all-pathways ED (mrem)[a]	2.59E-02	1.73E-02	8.63E-03	0.00E+00
Averted total annual all pathways ED (mrem)	8.63E-03	1.73E-02	2.59E-02	3.45E-02
Cost per averted annual dose[b,c]	**$3.6B per mrem**	**$1.8B per mrem**	**$1.2B per mrem**	**$899M per mrem**

a. Projected annual dose from the CSSF PA/CA total annual all-pathways ED for the 10,000-yr post-closure period (groundwater and atmospheric) of 3.45E-02 mrem (see Table 5-11).

b. Cost per averted annual dose was calculated following examples provided in NUREG-1854 (NRC 2007).

c. Cost per averted annual dose was estimated by using the least expensive of the two residual cleanout systems in Table 5-13, which is $31 million for the articulating arm.

CA composite analysis
CSSF Calcined Solids Storage Facility
ED effective dose
PA performance assessment

Table 5-15 shows the projected total ED of 7.1E+00 mrem from the CSSF PA/CA hypothetical inadvertent acute intruder scenario and associated averted dose for each increment of technology efficiency. Table 5-15 also shows the cost per mrem of deploying an additional removal technology. Regardless of cost or efficiency, any increment of averted total acute intruder dose would not significantly reduce the risks because the 7.1E+00-mrem total ED is only 1.4% of, or 70 times less than, the 500-mrem total ED performance objective and performance measure. In addition, increased worker doses would occur due to the installation of an additional removal technology into the CSSF bins.

Table 5-16 shows the projected annual ED of 3.6E+00 mrem from the CSSF PA/CA hypothetical inadvertent chronic intruder scenario and associated averted dose for each increment of technology efficiency. Table 5-16 also shows the cost per mrem of deploying a more efficient technology. Regardless of cost or efficiency, any increment of averted total chronic intruder dose would not significantly reduce the risks because the 3.6E+00 mrem annual ED is only 3.6% of, or 27 times less than, the 100-mrem annual ED performance objective.

As shown in Subsection 5.3.2, remaining waste residuals, if left in place, pose a potential total annual all-pathways dose to a member of the public on the order of 3.45E-02 mrem. Table 5-14 shows that the cost per mrem of deploying an additional technology would be at least $899 million per mrem for a 100% efficient system. Furthermore, any new waste removal systems could only achieve a reduction of approximately 3.45E-02 mrem regardless of cost. On average, Americans receive a radiation dose of approximately 6.20E+02 mrem each year (NCRP 2009). Half of this dose comes from natural background radiation. Most of this background exposure comes from radon in the air, with smaller amounts from cosmic rays and the Earth itself. The other half, approximately 3.10E+02 mrem, comes from man-made sources of radiation, including medical, commercial, and industrial sources. Substantial and convincing scientific data show evidence of human health effects following high-dose exposures greater than 10,000 mrem above background. However, the observed radiation health effects in people are not statistically different from zero below this exposure level (HPS 2019).

Table 5-15. Cost per averted acute intruder total effective dose for the Calcined Solids Storage Facility.

	Efficiency of Technology			
	25%	50%	75%	100%
Projected total ED (mrem)[a]	5.32E+00	3.55E+00	1.78E+00	0.00E+00
Averted total ED (mrem)	1.78E+00	3.55E+00	5.32E+00	7.10E+00
Cost per averted total ED[b,c]	$17.4M per mrem	$8.7M per mrem	$5.8M per mrem	$4.4M per mrem

a. Projected acute intruder total ED for the CSSF bins from the CSSF PA/CA is 7.1E+00 mrem (see Table 5-11).

b. Cost per averted total ED was calculated following examples provided in NUREG-1854 (NRC 2007).

c. Cost per averted total ED was estimated by using the least expensive of the two residual cleanout systems in Table 5-13, which is $32 million for the articulating arm.

CA composite analysis
CSSF Calcined Solids Storage Facility
ED effective dose
PA performance assessment

Table 5-16. Cost per averted chronic intruder annual effective dose for the Calcined Solids Storage Facility.

	Efficiency of Technology			
	25%	50%	75%	100%
Projected annual ED (mrem)[a]	2.70E+00	1.80E+00	9.00E-01	0.00E+00
Averted annual ED (mrem)	9.00E-01	1.80E+00	2.70E+00	3.60E+00
Cost per averted annual ED[b,c]	$34.4M per mrem	$17.2M per mrem	$11.5M per mrem	$8.6M per mrem

a. Projected chronic intruder annual ED for the CSSF bins from the CSSF PA/CA is 3.6E+00 mrem (see Table 5-11).

b. Cost per averted annual ED was calculated following examples provided in NUREG-1854 (NRC 2007).

c. Cost per averted annual ED was estimated by using the least expensive of the two residual cleanout systems in Table 5-13, which is $32 million for the articulating arm.

CA composite analysis
CSSF Calcined Solids Storage Facility
ED effective dose
PA performance assessment

5.3 Conclusion

Based on specific CSSF conditions described in Subsection 5.2, DOE has determined that a limited number of technologies would effectively remove residuals from the CSSF. Benefits of developing additional technologies, as well as complete waste removal, were evaluated for reducing residuals in the CSSF. It is likely that a system could be developed and deployed to remove a portion of the remaining residual by other remote means. However, initiating a long-term research project to develop other technologies would take many years (5 to 10 years) to complete, and the project would be costly. "Hands-on" cleaning would pose a radiological risk to involved workers and thus is not practical. Any new technology, including the two remote systems described above, would increase worker exposure to install the equipment in the bins while the projected dose to the public from the residual waste, which is already low, would be lowered only slightly by using the new technology. No appreciable decrease in the projected radiation dose to the public or the inadvertent intruder would be gained even if the 5.1-cm (2-in.) depth of residual waste remaining after retrieval operations is removed entirely.

Remaining waste residuals, if left in place, pose a projected total ED (mrem) from the CSSF PA/CA inadvertent acute intruder dose of 7.1E+00 mrem total ED. Table 5-15 shows the cost per mrem of deploying an additional removal technology would be at least approximately $4.4 million per mrem for a 100% efficient system. Regardless of cost or efficiency, any increment of averted total acute intruder dose would not significantly reduce the risks because the 7.1E+00-mrem total ED is only 1.4% of, or 70 times less than, the 500-mrem total ED performance measure.

Remaining waste residuals, if left in place, pose a projected annual ED (mrem) from the CSSF PA/CA inadvertent chronic intruder dose of 3.6E+00-mrem annual ED. Table 5-16 shows the cost per mrem of deploying an additional technology would cost at least approximately $8.6 million per mrem for a 100% efficient system. Regardless of cost or efficiency, any increment of averted total chronic intruder dose would not significantly reduce the risks because the 3.6E+00-mrem annual ED is only 3.6% of, or 27 times less than, the 100-mrem annual ED performance objective and performance measure.

Further removal of CSSF residual waste after the initial bulk retrieval—which is expected to result in a residual depth of approximately 5.1 cm (2 in.)—is not cost effective, and any efforts to remove more of the small quantity of waste remaining would increase risk to the workers and not significantly reduce the potential risk to the public and the environment. DOE has evaluated and selected (and continues to test in preparation for actual retrieval operations) a waste retrieval system that optimizes and successfully removes the waste, including HRRs, to the maximum extent practical. Various retrieval technologies were considered for effectively removing waste and residuals from the CSSF, with the goal of protecting public and occupational health and safety at the closure site.

Thus, for the reasons discussed previously, DOE has determined that HRRs have been, or will be, removed from the CSSF to the maximum extent practical.

6. RADIONUCLIDE CONCENTRATIONS OF STABILIZED WASTE IN THE CSSF

Section 6 Purpose

The purpose of this section is to demonstrate whether the CSSF stabilized residuals at closure will meet concentration limits for Class C LLW as set out in 10 CFR 61.55, criterion (3)(A) of NDAA Section 3116(a).

Section 6 Contents

This section provides the methodology and assumptions to demonstrate whether the CSSF stabilized residuals at closure meet Class C concentration limits.

Section 6 Key Points

- The stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein remaining in the CSSF will meet Criterion (3)(A) of NDAA Section 3116(a) concentration limits for Class C LLW as set out in 10 CFR 61.55.
- For this analysis, DOE is using the approach developed between DOE and NRC technical staff for site-specific concentration averaging expressions based on the site-specific intruder-driller scenarios and guidance provided in NUREG-1854 – Category 3.
- Stabilized residual waste in the CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are expected to meet the Class C LLW concentration limits in Tables 1 and 2 of 10 CFR 61.55.
- The overall total decay period for the CSSF bins and transport lines in this section are the same as for the analyses conducted in the CSSF PA/CA (i.e., 500 years of decay for the bins and transport lines). However, the analyses in this section involve 100 years of decay that have already been accounted for in the NRC Class A values used for comparisons. Therefore, only 400 years of additional inventory decay for the bins and transport lines were required, which, along with the inherent 100 years of decay in the NRC Class A values, results in 500 years of decay for the bins and transport lines.
- While DOE believes there is a reasonable basis to conclude that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will not exceed the Class C concentration limits in 10 CFR 61.55, DOE is consulting with the NRC on DOE's disposal plans, as described in this Draft CSSF Basis Document, to take full advantage of the NDAA Section 3116(a) consultation process.

This section demonstrates that stabilized CSSF bins (including integral equipment), transport lines and any residual calcine therein at the time of closure will meet concentration limits for Class C LLW—as established in Tables 1 and 2 of 10 CFR 61.55—per Criterion (3)(A) of NDAA Section 3116(a) (Public Law 108-375).

The NDAA Section 3116(a) provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy…, in consultation with the Nuclear Regulatory Commission…, determines –

(3)(A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations.

Based on DOE's closure approach described in Subsection 2.11.4, remaining residual waste will be stabilized in a solid physical form within the CSSF. The concentration of radionuclides in the stabilized waste is not expected to exceed the concentration limits for Class C LLW, which are provided in Tables 1 and 2 of 10 CFR 61.55 and reproduced in Tables 5-8 and 5-9 in Subsection 5.1.2 of this Draft CSSF 3116 Basis Document. The methodology for comparing the radionuclide concentration in stabilized residual waste to the Class C concentration limits is presented in the following subsections.

6.1 Approach to Calculating CSSF Residual Waste Concentrations

Prior NRC guidance to determine concentrations for comparison with Class C concentration limits of 10 CFR 61.55 was based on a basement excavation scenario as the likely pathway to expose an inadvertent intruder to waste in a commercial shallow land burial site (NRC 2007). Because of the disposal depth of the CSSF stabilized residuals in the bins, the basement excavation scenario associated with development of Tables 1 and 2 of 10 CFR 61.55 is not applicable. Residual waste in the bins is a minimum of 13.7 m (45 ft) bls, while potential residual waste in the transport lines at the shallowest point is approximately 3.5 m (11.5 ft) bls. A more appropriate scenario for the purposes of calculation and comparison with Class C concentration limits is one that assumes the inadvertent intruder drills into a bin or transport line.

The CSSF stores calcine in six discrete units. Each CSSF consists of several stainless-steel storage bins housed within a concrete vault. The reinforced-concrete vaults range in thickness from 0.53 m (1.75 ft) to 2 m (6.5 ft). The bins are constructed of Type 405, 304, and 304L stainless-steel that has minimum thickness of 3.18 mm (0.125 in.), 6.4 mm (0.25 in.), and 9.53 mm (0.375 in.), respectively. The fact that the bins are all constructed of stainless steel and contained in reinforced-concrete vaults provides assurance that any calcine left in the storage bins after retrieval operations has a reduced potential for inadvertent intrusion for a very long time. However, it is assumed in the CSSF PA/CA (DOE 2022a) that inadvertent intrusion into the bins occurs at 500 years after CSSF closure (DOE-ID 2022a). Subsection 7.3.3 presents a qualitative discussion of the likelihood of inadvertent human intrusion after closure of the CSSF.

The transport lines used to transfer solids (calcine) from WCF and NWCF to the CSSF storage bins vary in size, length, and depth for each line, as described in EDF-11119. The transport lines travel in a larger containment pipe that is encased in reinforced concrete for shielding. The configuration of the stainless-steel transport lines surrounded by reinforced concrete is considered to provide a robust barrier that precludes inadvertent intrusion directly into the residual waste until 500 years post-closure. The 3-in. transport lines are Schedule 40 stainless-steel pipe with a thickness of 5.49 mm (0.216 in.). The average corrosion rate for Type 304 stainless steel for CSSF calcine-specific corrosion coupons was found to be 2.06E-04 mm/year (see Table 4-4 of the CSSF PA/CA [DOE-ID 2022a]). At that rate, it would take 26,650 years to corrode through the total thickness of the pipe wall. That corrosion rate is considered to be bounding because corrosion rates in INL Site soil and concrete are expected to be much lower than the corrosion rates based on corrosion coupons in calcine (DOE-ID 2022a).

For this analysis, DOE is using the approach developed between DOE and NRC technical staff for site-specific concentration averaging expressions for residual waste based on the site-specific intruder-driller scenarios and the guidance in NUREG-1854 (NRC 2007). This methodology is also documented in the NRC's *Technical Evaluation Report: Draft Waste Incidental to Reprocessing Evaluation for Closure of Waste Management Area C, Hanford Site, Washington, Final Report* (NRC 2020). DOE believes these averaging expressions have limited consistency with the assumptions used to develop the 10 CFR 61.55 waste classification system, but they provide a method to evaluate both acute and chronic intruder scenarios consistent with the Category 3 approach and NRC guidance provided in NUREG-1854 (NRC 2007).

The following subsections present the methodology, inputs, and assumptions DOE used to compare the concentration of CSSF stabilized residual waste at closure to the Class C concentration limits.

6.2 Methodology

The Category 3 approach to concentration averaging reflects CSSF site-specific conditions and the final form of stabilized residual waste to account for the volume, concentration, and accessibility of the residual material. In order to account for the site-specific conditions relative to CSSF, DOE has developed, consistent with the Category 3 methodology, averaging expressions for CSSF based on the results of the inadvertent intruder analysis performed within the CSSF PA/CA (DOE-ID 2022a). As discussed in the following subsections, the concentrations of stabilized residual waste have been compared utilizing these averaging expressions against the concentration limits for Class C LLW as set out in Tables 1 and 2 of 10 CFR 61.55.[59] For the CSSF bins and transport lines, this comparison was based on the projected inventories at closure as presented in the CSSF PA/CA.

For purposes of comparison to the Class C concentration limits and to align with the inputs used in developing the averaging expressions for CSSF, the residual inventory used for these calculations is based on the CSSF PA/CA inventory at 2016 as the starting inventory for all subsequent decay and ingrowth calculations.

To demonstrate compliance with the performance objectives set out in 10 CFR 61, Subpart C, DOE developed a PA covering closure activities within the CSSF. To demonstrate compliance with 10 CFR 61.42, the CSSF PA/CA is used to demonstrate that there is reasonable assurance the total ED to an inadvertent intruder will remain below the 500-mrem dose limit, taking into consideration a variety of intruder scenarios. A detailed discussion of the intruder analyses is provided in the CSSF PA/CA (DOE-ID 2022a).

Stabilized residual waste after closure of the CSSF will be located primarily in areas protected by significant materials that are clearly distinguishable from the surrounding soil and make drilling an unlikely scenario due to the presence of multiple barriers, including reinforced concrete, grout, and stainless-steel bins. Such barriers in the CSSF are assumed to be effective in precluding inadvertent drilling for 500 years. The transport lines travel in a larger containment pipe that is encased in reinforced concrete for shielding. The configuration of the stainless-steel transport lines surrounded by reinforced concrete is considered to provide a robust barrier that precludes inadvertent intrusion directly into the residual waste until 500 years post-closure (see Subsection 2.11.4.3 for configuration at closure). Future signage providing warnings about potential risk from buried waste at the CSSF will be managed in accordance with the *INL Site-Wide Institutional Controls, and Operations and Maintenance Plan for CERCLA Response Actions* (DOE-ID 2022d) and any post-closure HWMA/RCRA permit requirements.

The following subsections describe how the sum of fractions (SOF) is calculated for the CSSF bins and transport lines.

6.2.1 Site-Specific CSSF Waste Concentration Calculation Averaging Expression

This methodology is documented in NRC's Technical Evaluation Report for the Hanford Waste Management Area C Waste Incidental to Reprocessing Evaluation (NRC 2020). As noted above, DOE and NRC technical staff conferred on an approach to develop averaging expressions for the Class C calculations (DOE 2019). As recommended by the NRC in Table 4-3 of the Technical Evaluation Report, the concentration values in Tables 1 and 2 of 10 CFR 61.55 should be divided by 10 (except for Cs-137)

59. As recommended by the NRC in Table 4-3 of the Technical Evaluation Report (NRC 2020), the concentration values in Tables 1 and 2 of 10 CFR 61.55 should be divided by 10 (except for Cs-137) to account for an initial increase of the values during their development. This results in a comparison to the Class A concentration limits rather than the Class C limits (see Subsection 6.2.1 of this document).

to account for an initial increase of the values during their development. This results in a comparison to the Class A concentration limits.

Consistent with the Category 3 approach outlined in NUREG-1854 (NRC 2007) and NRC guidance to use Class A concentration limits (DOE 2019; NRC 2020), the averaging expression used to determine the individual radionuclide contribution to the SOF based on the acute drilling scenario is represented by the Equation 6-1:

$$SOF_i = \frac{C_{Ri}}{Table_{Value_i}} * \left(\frac{Waste_{thickness}}{Drill_{depth}}\right) * \left(\frac{Exposure_{drill}}{Exposure_{NRC}}\right) * \left(\frac{1}{0.5}\right) \quad (6\text{-}1)$$

where:

SOF_i	=	radionuclide "i" contribution to the sum of fractions
C_{Ri}	=	concentration of radionuclide "i" at closure decayed 400 years for the bins and 400 years for the transport lines (Ci/m^3 or nCi/g)
$Table_{Valuei}$	=	Class A concentration limit from 10 CFR 61.55 Table 1 or 2 for radionuclide "i"
$Waste_{thickness}$	=	thickness of the CSSF residual waste for radionuclide "i" (m)
$Drill_{depth}$	=	total depth of the well at the CSSF (m)
$Exposure_{drill}$	=	time of exposure for the CSSF PA/CA acute drilling scenario (hours)
$Exposure_{NRC}$	=	time of exposure for the NRC acute excavation scenario (hours)
0.5	=	NRC dilution factor assumption for Class C limit; waste barrels are 50% full of waste (dimensionless).

C_{Ri} in Equation 6-1 is calculated as shown in Equation 6-2:

$$C_{Ri} = \frac{I_{Ri}}{V_w} \; or \; \frac{I_{Ri}}{M_w} \quad (6\text{-}2)$$

where:

I_{Ri}	=	inventory of radionuclide "i" at closure decayed 400 years for the bins and 400 years for the transport lines (Ci or nCi)
V_w	=	residual waste volume (m^3)
M_w	=	residual waste mass (g).

For the chronic post-drilling scenario, the averaging expression used to determine the individual radionuclide contribution to the SOF is represented by the Equation 6-3:

$$SOF_i = \frac{C_{Ri}}{Table_{Value_i}} * \left(\frac{\frac{V_{w,drill}}{V_{T,drill}}}{\frac{V_{w,NRC}}{V_{T,NRC}}}\right) * \left(\frac{1}{0.5}\right) * 4 \quad (6\text{-}3)$$

where:

SOF_i	=	radionuclide "i" contribution to the SOF
C_{Ri}	=	concentration of radionuclide "i" at closure decayed 400 years for the bins and 400 years for the transport lines (Ci/m^3 or nCi/g)
$Table_{Valuei}$	=	Class A concentration limit from 10 CFR 61.55 Table 1 or 2 for radionuclide "i"
$V_{w,drill}$	=	volume of waste brought to the surface from drilling (m^3)
$V_{T,drill}$	=	total volume of soil brought to the surface from drilling (m^3)

$V_{w,NRC}$	=	volume of waste brought to the surface from NRC excavation scenario (m^3)
$V_{T,NRC}$	=	total volume of soil brought to the surface from NRC excavation scenario (m^3)
0.5	=	dilution factor assumption for Class C limit, waste barrels are 50% full of waste (dimensionless)
4	=	factor to account for contaminated areas that are not plowed (dimensionless).

The overall total decay period for the bins and transport lines in this section are the same as for the analysis conducted in the CSSF PA/CA (i.e., 500 years of decay for the bins and transport lines models). However, the analysis in this section involves 100 years of decay that have already been accounted for in the NRC Class A values used for comparisons. Therefore, only 400 years of additional inventory decay for the bins and transport lines were added, which, along with the inherent 100 years of decay in the NRC Class A values, results in 500 years of decay for the bins and transport lines.

Intrusion is assumed to occur 500 years after facility closure[60] for components with a robust barrier. Because the Class A limits from 10 CFR 61.55 have a 100-year decay accounted for in their derivation, another 400 years is added for the bins and transport lines in the equations.

6.2.2 Parameters and Inputs

The residual inventory used for the concentration calculations is the residual calcine material within the bins (5.1-cm-thick [2-in.-thick] residual) or transport lines (approximately one-twenty-fifth of the pipe volume is assumed to contain residual waste). The residual material layer in the bins is assumed to be spread evenly across the floor of the bins. The residual material within transport lines is assumed to be spread evenly over the drill hole diameter. Table 6-1 provides the input data used for the calculations presented in the remainder of this section. Table 6-2 provides the waste volume and waste mass for 5.1 cm (2 in.) of residual waste in the CSSF bins and the residual waste inventory decayed 400 years from 2016, which is the date of the most recent radionuclide activity estimates for the CSSF calcine (Staiger and Swenson 2021; EDF-11126) and was also the inventory date used in the CSSF PA/CA (DOE 2022a).

Table 6-3 provides the waste volume and waste mass for residual waste in the transport lines based on each CSSF inventory decayed 400 years from 2016. A portion of the CSSF 1 transport lines (a section approximately 6.1 to 9.1 m [20 to 30 ft] in length) was removed during the HWMA/RCRA closure because it was partially filled with CSSF 2 cold startup material. In addition, the CSSF 1 transport line was grouted on both sides of the removed section—west from the excavation point to the WCF and west from CSSF 1 to the excavation point and left in place (EDF-11119). Therefore, no deposits remain in the CSSF 1 transport lines and the CSSF 1 waste profile is not considered in the transport line intruder analyses.

60. The CSSF PA/CA analysis assumed 2016 closure based on the date used for the development of the inventory. Therefore, intrusion in the PA/CA was assumed to occur 500 years after closure, which would be 2516.

Table 6-1. Class C calculation input parameter values.

Parameter	Notation	Value	Reference
400-year decayed radionuclide inventory[a]	I_{Ri}	See Table 6-2	CSSF PA/CA (DOE-ID 2022a)
Waste volume	V_w	See Table 6-2	CSSF PA/CA (DOE-ID 2022a)
Waste mass	M_w	See Table 6-2	CSSF PA/CA (DOE-ID 2022a)
Waste thickness	$Waste_{thickness}$	5.08E-02 m (bins) 1.13E-03 m (transport lines)	EDF-11132
Drill depth	$Drill_{depth}$	122 m	EDF-11132
Exposure time for drilling	$Exposure_{drill}$	160 hr	EDF-11132
Exposure time for NRC scenario[b]	$Exposure_{NRC}$	500 hr	NUREG-0782 (NRC 1981)
Volume of waste from drilling	$V_{w,drill}$	1.60E-03 m^3 (bins) 3.56E-05 m^3 (transport lines)	EDF-11132
Volume of waste for NRC scenario[b]	$V_{w,NRC}$	150 m^3	NUREG-0782 (NRC 1981)
Total soil volume from drilling	$V_{T,drill}$	3.83 m^3	EDF-11132
Total soil volume for NRC scenario[b]	$V_{T,NRC}$	600 m^3	NUREG-0782 (NRC 1981)
10 CFR 61.55 limits	Table $value_i$	See Class A table values in Tables 6-4 through 6-25	10 CFR 61.55

a. The overall total decay period for the bins and transport lines in this section are the same as those used for the analyses conducted in the CSSF PA/CA (i.e., 500 years of decay for the bins and transport lines models). However, the analyses in this section involve 100 years of decay that have already been accounted for in the NRC Class A values used for comparisons. Therefore, only 400 years of additional inventory decay for the bins and transport lines were added, which, along with the inherent 100 years of decay in the NRC Class A values, results in 500 years of decay for the bins and transport lines.

b. The NRC scenario refers to the values used in the intruder analyses presented in NUREG-0782 (NRC 1981) in support of the development of the 10 CFR 61.55 limits.

CA composite analysis
CSSF Calcined Solids Storage Facility
NRC U.S. Nuclear Regulatory Commission
PA performance assessment

Table 6-2. Waste volume and waste mass for 5.1 cm (2 in.) of residual waste in the Calcined Solids Storage Facility bins and the residual waste inventory decayed 400 years from 2016.

Nuclide	CSSF 1 (Ci)	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)
waste (m^3)	*2.13*	*3.74*	*3.74*	*1.6*	*3.32*	*4.08*
density (g/m^3)	*8.40E+05*	*1.23E+06*	*1.29E+06*	*1.36E+06*	*1.38E+06*	*1.23E+06*
waste (g)	*1.79E+06*	*4.60E+06*	*4.83E+06*	*2.18E+06*	*4.58E+06*	*5.02E+06*
C-14	6.32E-06	3.76E-06	2.48E-09	1.42E-09	3.06E-09	1.43E-09
Ni-59	0.00E+00	8.95E-02	2.01E-01	1.16E-01	2.48E-01	1.16E-01
Nb-94	1.27E-05	7.13E-02	1.60E-01	9.20E-02	1.98E-01	9.24E-02
Tc-99	4.12E+00	3.38E+00	2.47E+00	1.41E+00	3.08E+00	1.43E+00
I-129	6.68E-05	5.51E-05	4.08E-05	2.33E-05	5.07E-05	2.35E-05
Np-237	1.07E-02	8.28E-03	2.76E-02	6.49E-02	1.25E-01	3.17E-02
Pu-238	1.30E-01	1.52E+00	2.36E+00	2.31E+00	4.50E+00	1.21E+00
Pu-239	4.10E-01	7.92E-01	1.47E+00	1.50E+00	2.92E+00	1.89E+00
Pu-240	1.59E-01	6.07E-01	1.03E+00	9.36E-01	1.99E+00	9.87E-01
Pu-241	4.66E-08	1.67E-06	3.65E-06	2.14E-06	4.61E-06	2.16E-06
Pu-242	9.57E-05	1.48E-03	2.73E-03	2.73E-03	4.88E-03	2.15E-03
Pu-244	1.87E-12	2.32E-12	2.72E-12	1.56E-12	3.36E-12	1.57E-12
Am-241	6.47E-01	2.95E+00	4.77E+00	3.14E+00	6.01E+00	1.71E+00
Am-242m	2.74E-05	3.76E-04	8.08E-04	4.65E-04	9.99E-04	4.66E-04
Am-243	8.09E-05	3.55E-04	9.92E-04	6.20E-04	1.09E-03	6.17E-04
Cm-242	2.26E-05	3.10E-04	6.67E-04	3.84E-04	8.25E-04	3.85E-04
Cm-243	4.83E-10	1.68E-08	3.71E-08	2.13E-08	4.59E-08	2.14E-08
Cm-244	2.81E-11	8.78E-10	1.80E-09	1.10E-09	2.06E-09	1.08E-09
Cm-245	4.15E-08	1.59E-06	3.52E-06	2.02E-06	4.36E-06	2.03E-06
Cm-246	9.23E-10	1.64E-07	3.69E-07	2.12E-07	4.56E-07	2.12E-07
Cm-247	3.51E-16	2.68E-13	6.02E-13	3.46E-13	7.45E-13	3.47E-13
Cm-248	1.12E-16	3.76E-13	8.44E-13	4.86E-13	1.05E-12	4.87E-13
Cf-249	9.89E-17	1.07E-12	2.39E-12	1.37E-12	2.96E-12	1.38E-12
Cf-250	1.16E-26	6.85E-22	1.54E-21	8.84E-22	1.90E-21	8.84E-22
Cf-251	4.28E-19	3.08E-14	6.92E-14	3.98E-14	8.52E-14	3.99E-14
Ni-63	0.00E+00	3.01E-01	5.82E-01	3.74E-01	6.58E-01	2.09E-01
Sr-90	4.02E-01	4.02E-01	3.24E-01	2.02E-01	4.33E-01	1.89E-01
Cs-137	8.02E-01	7.56E-01	6.27E-01	3.60E-01	7.75E-01	3.61E-01

CSSF Calcined Solids Storage Facility

Table 6-3. Waste volume and waste mass for 5.1 cm (2 in.) of residual waste in each Calcined Solids Storage Facility transport line (except CSSF 1) and the residual waste inventory decayed 400 years from 2016.

Nuclide	CSSF 2 (Ci)	CSSF 3 (Ci)	CSSF 4 (Ci)	CSSF 5 (Ci)	CSSF 6 (Ci)
waste (m^3)	*1.09E-01*	*1.09E-01*	*1.09E-01*	*1.09E-01*	*1.09E-01*
density (g/m^3)	*1.23E+06*	*1.29E+06*	*1.36E+06*	*1.38E+06*	*1.23E+06*
waste (g)	*1.34E+05*	*1.41E+05*	*1.48E+05*	*1.50E+05*	*1.34E+05*
C-14	6.89E-11	4.10E-11	6.31E-14	1.74E-14	3.06E-14
Ni-59	0.00E+00	9.75E-07	5.13E-06	1.42E-06	2.48E-06
Nb-94	1.39E-10	7.77E-07	4.07E-06	1.13E-06	1.98E-06
Tc-99	4.50E-05	3.68E-05	6.28E-05	1.73E-05	3.07E-05
I-129	7.28E-10	6.00E-10	1.04E-09	2.86E-10	5.07E-10
Np-237	1.17E-07	9.02E-08	7.02E-07	7.96E-07	1.25E-06
Pu-238	1.42E-06	1.65E-05	6.01E-05	2.83E-05	4.50E-05
Pu-239	4.47E-06	8.63E-06	3.75E-05	1.85E-05	2.92E-05
Pu-240	1.74E-06	6.61E-06	2.61E-05	1.15E-05	1.99E-05
Pu-241	5.08E-13	1.82E-11	9.30E-11	2.63E-11	4.61E-11
Pu-242	1.04E-09	1.61E-08	6.95E-08	3.35E-08	4.87E-08
Pu-244	2.04E-17	2.53E-17	6.93E-17	1.92E-17	3.36E-17
Am-241	7.06E-06	3.22E-05	1.21E-04	3.85E-05	6.01E-05
Am-242m	2.99E-10	4.10E-09	2.06E-08	5.71E-09	9.99E-09
Am-243	8.82E-10	3.87E-09	2.53E-08	7.62E-09	1.09E-08
Cm-242	2.47E-10	3.38E-09	1.70E-08	4.71E-09	8.24E-09
Cm-243	5.27E-15	1.83E-13	9.45E-13	2.62E-13	4.59E-13
Cm-244	3.06E-16	9.56E-15	4.60E-14	1.35E-14	2.06E-14
Cm-245	4.53E-13	1.73E-11	8.97E-11	2.48E-11	4.35E-11
Cm-246	1.01E-14	1.79E-12	9.39E-12	2.61E-12	4.55E-12
Cm-247	3.83E-21	2.92E-18	1.53E-17	4.25E-18	7.45E-18
Cm-248	1.22E-21	4.09E-18	2.15E-17	5.96E-18	1.05E-17
Cf-249	1.08E-21	1.16E-17	6.10E-17	1.69E-17	2.96E-17
Cf-250	1.26E-31	7.46E-27	3.92E-26	1.09E-26	1.90E-26
Cf-251	4.66E-24	3.36E-19	1.76E-18	4.88E-19	8.52E-19
Ni-63	0.00E+00	3.28E-06	1.48E-05	4.59E-06	6.58E-06
Sr-90	4.39E-06	4.39E-06	8.26E-06	2.47E-06	4.33E-06
Cs-137	8.75E-06	8.24E-06	1.60E-05	4.42E-06	7.74E-06

CSSF Calcined Solids Storage Facility

6.3 Waste Concentration Calculations

The remaining tables in this subsection provide comparisons of the CSSF radionuclide concentrations to the NRC waste classification concentration limits. A summary of the Class C SOF radionuclide concentration comparisons is provided in Table 6-4 for the bins and Table 6-5 for the transport lines. A SOF that is less than or equal to 1.0 indicates that the radionuclide concentrations in the residual waste (see Tables 6-6 through 6-27 for detailed SOF calculations) will meet the concentration limits for Class C LLW as set out in 10 CFR 61.55. The Class C concentration comparisons based on the acute intruder drilling scenario are provided in Tables 6-6 through 6-11 for the CSSF bins and Tables 6-12 through 6-16 for the transport lines based on each waste profile. The Class C concentration comparisons based on the chronic intruder drilling scenario are provided in Tables 6-17 through 6-22 for the CSSF bins and Tables 6-23 through 6-27 for the transport lines based on each waste profile.

To calculate the Class C waste concentration for the transport lines, the inventory for each CSSF—except for CSSF 1 because the lines were either removed or grouted under a HWMA/RCRA closure action[61]—was used with the assumption that the transport lines are one-twenty-fifth full. In the CSSF PA/CA inadvertent intruder dose assessment, it was assumed that one-twenty-fifth (3.9%) of the transport line volume was filled with residual waste, based on information provided in EDF-11119. Potential contamination in the CSSF transport lines is estimated to be 23.8 m (78 ft), while the total length of piping at the CSSF is 613.3 m (2,012 ft). Though most of the lines are empty after the calcining facility systems (calcining, processing, and transfer equipment) were scoured with high-velocity air or nonradioactive material to remove residual waste, some of the lines may have deposits or residual accumulation areas containing waste. The probability of drilling into one of these areas is low in comparison to the overall length of lines at the CSSF.

In addition, the Class C calculations for the transport lines were based on the equivalent waste thickness of the 3-in. transport line averaged over the well diameter. The 8-in. well diameter provides the maximum waste thickness (i.e., 1.13E-03 m) for calculation of Class C concentrations for the transport lines.

Table 6-4. Class C limit comparison summary for the Calcined Solids Storage Facility bins.

	Acute Class C Equation		Chronic Class C Equation	
Bin Set	Table 1 SOF	Table 2 SOF	Table 1 SOF	Table 2 SOF
CSSF 1	2.1E-02	1.4E-03	6.2E-03	3.9E-04
CSSF 2	3.5E-02	7.8E-04	9.9E-03	2.2E-04
CSSF 3	5.4E-02	6.3E-04	1.5E-02	1.8E-04
CSSF 4	9.9E-02	9.2E-04	2.8E-02	2.6E-04
CSSF 5	9.2E-02	9.5E-04	2.6E-02	2.7E-04
CSSF 6	3.2E-02	3.4E-04	8.9E-03	9.6E-05

CSSF Calcined Solids Storage Facility
SOF sum of fractions

61. The CSSF 1 solid transport line 3" TAA-3009 was partially filled in 1966 with nonradioactive material from the CSSF 2 startup operation. This section of transport lines (solids transport line 3" TAA-3009 and air return line 3" TAA-3001) was later removed during the WCF HWMA/RCRA closure in 1999 (Wessman 1999). A 6.1- to 9.1-m (20- to 30-ft) section of encased transport lines was removed, placed in the WCF operating corridor, and grouted with other building components. Transport line 3" TAA-3009 was grouted on both sides of the removed section—west from the excavation point to the WCF and west from CSSF 1 to the excavation point. The air return line 3" TAA-3001 was grouted west from CSSF 1 to the WCF (Wessman 1999).

Table 6-5. Class C limit comparison summary for the Calcine Solids Storage Facility transport lines.

Waste Profile[a]	Acute Class C Equation[b] Table 1 SOF	Acute Class C Equation[b] Table 2 SOF	Chronic Class C Equation[c] Table 1 SOF	Chronic Class C Equation[c] Table 2 SOF
CSSF 2	7.8E-04	1.7E-05	2.2E-04	4.9E-06
CSSF 3	1.2E-03	1.4E-05	3.4E-04	4.0E-06
CSSF 4	2.2E-03	2.0E-05	6.2E-04	5.8E-06
CSSF 5	2.1E-03	2.1E-05	5.8E-04	6.0E-06
CSSF 6	7.0E-04	7.5E-06	2.0E-04	2.1E-06

a. The CSSF 1 waste profile was not evaluated for the transport lines because these lines were closed (removed or grouted) during a previous HWMA/RCRA action.

b. Acute SOF calculations based on Equation 6-1.

c. Chronic SOF calculations based on Equation 6-3.

CSSF Calcined Solids Storage Facility
HWMA Hazardous Waste Management Act
RCRA Resource Conservation and Recovery Act
SOF sum of fractions

Table 6-6. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 for acute intruder drilling at the Calcined Solids Storage Facility 1 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	9.86E-10	0.8	Ci/m^3	2.96E-06	Ci/m^3	6.32E-06	Ci	2.13E+00	m^3
Ni-59	0.00E+00	22	Ci/m^3	0.00E+00	Ci/m^3	0.00E+00	Ci	3.74E+00	m^3
Nb-94	4.54E-08	0.02	Ci/m^3	3.40E-06	Ci/m^3	1.27E-05	Ci	3.74E+00	m^3
Tc-99	9.80E-04	0.30	Ci/m^3	1.10E+00	Ci/m^3	4.12E+00	Ci	3.74E+00	m^3
I-129	5.95E-07	0.008	Ci/m^3	1.79E-05	Ci/m^3	6.68E-05	Ci	3.74E+00	m^3
Np-237	1.59E-04	10	nCi/g	5.98E+00	nCi/g	1.07E-02	Ci	1.79E+06	g
Pu-238	1.94E-03	10	nCi/g	7.27E+01	nCi/g	1.30E-01	Ci	1.79E+06	g
Pu-239	6.10E-03	10	nCi/g	2.29E+02	nCi/g	4.10E-01	Ci	1.79E+06	g
Pu-240	2.37E-03	10	nCi/g	8.88E+01	nCi/g	1.59E-01	Ci	1.79E+06	g
Pu-241	1.98E-11	350	nCi/g	2.60E-05	nCi/g	4.66E-08	Ci	1.79E+06	g
Pu-242	1.42E-06	10	nCi/g	5.34E-02	nCi/g	9.57E-05	Ci	1.79E+06	g
Pu-244	2.78E-14	10	nCi/g	1.04E-09	nCi/g	1.87E-12	Ci	1.79E+06	g
Am-241	9.62E-03	10	nCi/g	3.61E+02	nCi/g	6.47E-01	Ci	1.79E+06	g
Am-242m	4.08E-07	10	nCi/g	1.53E-02	nCi/g	2.74E-05	Ci	1.79E+06	g
Am-243	1.20E-06	10	nCi/g	4.51E-02	nCi/g	8.09E-05	Ci	1.79E+06	g
Cm-242	1.68E-09	2,000	nCi/g	1.26E-02	nCi/g	2.26E-05	Ci	1.79E+06	g
Cm-243	7.18E-12	10	nCi/g	2.70E-07	nCi/g	4.83E-10	Ci	1.79E+06	g
Cm-244	4.17E-13	10	nCi/g	1.57E-08	nCi/g	2.81E-11	Ci	1.79E+06	g
Cm-245	6.17E-10	10	nCi/g	2.32E-05	nCi/g	4.15E-08	Ci	1.79E+06	g
Cm-246	1.37E-11	10	nCi/g	5.15E-07	nCi/g	9.23E-10	Ci	1.79E+06	g
Cm-247	5.22E-18	10	nCi/g	1.96E-13	nCi/g	3.51E-16	Ci	1.79E+06	g
Cm-248	1.66E-18	10	nCi/g	6.24E-14	nCi/g	1.12E-16	Ci	1.79E+06	g
Cf-249	1.47E-18	10	nCi/g	5.52E-14	nCi/g	9.89E-17	Ci	1.79E+06	g
Cf-250	1.72E-28	10	nCi/g	6.46E-24	nCi/g	1.16E-26	Ci	1.79E+06	g
Cf-251	6.36E-21	10	nCi/g	2.39E-16	nCi/g	4.28E-19	Ci	1.79E+06	g
Sum of Fractions	**2.12E-02**								

Table 6-6. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	0.00E+00	3.5	Ci/m^3	0.00E+00	Ci/m^3	0.00E+00	Ci	2.13E+00	m^3
Sr-90	1.26E-03	0.04	Ci/m^3	1.89E-01	Ci/m^3	4.02E-01	Ci	2.13E+00	m^3
Cs-137	1.00E-04	1	Ci/m^3	3.76E-01	Ci/m^3	8.02E-01	Ci	2.13E+00	m^3
Sum of Fractions	**1.36E-03**								

Table 6-7. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario at the Calcine Solids Storage Facility 2 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	3.35E-10	0.8	Ci/m^3	1.01E-06	Ci/m^3	3.76E-06	Ci	3.74E+00	m^3
Ni-59	2.90E-07	22	Ci/m^3	2.39E-02	Ci/m^3	8.95E-02	Ci	3.74E+00	m^3
Nb-94	2.54E-04	0.02	Ci/m^3	1.91E-02	Ci/m^3	7.13E-02	Ci	3.74E+00	m^3
Tc-99	8.02E-04	0.30	Ci/m^3	9.03E-01	Ci/m^3	3.38E+00	Ci	3.74E+00	m^3
I-129	4.91E-07	0.008	Ci/m^3	1.47E-05	Ci/m^3	5.51E-05	Ci	3.74E+00	m^3
Np-237	4.80E-05	10	nCi/g	1.80E+00	nCi/g	8.28E-03	Ci	4.60E+06	g
Pu-238	8.78E-03	10	nCi/g	3.30E+02	nCi/g	1.52E+00	Ci	4.60E+06	g
Pu-239	4.59E-03	10	nCi/g	1.72E+02	nCi/g	7.92E-01	Ci	4.60E+06	g
Pu-240	3.52E-03	10	nCi/g	1.32E+02	nCi/g	6.07E-01	Ci	4.60E+06	g
Pu-241	2.76E-10	350	nCi/g	3.63E-04	nCi/g	1.67E-06	Ci	4.60E+06	g
Pu-242	8.57E-06	10	nCi/g	3.22E-01	nCi/g	1.48E-03	Ci	4.60E+06	g
Pu-244	1.34E-14	10	nCi/g	5.05E-10	nCi/g	2.32E-12	Ci	4.60E+06	g
Am-241	1.71E-02	10	nCi/g	6.42E+02	nCi/g	2.95E+00	Ci	4.60E+06	g
Am-242m	2.18E-06	10	nCi/g	8.18E-02	nCi/g	3.76E-04	Ci	4.60E+06	g
Am-243	2.06E-06	10	nCi/g	7.73E-02	nCi/g	3.55E-04	Ci	4.60E+06	g
Cm-242	8.99E-09	2,000	nCi/g	6.75E-02	nCi/g	3.10E-04	Ci	4.60E+06	g

Table 6-7 (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-243	9.74E-11	10	nCi/g	3.65E-06	nCi/g	1.68E-08	Ci	4.60E+06	g
Cm-244	5.09E-12	10	nCi/g	1.91E-07	nCi/g	8.78E-10	Ci	4.60E+06	g
Cm-245	9.20E-09	10	nCi/g	3.45E-04	nCi/g	1.59E-06	Ci	4.60E+06	g
Cm-246	9.51E-10	10	nCi/g	3.57E-05	nCi/g	1.64E-07	Ci	4.60E+06	g
Cm-247	1.55E-15	10	nCi/g	5.83E-11	nCi/g	2.68E-13	Ci	4.60E+06	g
Cm-248	2.18E-15	10	nCi/g	8.17E-11	nCi/g	3.76E-13	Ci	4.60E+06	g
Cf-249	6.18E-15	10	nCi/g	2.32E-10	nCi/g	1.07E-12	Ci	4.60E+06	g
Cf-250	3.97E-24	10	nCi/g	1.49E-19	nCi/g	6.85E-22	Ci	4.60E+06	g
Cf-251	1.78E-16	10	nCi/g	6.70E-12	nCi/g	3.08E-14	Ci	4.60E+06	g
Sum of Fractions	**3.51E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	6.13E-06	3.5	Ci/m^3	8.05E-02	Ci/m^3	3.01E-01	Ci	3.74E+00	m^3
Sr-90	7.17E-04	0.04	Ci/m^3	1.08E-01	Ci/m^3	4.02E-01	Ci	3.74E+00	m^3
Cs-137	5.39E-05	1	Ci/m^3	2.02E-01	Ci/m^3	7.56E-01	Ci	3.74E+00	m^3
Sum of Fractions	**7.77E-04**								

Table 6-8. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario at the Calcined Solids Storage Facility 3 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	2.21E-13	0.8	Ci/m^3	6.62E-10	Ci/m^3	2.48E-09	Ci	3.74E+00	m^3
Ni-59	6.52E-07	22	Ci/m^3	5.38E-02	Ci/m^3	2.01E-01	Ci	3.74E+00	m^3
Nb-94	5.69E-04	0.02	Ci/m^3	4.27E-02	Ci/m^3	1.60E-01	Ci	3.74E+00	m^3
Tc-99	5.86E-04	0.30	Ci/m^3	6.59E-01	Ci/m^3	2.47E+00	Ci	3.74E+00	m^3
I-129	3.63E-07	0.008	Ci/m^3	1.09E-05	Ci/m^3	4.08E-05	Ci	3.74E+00	m^3
Np-237	1.52E-04	10	nCi/g	5.71E+00	nCi/g	2.76E-02	Ci	4.83E+06	g

Table 6-8. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-238	1.30E-02	10	nCi/g	4.89E+02	nCi/g	2.36E+00	Ci	4.83E+06	g
Pu-239	8.13E-03	10	nCi/g	3.05E+02	nCi/g	1.47E+00	Ci	4.83E+06	g
Pu-240	5.66E-03	10	nCi/g	2.13E+02	nCi/g	1.03E+00	Ci	4.83E+06	g
Pu-241	5.76E-10	350	nCi/g	7.57E-04	nCi/g	3.65E-06	Ci	4.83E+06	g
Pu-242	1.51E-05	10	nCi/g	5.65E-01	nCi/g	2.73E-03	Ci	4.83E+06	g
Pu-244	1.50E-14	10	nCi/g	5.64E-10	nCi/g	2.72E-12	Ci	4.83E+06	g
Am-241	2.63E-02	10	nCi/g	9.88E+02	nCi/g	4.77E+00	Ci	4.83E+06	g
Am-242m	4.46E-06	10	nCi/g	1.68E-01	nCi/g	8.08E-04	Ci	4.83E+06	g
Am-243	5.48E-06	10	nCi/g	2.06E-01	nCi/g	9.92E-04	Ci	4.83E+06	g
Cm-242	1.84E-08	2,000	nCi/g	1.38E-01	nCi/g	6.67E-04	Ci	4.83E+06	g
Cm-243	2.05E-10	10	nCi/g	7.69E-06	nCi/g	3.71E-08	Ci	4.83E+06	g
Cm-244	9.96E-12	10	nCi/g	3.74E-07	nCi/g	1.80E-09	Ci	4.83E+06	g
Cm-245	1.95E-08	10	nCi/g	7.30E-04	nCi/g	3.52E-06	Ci	4.83E+06	g
Cm-246	2.04E-09	10	nCi/g	7.64E-05	nCi/g	3.69E-07	Ci	4.83E+06	g
Cm-247	3.32E-15	10	nCi/g	1.25E-10	nCi/g	6.02E-13	Ci	4.83E+06	g
Cm-248	4.66E-15	10	nCi/g	1.75E-10	nCi/g	8.44E-13	Ci	4.83E+06	g
Cf-249	1.32E-14	10	nCi/g	4.96E-10	nCi/g	2.39E-12	Ci	4.83E+06	g
Cf-250	8.49E-24	10	nCi/g	3.19E-19	nCi/g	1.54E-21	Ci	4.83E+06	g
Cf-251	3.82E-16	10	nCi/g	1.43E-11	nCi/g	6.92E-14	Ci	4.83E+06	g
Sum of Fractions	**5.45E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	1.18E-05	3.5	Ci/m^3	1.56E-01	Ci/m^3	5.82E-01	Ci	3.74E+00	m^3
Sr-90	5.77E-04	0.04	Ci/m^3	8.66E-02	Ci/m^3	3.24E-01	Ci	3.74E+00	m^3
Cs-137	4.46E-05	1	Ci/m^3	1.68E-01	Ci/m^3	6.27E-01	Ci	3.74E+00	m^3
Sum of Fractions	**6.34E-04**								

Table 6-9. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario at the Calcined Solids Storage Facility 4 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	2.95E-13	0.8	Ci/m^3	8.87E-10	Ci/m^3	1.42E-09	Ci	1.60E+00	m^3
Ni-59	8.75E-07	22	Ci/m^3	7.22E-02	Ci/m^3	1.16E-01	Ci	1.60E+00	m^3
Nb-94	7.66E-04	0.02	Ci/m^3	5.75E-02	Ci/m^3	9.20E-02	Ci	1.60E+00	m^3
Tc-99	7.82E-04	0.30	Ci/m^3	8.80E-01	Ci/m^3	1.41E+00	Ci	1.60E+00	m^3
I-129	4.85E-07	0.008	Ci/m^3	1.46E-05	Ci/m^3	2.33E-05	Ci	1.60E+00	m^3
Np-237	7.94E-04	10	nCi/g	2.98E+01	nCi/g	6.49E-02	Ci	2.18E+06	g
Pu-238	2.82E-02	10	nCi/g	1.06E+03	nCi/g	2.31E+00	Ci	2.18E+06	g
Pu-239	1.84E-02	10	nCi/g	6.90E+02	nCi/g	1.50E+00	Ci	2.18E+06	g
Pu-240	1.15E-02	10	nCi/g	4.30E+02	nCi/g	9.36E-01	Ci	2.18E+06	g
Pu-241	7.49E-10	350	nCi/g	9.83E-04	nCi/g	2.14E-06	Ci	2.18E+06	g
Pu-242	3.34E-05	10	nCi/g	1.25E+00	nCi/g	2.73E-03	Ci	2.18E+06	g
Pu-244	1.91E-14	10	nCi/g	7.17E-10	nCi/g	1.56E-12	Ci	2.18E+06	g
Am-241	3.84E-02	10	nCi/g	1.44E+03	nCi/g	3.14E+00	Ci	2.18E+06	g
Am-242m	5.69E-06	10	nCi/g	2.14E-01	nCi/g	4.65E-04	Ci	2.18E+06	g
Am-243	7.59E-06	10	nCi/g	2.85E-01	nCi/g	6.20E-04	Ci	2.18E+06	g
Cm-242	2.35E-08	2,000	nCi/g	1.76E-01	nCi/g	3.84E-04	Ci	2.18E+06	g
Cm-243	2.61E-10	10	nCi/g	9.80E-06	nCi/g	2.13E-08	Ci	2.18E+06	g
Cm-244	1.35E-11	10	nCi/g	5.06E-07	nCi/g	1.10E-09	Ci	2.18E+06	g
Cm-245	2.48E-08	10	nCi/g	9.29E-04	nCi/g	2.02E-06	Ci	2.18E+06	g
Cm-246	2.60E-09	10	nCi/g	9.75E-05	nCi/g	2.12E-07	Ci	2.18E+06	g
Cm-247	4.24E-15	10	nCi/g	1.59E-10	nCi/g	3.46E-13	Ci	2.18E+06	g
Cm-248	5.95E-15	10	nCi/g	2.23E-10	nCi/g	4.86E-13	Ci	2.18E+06	g
Cf-249	1.68E-14	10	nCi/g	6.31E-10	nCi/g	1.37E-12	Ci	2.18E+06	g
Cf-250	1.08E-23	10	nCi/g	4.06E-19	nCi/g	8.84E-22	Ci	2.18E+06	g
Cf-251	4.87E-16	10	nCi/g	1.83E-11	nCi/g	3.98E-14	Ci	2.18E+06	g
Sum of Fractions	**9.89E-02**								

Table 6-9. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	1.78E-05	3.5	Ci/m^3	2.33E-01	Ci/m^3	3.74E-01	Ci	1.60E+00	m^3
Sr-90	8.39E-04	0.04	Ci/m^3	1.26E-01	Ci/m^3	2.02E-01	Ci	1.60E+00	m^3
Cs-137	6.00E-05	1	Ci/m^3	2.25E-01	Ci/m^3	3.60E-01	Ci	1.60E+00	m^3
Sum of Fractions	**9.17E-04**								

Table 6-10. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario at the Calcined Solids Storage Facility 5 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	3.07E-13	0.8	Ci/m^3	9.21E-10	Ci/m^3	3.06E-09	Ci	3.32E+00	m^3
Ni-59	9.05E-07	22	Ci/m^3	7.47E-02	Ci/m^3	2.48E-01	Ci	3.32E+00	m^3
Nb-94	7.96E-04	0.02	Ci/m^3	5.97E-02	Ci/m^3	1.98E-01	Ci	3.32E+00	m^3
Tc-99	8.23E-04	0.30	Ci/m^3	9.27E-01	Ci/m^3	3.08E+00	Ci	3.32E+00	m^3
I-129	5.09E-07	0.008	Ci/m^3	1.53E-05	Ci/m^3	5.07E-05	Ci	3.32E+00	m^3
Np-237	7.28E-04	10	nCi/g	2.73E+01	nCi/g	1.25E-01	Ci	4.58E+06	g
Pu-238	2.62E-02	10	nCi/g	9.82E+02	nCi/g	4.50E+00	Ci	4.58E+06	g
Pu-239	1.70E-02	10	nCi/g	6.37E+02	nCi/g	2.92E+00	Ci	4.58E+06	g
Pu-240	1.16E-02	10	nCi/g	4.35E+02	nCi/g	1.99E+00	Ci	4.58E+06	g
Pu-241	7.66E-10	350	nCi/g	1.01E-03	nCi/g	4.61E-06	Ci	4.58E+06	g
Pu-242	2.84E-05	10	nCi/g	1.06E+00	nCi/g	4.88E-03	Ci	4.58E+06	g
Pu-244	1.95E-14	10	nCi/g	7.33E-10	nCi/g	3.36E-12	Ci	4.58E+06	g
Am-241	3.50E-02	10	nCi/g	1.31E+03	nCi/g	6.01E+00	Ci	4.58E+06	g
Am-242m	5.81E-06	10	nCi/g	2.18E-01	nCi/g	9.99E-04	Ci	4.58E+06	g
Am-243	6.33E-06	10	nCi/g	2.38E-01	nCi/g	1.09E-03	Ci	4.58E+06	g
Cm-242	2.40E-08	2,000	nCi/g	1.80E-01	nCi/g	8.25E-04	Ci	4.58E+06	g

Table 6-10. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-243	2.67E-10	10	nCi/g	1.00E-05	nCi/g	4.59E-08	Ci	4.58E+06	g
Cm-244	1.20E-11	10	nCi/g	4.51E-07	nCi/g	2.06E-09	Ci	4.58E+06	g
Cm-245	2.53E-08	10	nCi/g	9.51E-04	nCi/g	4.36E-06	Ci	4.58E+06	g
Cm-246	2.65E-09	10	nCi/g	9.95E-05	nCi/g	4.56E-07	Ci	4.58E+06	g
Cm-247	4.33E-15	10	nCi/g	1.63E-10	nCi/g	7.45E-13	Ci	4.58E+06	g
Cm-248	6.10E-15	10	nCi/g	2.29E-10	nCi/g	1.05E-12	Ci	4.58E+06	g
Cf-249	1.72E-14	10	nCi/g	6.46E-10	nCi/g	2.96E-12	Ci	4.58E+06	g
Cf-250	1.10E-23	10	nCi/g	4.15E-19	nCi/g	1.90E-21	Ci	4.58E+06	g
Cf-251	4.96E-16	10	nCi/g	1.86E-11	nCi/g	8.52E-14	Ci	4.58E+06	g
Sum of Fractions	**9.21E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	1.51E-05	3.5	Ci/m^3	1.98E-01	Ci/m^3	6.58E-01	Ci	3.32E+00	m^3
Sr-90	8.70E-04	0.04	Ci/m^3	1.31E-01	Ci/m^3	4.33E-01	Ci	3.32E+00	m^3
Cs-137	6.22E-05	1	Ci/m^3	2.33E-01	Ci/m^3	7.75E-01	Ci	3.32E+00	m^3
Sum of Fractions	**9.47E-04**								

Table 6-11. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario at the Calcined Solids Storage Facility 6 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	1.17E-13	0.8	Ci/m^3	3.50E-10	Ci/m^3	1.43E-09	Ci	4.08E+00	m^3
Ni-59	3.43E-07	22	Ci/m^3	2.83E-02	Ci/m^3	1.16E-01	Ci	4.08E+00	m^3
Nb-94	3.02E-04	0.02	Ci/m^3	2.27E-02	Ci/m^3	9.24E-02	Ci	4.08E+00	m^3
Tc-99	3.11E-04	0.30	Ci/m^3	3.50E-01	Ci/m^3	1.43E+00	Ci	4.08E+00	m^3
I-129	1.92E-07	0.008	Ci/m^3	5.76E-06	Ci/m^3	2.35E-05	Ci	4.08E+00	m^3
Np-237	1.68E-04	10	nCi/g	6.32E+00	nCi/g	3.17E-02	Ci	5.02E+06	g

Table 6-11. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-238	6.43E-03	10	nCi/g	2.41E+02	nCi/g	1.21E+00	Ci	5.02E+06	g
Pu-239	1.00E-02	10	nCi/g	3.76E+02	nCi/g	1.89E+00	Ci	5.02E+06	g
Pu-240	5.25E-03	10	nCi/g	1.97E+02	nCi/g	9.87E-01	Ci	5.02E+06	g
Pu-241	3.28E-10	350	nCi/g	4.30E-04	nCi/g	2.16E-06	Ci	5.02E+06	g
Pu-242	1.14E-05	10	nCi/g	4.28E-01	nCi/g	2.15E-03	Ci	5.02E+06	g
Pu-244	8.34E-15	10	nCi/g	3.13E-10	nCi/g	1.57E-12	Ci	5.02E+06	g
Am-241	9.11E-03	10	nCi/g	3.42E+02	nCi/g	1.71E+00	Ci	5.02E+06	g
Am-242m	2.48E-06	10	nCi/g	9.30E-02	nCi/g	4.66E-04	Ci	5.02E+06	g
Am-243	3.28E-06	10	nCi/g	1.23E-01	nCi/g	6.17E-04	Ci	5.02E+06	g
Cm-242	1.02E-08	2,000	nCi/g	7.67E-02	nCi/g	3.85E-04	Ci	5.02E+06	g
Cm-243	1.14E-10	10	nCi/g	4.28E-06	nCi/g	2.14E-08	Ci	5.02E+06	g
Cm-244	5.76E-12	10	nCi/g	2.16E-07	nCi/g	1.08E-09	Ci	5.02E+06	g
Cm-245	1.08E-08	10	nCi/g	4.05E-04	nCi/g	2.03E-06	Ci	5.02E+06	g
Cm-246	1.13E-09	10	nCi/g	4.23E-05	nCi/g	2.12E-07	Ci	5.02E+06	g
Cm-247	1.84E-15	10	nCi/g	6.92E-11	nCi/g	3.47E-13	Ci	5.02E+06	g
Cm-248	2.59E-15	10	nCi/g	9.70E-11	nCi/g	4.87E-13	Ci	5.02E+06	g
Cf-249	7.35E-15	10	nCi/g	2.76E-10	nCi/g	1.38E-12	Ci	5.02E+06	g
Cf-250	4.70E-24	10	nCi/g	1.76E-19	nCi/g	8.84E-22	Ci	5.02E+06	g
Cf-251	2.12E-16	10	nCi/g	7.96E-12	nCi/g	3.99E-14	Ci	5.02E+06	g
Sum of Fractions	**3.16E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	3.91E-06	3.5	Ci/m^3	5.13E-02	Ci/m^3	2.09E-01	Ci	4.08E+00	m^3
Sr-90	3.08E-04	0.04	Ci/m^3	4.62E-02	Ci/m^3	1.89E-01	Ci	4.08E+00	m^3
Cs-137	2.36E-05	1	Ci/m^3	8.86E-02	Ci/m^3	3.61E-01	Ci	4.08E+00	m^3
Sum of Fractions	**3.36E-04**								

Table 6-12. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario for the Calcined Solids Storage Facility 2 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	7.47E-12	0.8	Ci/m^3	1.01E-06	Ci/m^3	1.10E-07	Ci	1.09E-01	m^3
Ni-59	6.46E-09	22	Ci/m^3	2.39E-02	Ci/m^3	2.61E-03	Ci	1.09E-01	m^3
Nb-94	5.67E-06	0.02	Ci/m^3	1.91E-02	Ci/m^3	2.08E-03	Ci	1.09E-01	m^3
Tc-99	1.79E-05	0.30	Ci/m^3	9.03E-01	Ci/m^3	9.84E-02	Ci	1.09E-01	m^3
I-129	1.09E-08	0.008	Ci/m^3	1.47E-05	Ci/m^3	1.61E-06	Ci	1.09E-01	m^3
Np-237	1.07E-06	10	nCi/g	1.80E+00	nCi/g	2.41E-04	Ci	1.34E+05	g
Pu-238	1.96E-04	10	nCi/g	3.30E+02	nCi/g	4.42E-02	Ci	1.34E+05	g
Pu-239	1.02E-04	10	nCi/g	1.72E+02	nCi/g	2.31E-02	Ci	1.34E+05	g
Pu-240	7.84E-05	10	nCi/g	1.32E+02	nCi/g	1.77E-02	Ci	1.34E+05	g
Pu-241	6.16E-12	350	nCi/g	3.63E-04	nCi/g	4.86E-08	Ci	1.34E+05	g
Pu-242	1.91E-07	10	nCi/g	3.22E-01	nCi/g	4.31E-05	Ci	1.34E+05	g
Pu-244	3.00E-16	10	nCi/g	5.05E-10	nCi/g	6.77E-14	Ci	1.34E+05	g
Am-241	3.81E-04	10	nCi/g	6.42E+02	nCi/g	8.61E-02	Ci	1.34E+05	g
Am-242m	4.86E-08	10	nCi/g	8.18E-02	nCi/g	1.10E-05	Ci	1.34E+05	g
Am-243	4.59E-08	10	nCi/g	7.73E-02	nCi/g	1.04E-05	Ci	1.34E+05	g
Cm-242	2.00E-10	2,000	nCi/g	6.75E-02	nCi/g	9.05E-06	Ci	1.34E+05	g
Cm-243	2.17E-12	10	nCi/g	3.65E-06	nCi/g	4.90E-10	Ci	1.34E+05	g
Cm-244	1.13E-13	10	nCi/g	1.91E-07	nCi/g	2.56E-11	Ci	1.34E+05	g
Cm-245	2.05E-10	10	nCi/g	3.45E-04	nCi/g	4.63E-08	Ci	1.34E+05	g
Cm-246	2.12E-11	10	nCi/g	3.57E-05	nCi/g	4.79E-09	Ci	1.34E+05	g
Cm-247	3.46E-17	10	nCi/g	5.83E-11	nCi/g	7.81E-15	Ci	1.34E+05	g
Cm-248	4.85E-17	10	nCi/g	8.17E-11	nCi/g	1.10E-14	Ci	1.34E+05	g
Cf-249	1.38E-16	10	nCi/g	2.32E-10	nCi/g	3.11E-14	Ci	1.34E+05	g
Cf-250	8.85E-26	10	nCi/g	1.49E-19	nCi/g	2.00E-23	Ci	1.34E+05	g
Cf-251	3.98E-18	10	nCi/g	6.70E-12	nCi/g	8.98E-16	Ci	1.34E+05	g
Sum of Fractions	**7.83E-04**								

Table 6-12. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	1.37E-07	3.5	Ci/m^3	8.05E-02	Ci/m^3	8.77E-03	Ci	1.09E-01	m^3
Sr-90	1.60E-05	0.04	Ci/m^3	1.08E-01	Ci/m^3	1.17E-02	Ci	1.09E-01	m^3
Cs-137	1.20E-06	1	Ci/m^3	2.02E-01	Ci/m^3	2.21E-02	Ci	1.09E-01	m^3
Sum of Fractions	**1.73E-05**								

Table 6-13. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario for the Calcined Solids Storage Facility 3 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	4.92E-15	0.8	Ci/m^3	6.62E-10	Ci/m^3	7.22E-11	Ci	1.09E-01	m^3
Ni-59	1.45E-08	22	Ci/m^3	5.38E-02	Ci/m^3	5.86E-03	Ci	1.09E-01	m^3
Nb-94	1.27E-05	0.02	Ci/m^3	4.27E-02	Ci/m^3	4.66E-03	Ci	1.09E-01	m^3
Tc-99	1.31E-05	0.30	Ci/m^3	6.59E-01	Ci/m^3	7.19E-02	Ci	1.09E-01	m^3
I-129	8.10E-09	0.008	Ci/m^3	1.09E-05	Ci/m^3	1.19E-06	Ci	1.09E-01	m^3
Np-237	3.39E-06	10	nCi/g	5.71E+00	nCi/g	8.03E-04	Ci	1.41E+05	g
Pu-238	2.91E-04	10	nCi/g	4.89E+02	nCi/g	6.88E-02	Ci	1.41E+05	g
Pu-239	1.81E-04	10	nCi/g	3.05E+02	nCi/g	4.29E-02	Ci	1.41E+05	g
Pu-240	1.26E-04	10	nCi/g	2.13E+02	nCi/g	2.99E-02	Ci	1.41E+05	g
Pu-241	1.28E-11	350	nCi/g	7.57E-04	nCi/g	1.06E-07	Ci	1.41E+05	g
Pu-242	3.36E-07	10	nCi/g	5.65E-01	nCi/g	7.95E-05	Ci	1.41E+05	g
Pu-244	3.35E-16	10	nCi/g	5.64E-10	nCi/g	7.93E-14	Ci	1.41E+05	g
Am-241	5.87E-04	10	nCi/g	9.88E+02	nCi/g	1.39E-01	Ci	1.41E+05	g
Am-242m	9.95E-08	10	nCi/g	1.68E-01	nCi/g	2.36E-05	Ci	1.41E+05	g
Am-243	1.22E-07	10	nCi/g	2.06E-01	nCi/g	2.89E-05	Ci	1.41E+05	g
Cm-242	4.11E-10	2,000	nCi/g	1.38E-01	nCi/g	1.94E-05	Ci	1.41E+05	g
Cm-243	4.57E-12	10	nCi/g	7.69E-06	nCi/g	1.08E-09	Ci	1.41E+05	g

Table 6-13. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-244	2.22E-13	10	nCi/g	3.74E-07	nCi/g	5.26E-11	Ci	1.41E+05	g
Cm-245	4.34E-10	10	nCi/g	7.30E-04	nCi/g	1.03E-07	Ci	1.41E+05	g
Cm-246	4.54E-11	10	nCi/g	7.64E-05	nCi/g	1.07E-08	Ci	1.41E+05	g
Cm-247	7.41E-17	10	nCi/g	1.25E-10	nCi/g	1.75E-14	Ci	1.41E+05	g
Cm-248	1.04E-16	10	nCi/g	1.75E-10	nCi/g	2.46E-14	Ci	1.41E+05	g
Cf-249	2.95E-16	10	nCi/g	4.96E-10	nCi/g	6.98E-14	Ci	1.41E+05	g
Cf-250	1.89E-25	10	nCi/g	3.19E-19	nCi/g	4.48E-23	Ci	1.41E+05	g
Cf-251	8.51E-18	10	nCi/g	1.43E-11	nCi/g	2.01E-15	Ci	1.41E+05	g
Sum of Fractions	**1.21E-03**								
Table 2 of 10 CFR 61.55									
Ni-63	2.64E-07	3.5	Ci/m^3	1.56E-01	Ci/m^3	1.70E-02	Ci	1.09E-01	m^3
Sr-90	1.29E-05	0.04	Ci/m^3	8.66E-02	Ci/m^3	9.45E-03	Ci	1.09E-01	m^3
Cs-137	9.95E-07	1	Ci/m^3	1.68E-01	Ci/m^3	1.83E-02	Ci	1.09E-01	m^3
Sum of Fractions	**1.41E-05**								

Table 6-14. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario for the Calcined Solids Storage Facility 4 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	6.59E-15	0.8	Ci/m^3	8.87E-10	Ci/m^3	9.67E-11	Ci	1.09E-01	m^3
Ni-59	1.95E-08	22	Ci/m^3	7.22E-02	Ci/m^3	7.87E-03	Ci	1.09E-01	m^3
Nb-94	1.71E-05	0.02	Ci/m^3	5.75E-02	Ci/m^3	6.27E-03	Ci	1.09E-01	m^3
Tc-99	1.74E-05	0.30	Ci/m^3	8.80E-01	Ci/m^3	9.59E-02	Ci	1.09E-01	m^3
I-129	1.08E-08	0.008	Ci/m^3	1.46E-05	Ci/m^3	1.59E-06	Ci	1.09E-01	m^3
Np-237	1.77E-05	10	nCi/g	2.98E+01	nCi/g	4.42E-03	Ci	1.48E+05	g
Pu-238	6.29E-04	10	nCi/g	1.06E+03	nCi/g	1.57E-01	Ci	1.48E+05	g
Pu-239	4.10E-04	10	nCi/g	6.90E+02	nCi/g	1.02E-01	Ci	1.48E+05	g

Table 6-14. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-240	2.55E-04	10	nCi/g	4.30E+02	nCi/g	6.37E-02	Ci	1.48E+05	g
Pu-241	1.67E-11	350	nCi/g	9.83E-04	nCi/g	1.46E-07	Ci	1.48E+05	g
Pu-242	7.44E-07	10	nCi/g	1.25E+00	nCi/g	1.86E-04	Ci	1.48E+05	g
Pu-244	4.26E-16	10	nCi/g	7.17E-10	nCi/g	1.06E-13	Ci	1.48E+05	g
Am-241	8.56E-04	10	nCi/g	1.44E+03	nCi/g	2.14E-01	Ci	1.48E+05	g
Am-242m	1.27E-07	10	nCi/g	2.14E-01	nCi/g	3.17E-05	Ci	1.48E+05	g
Am-243	1.69E-07	10	nCi/g	2.85E-01	nCi/g	4.22E-05	Ci	1.48E+05	g
Cm-242	5.23E-10	2,000	nCi/g	1.76E-01	nCi/g	2.61E-05	Ci	1.48E+05	g
Cm-243	5.82E-12	10	nCi/g	9.80E-06	nCi/g	1.45E-09	Ci	1.48E+05	g
Cm-244	3.01E-13	10	nCi/g	5.06E-07	nCi/g	7.51E-11	Ci	1.48E+05	g
Cm-245	5.52E-10	10	nCi/g	9.29E-04	nCi/g	1.38E-07	Ci	1.48E+05	g
Cm-246	5.79E-11	10	nCi/g	9.75E-05	nCi/g	1.45E-08	Ci	1.48E+05	g
Cm-247	9.44E-17	10	nCi/g	1.59E-10	nCi/g	2.36E-14	Ci	1.48E+05	g
Cm-248	1.33E-16	10	nCi/g	2.23E-10	nCi/g	3.31E-14	Ci	1.48E+05	g
Cf-249	3.75E-16	10	nCi/g	6.31E-10	nCi/g	9.36E-14	Ci	1.48E+05	g
Cf-250	2.41E-25	10	nCi/g	4.06E-19	nCi/g	6.02E-23	Ci	1.48E+05	g
Cf-251	1.08E-17	10	nCi/g	1.83E-11	nCi/g	2.71E-15	Ci	1.48E+05	g
Sum of Fractions	**2.20E-03**								
Table 2 of 10 CFR 61.55									
Ni-63	3.96E-07	3.5	Ci/m^3	2.33E-01	Ci/m^3	2.54E-02	Ci	1.09E-01	m^3
Sr-90	1.87E-05	0.04	Ci/m^3	1.26E-01	Ci/m^3	1.37E-02	Ci	1.09E-01	m^3
Cs-137	1.34E-06	1	Ci/m^3	2.25E-01	Ci/m^3	2.45E-02	Ci	1.09E-01	m^3
Sum of Fractions	**2.04E-05**								

Table 6-15. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario for the Calcined Solids Storage Facility 5 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	6.84E-15	0.8	Ci/m^3	9.21E-10	Ci/m^3	1.00E-10	Ci	1.09E-01	m^3
Ni-59	2.02E-08	22	Ci/m^3	7.47E-02	Ci/m^3	8.15E-03	Ci	1.09E-01	m^3
Nb-94	1.77E-05	0.02	Ci/m^3	5.97E-02	Ci/m^3	6.51E-03	Ci	1.09E-01	m^3
Tc-99	1.83E-05	0.30	Ci/m^3	9.27E-01	Ci/m^3	1.01E-01	Ci	1.09E-01	m^3
I-129	1.13E-08	0.008	Ci/m^3	1.53E-05	Ci/m^3	1.67E-06	Ci	1.09E-01	m^3
Np-237	1.62E-05	10	nCi/g	2.73E+01	nCi/g	4.11E-03	Ci	1.50E+05	g
Pu-238	5.84E-04	10	nCi/g	9.82E+02	nCi/g	1.48E-01	Ci	1.50E+05	g
Pu-239	3.78E-04	10	nCi/g	6.37E+02	nCi/g	9.58E-02	Ci	1.50E+05	g
Pu-240	2.59E-04	10	nCi/g	4.35E+02	nCi/g	6.55E-02	Ci	1.50E+05	g
Pu-241	1.71E-11	350	nCi/g	1.01E-03	nCi/g	1.51E-07	Ci	1.50E+05	g
Pu-242	6.32E-07	10	nCi/g	1.06E+00	nCi/g	1.60E-04	Ci	1.50E+05	g
Pu-244	4.36E-16	10	nCi/g	7.33E-10	nCi/g	1.10E-13	Ci	1.50E+05	g
Am-241	7.80E-04	10	nCi/g	1.31E+03	nCi/g	1.98E-01	Ci	1.50E+05	g
Am-242m	1.30E-07	10	nCi/g	2.18E-01	nCi/g	3.28E-05	Ci	1.50E+05	g
Am-243	1.41E-07	10	nCi/g	2.38E-01	nCi/g	3.57E-05	Ci	1.50E+05	g
Cm-242	5.35E-10	2,000	nCi/g	1.80E-01	nCi/g	2.71E-05	Ci	1.50E+05	g
Cm-243	5.96E-12	10	nCi/g	1.00E-05	nCi/g	1.51E-09	Ci	1.50E+05	g
Cm-244	2.68E-13	10	nCi/g	4.51E-07	nCi/g	6.78E-11	Ci	1.50E+05	g
Cm-245	5.65E-10	10	nCi/g	9.51E-04	nCi/g	1.43E-07	Ci	1.50E+05	g
Cm-246	5.91E-11	10	nCi/g	9.95E-05	nCi/g	1.50E-08	Ci	1.50E+05	g
Cm-247	9.66E-17	10	nCi/g	1.63E-10	nCi/g	2.45E-14	Ci	1.50E+05	g
Cm-248	1.36E-16	10	nCi/g	2.29E-10	nCi/g	3.45E-14	Ci	1.50E+05	g
Cf-249	3.84E-16	10	nCi/g	6.46E-10	nCi/g	9.72E-14	Ci	1.50E+05	g
Cf-250	2.46E-25	10	nCi/g	4.15E-19	nCi/g	6.24E-23	Ci	1.50E+05	g
Cf-251	1.11E-17	10	nCi/g	1.86E-11	nCi/g	2.80E-15	Ci	1.50E+05	g
Sum of Fractions	**2.05E-03**								
Table 2 of 10 CFR 61.55									
Ni-63	3.36E-07	3.5	Ci/m^3	1.98E-01	Ci/m^3	2.16E-02	Ci	1.09E-01	m^3
Sr-90	1.94E-05	0.04	Ci/m^3	1.31E-01	Ci/m^3	1.42E-02	Ci	1.09E-01	m^3
Cs-137	1.39E-06	1	Ci/m^3	2.33E-01	Ci/m^3	2.54E-02	Ci	1.09E-01	m^3
Sum of Fractions	**2.11E-05**								

Table 6-16. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the acute intruder drilling scenario for the Calcined Solids Storage Facility 6 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	2.60E-15	0.8	Ci/m^3	3.50E-10	Ci/m^3	3.82E-11	Ci	1.09E-01	m^3
Ni-59	7.65E-09	22	Ci/m^3	2.83E-02	Ci/m^3	3.09E-03	Ci	1.09E-01	m^3
Nb-94	6.73E-06	0.02	Ci/m^3	2.27E-02	Ci/m^3	2.47E-03	Ci	1.09E-01	m^3
Tc-99	6.93E-06	0.30	Ci/m^3	3.50E-01	Ci/m^3	3.82E-02	Ci	1.09E-01	m^3
I-129	4.28E-09	0.008	Ci/m^3	5.76E-06	Ci/m^3	6.28E-07	Ci	1.09E-01	m^3
Np-237	3.75E-06	10	nCi/g	6.32E+00	nCi/g	8.47E-04	Ci	1.34E+05	g
Pu-238	1.43E-04	10	nCi/g	2.41E+02	nCi/g	3.24E-02	Ci	1.34E+05	g
Pu-239	2.24E-04	10	nCi/g	3.76E+02	nCi/g	5.05E-02	Ci	1.34E+05	g
Pu-240	1.17E-04	10	nCi/g	1.97E+02	nCi/g	2.64E-02	Ci	1.34E+05	g
Pu-241	7.30E-12	350	nCi/g	4.30E-04	nCi/g	5.77E-08	Ci	1.34E+05	g
Pu-242	2.54E-07	10	nCi/g	4.28E-01	nCi/g	5.74E-05	Ci	1.34E+05	g
Pu-244	1.86E-16	10	nCi/g	3.13E-10	nCi/g	4.20E-14	Ci	1.34E+05	g
Am-241	2.03E-04	10	nCi/g	3.42E+02	nCi/g	4.58E-02	Ci	1.34E+05	g
Am-242m	5.52E-08	10	nCi/g	9.30E-02	nCi/g	1.25E-05	Ci	1.34E+05	g
Am-243	7.31E-08	10	nCi/g	1.23E-01	nCi/g	1.65E-05	Ci	1.34E+05	g
Cm-242	2.28E-10	2,000	nCi/g	7.67E-02	nCi/g	1.03E-05	Ci	1.34E+05	g
Cm-243	2.54E-12	10	nCi/g	4.28E-06	nCi/g	5.73E-10	Ci	1.34E+05	g
Cm-244	1.28E-13	10	nCi/g	2.16E-07	nCi/g	2.90E-11	Ci	1.34E+05	g
Cm-245	2.41E-10	10	nCi/g	4.05E-04	nCi/g	5.43E-08	Ci	1.34E+05	g
Cm-246	2.51E-11	10	nCi/g	4.23E-05	nCi/g	5.67E-09	Ci	1.34E+05	g
Cm-247	4.11E-17	10	nCi/g	6.92E-11	nCi/g	9.28E-15	Ci	1.34E+05	g
Cm-248	5.76E-17	10	nCi/g	9.70E-11	nCi/g	1.30E-14	Ci	1.34E+05	g
Cf-249	1.64E-16	10	nCi/g	2.76E-10	nCi/g	3.70E-14	Ci	1.34E+05	g
Cf-250	1.05E-25	10	nCi/g	1.76E-19	nCi/g	2.36E-23	Ci	1.34E+05	g
Cf-251	4.73E-18	10	nCi/g	7.96E-12	nCi/g	1.07E-15	Ci	1.34E+05	g
Sum of Fractions	**7.05E-04**								

Table 6-16. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	8.71E-08	3.5	Ci/m^3	5.13E-02	Ci/m^3	5.60E-03	Ci	1.09E-01	m^3
Sr-90	6.87E-06	0.04	Ci/m^3	4.62E-02	Ci/m^3	5.04E-03	Ci	1.09E-01	m^3
Cs-137	5.26E-07	1	Ci/m^3	8.86E-02	Ci/m^3	9.66E-03	Ci	1.09E-01	m^3
Sum of Fractions	**7.48E-06**								

Table 6-17. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 1 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	2.81E-10	0.8	Ci/m^3	2.96E-06	Ci/m^3	6.32E-06	Ci	2.13E+00	m^3
Ni-59	0.00E+00	22	Ci/m^3	0.00E+00	Ci/m^3	0.00E+00	Ci	2.13E+00	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	1.27E-05	Ci	2.13E+00	m^3
Tc-99	4.89E-04	0.30	Ci/m^3	1.93E+00	Ci/m^3	4.12E+00	Ci	2.13E+00	m^3
I-129	2.97E-07	0.008	Ci/m^3	3.13E-05	Ci/m^3	6.68E-05	Ci	2.13E+00	m^3
Np-237	4.54E-05	10	nCi/g	5.98E+00	nCi/g	1.07E-02	Ci	1.79E+06	g
Pu-238	5.52E-04	10	nCi/g	7.27E+01	nCi/g	1.30E-01	Ci	1.79E+06	g
Pu-239	1.74E-03	10	nCi/g	2.29E+02	nCi/g	4.10E-01	Ci	1.79E+06	g
Pu-240	6.74E-04	10	nCi/g	8.88E+01	nCi/g	1.59E-01	Ci	1.79E+06	g
Pu-241	5.63E-12	350	nCi/g	2.60E-05	nCi/g	4.66E-08	Ci	1.79E+06	g
Pu-242	4.05E-07	10	nCi/g	5.34E-02	nCi/g	9.57E-05	Ci	1.79E+06	g
Pu-244	7.92E-15	10	nCi/g	1.04E-09	nCi/g	1.87E-12	Ci	1.79E+06	g
Am-241	2.74E-03	10	nCi/g	3.61E+02	nCi/g	6.47E-01	Ci	1.79E+06	g
Am-242m	1.16E-07	10	nCi/g	1.53E-02	nCi/g	2.74E-05	Ci	1.79E+06	g
Am-243	3.42E-07	10	nCi/g	4.51E-02	nCi/g	8.09E-05	Ci	1.79E+06	g
Cm-242	4.79E-10	2,000	nCi/g	1.26E-02	nCi/g	2.26E-05	Ci	1.79E+06	g
Cm-243	2.04E-12	10	nCi/g	2.70E-07	nCi/g	4.83E-10	Ci	1.79E+06	g

Table 6-17. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-244	1.19E-13	10	nCi/g	1.57E-08	nCi/g	2.81E-11	Ci	1.79E+06	g
Cm-245	1.76E-10	10	nCi/g	2.32E-05	nCi/g	4.15E-08	Ci	1.79E+06	g
Cm-246	3.90E-12	10	nCi/g	5.15E-07	nCi/g	9.23E-10	Ci	1.79E+06	g
Cm-247	1.49E-18	10	nCi/g	1.96E-13	nCi/g	3.51E-16	Ci	1.79E+06	g
Cm-248	4.74E-19	10	nCi/g	6.24E-14	nCi/g	1.12E-16	Ci	1.79E+06	g
Cf-249	4.18E-19	10	nCi/g	5.52E-14	nCi/g	9.89E-17	Ci	1.79E+06	g
Cf-250	4.90E-29	10	nCi/g	6.46E-24	nCi/g	1.16E-26	Ci	1.79E+06	g
Cf-251	1.81E-21	10	nCi/g	2.39E-16	nCi/g	4.28E-19	Ci	1.79E+06	g
Sum of Fractions	**6.24E-03**								
Table 2 of 10 CFR 61.55									
Ni-63	0.00E+00	3.5	Ci/m^3	0.00E+00	Ci/m^3	0.00E+00	Ci	2.13E+00	m^3
Sr-90	3.58E-04	0.04	Ci/m^3	1.89E-01	Ci/m^3	4.02E-01	Ci	2.13E+00	m^3
Cs-137	2.85E-05	1	Ci/m^3	3.76E-01	Ci/m^3	8.02E-01	Ci	2.13E+00	m^3
Sum of Fractions	**3.86E-04**								

Table 6-18. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 2 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	9.55E-11	0.8	Ci/m^3	1.01E-06	Ci/m^3	3.76E-06	Ci	3.74E+00	m^3
Ni-59	8.25E-08	22	Ci/m^3	2.39E-02	Ci/m^3	8.95E-02	Ci	3.74E+00	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	7.13E-02	Ci	3.74E+00	m^3
Tc-99	2.28E-04	0.30	Ci/m^3	9.03E-01	Ci/m^3	3.38E+00	Ci	3.74E+00	m^3
I-129	1.40E-07	0.008	Ci/m^3	1.47E-05	Ci/m^3	5.51E-05	Ci	3.74E+00	m^3
Np-237	1.37E-05	10	nCi/g	1.80E+00	nCi/g	8.28E-03	Ci	4.60E+06	g
Pu-238	2.50E-03	10	nCi/g	3.30E+02	nCi/g	1.52E+00	Ci	4.60E+06	g

Table 6-18 (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-239	1.31E-03	10	nCi/g	1.72E+02	nCi/g	7.92E-01	Ci	4.60E+06	g
Pu-240	1.00E-03	10	nCi/g	1.32E+02	nCi/g	6.07E-01	Ci	4.60E+06	g
Pu-241	7.86E-11	350	nCi/g	3.63E-04	nCi/g	1.67E-06	Ci	4.60E+06	g
Pu-242	2.44E-06	10	nCi/g	3.22E-01	nCi/g	1.48E-03	Ci	4.60E+06	g
Pu-244	3.83E-15	10	nCi/g	5.05E-10	nCi/g	2.32E-12	Ci	4.60E+06	g
Am-241	4.87E-03	10	nCi/g	6.42E+02	nCi/g	2.95E+00	Ci	4.60E+06	g
Am-242m	6.20E-07	10	nCi/g	8.18E-02	nCi/g	3.76E-04	Ci	4.60E+06	g
Am-243	5.86E-07	10	nCi/g	7.73E-02	nCi/g	3.55E-04	Ci	4.60E+06	g
Cm-242	2.56E-09	2,000	nCi/g	6.75E-02	nCi/g	3.10E-04	Ci	4.60E+06	g
Cm-243	2.77E-11	10	nCi/g	3.65E-06	nCi/g	1.68E-08	Ci	4.60E+06	g
Cm-244	1.45E-12	10	nCi/g	1.91E-07	nCi/g	8.78E-10	Ci	4.60E+06	g
Cm-245	2.62E-09	10	nCi/g	3.45E-04	nCi/g	1.59E-06	Ci	4.60E+06	g
Cm-246	2.71E-10	10	nCi/g	3.57E-05	nCi/g	1.64E-07	Ci	4.60E+06	g
Cm-247	4.42E-16	10	nCi/g	5.83E-11	nCi/g	2.68E-13	Ci	4.60E+06	g
Cm-248	6.20E-16	10	nCi/g	8.17E-11	nCi/g	3.76E-13	Ci	4.60E+06	g
Cf-249	1.76E-15	10	nCi/g	2.32E-10	nCi/g	1.07E-12	Ci	4.60E+06	g
Cf-250	1.13E-24	10	nCi/g	1.49E-19	nCi/g	6.85E-22	Ci	4.60E+06	g
Cf-251	5.08E-17	10	nCi/g	6.70E-12	nCi/g	3.08E-14	Ci	4.60E+06	g
Sum of Fractions	**9.93E-03**								
Table 2 of 10 CFR 61.55									
Ni-63	1.74E-06	3.5	Ci/m^3	8.05E-02	Ci/m^3	3.01E-01	Ci	3.74E+00	m^3
Sr-90	2.04E-04	0.04	Ci/m^3	1.08E-01	Ci/m^3	4.02E-01	Ci	3.74E+00	m^3
Cs-137	1.53E-05	1	Ci/m^3	2.02E-01	Ci/m^3	7.56E-01	Ci	3.74E+00	m^3
Sum of Fractions	**2.21E-04**								

Table 6-19. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 3 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	6.28E-14	0.8	Ci/m^3	6.62E-10	Ci/m^3	2.48E-09	Ci	3.74E+00	m^3
Ni-59	1.86E-07	22	Ci/m^3	5.38E-02	Ci/m^3	2.01E-01	Ci	3.74E+00	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	1.60E-01	Ci	3.74E+00	m^3
Tc-99	1.67E-04	0.30	Ci/m^3	6.59E-01	Ci/m^3	2.47E+00	Ci	3.74E+00	m^3
I-129	1.03E-07	0.008	Ci/m^3	1.09E-05	Ci/m^3	4.08E-05	Ci	3.74E+00	m^3
Np-237	4.33E-05	10	nCi/g	5.71E+00	nCi/g	2.76E-02	Ci	4.83E+06	g
Pu-238	3.71E-03	10	nCi/g	4.89E+02	nCi/g	2.36E+00	Ci	4.83E+06	g
Pu-239	2.32E-03	10	nCi/g	3.05E+02	nCi/g	1.47E+00	Ci	4.83E+06	g
Pu-240	1.61E-03	10	nCi/g	2.13E+02	nCi/g	1.03E+00	Ci	4.83E+06	g
Pu-241	1.64E-10	350	nCi/g	7.57E-04	nCi/g	3.65E-06	Ci	4.83E+06	g
Pu-242	4.29E-06	10	nCi/g	5.65E-01	nCi/g	2.73E-03	Ci	4.83E+06	g
Pu-244	4.28E-15	10	nCi/g	5.64E-10	nCi/g	2.72E-12	Ci	4.83E+06	g
Am-241	7.49E-03	10	nCi/g	9.88E+02	nCi/g	4.77E+00	Ci	4.83E+06	g
Am-242m	1.27E-06	10	nCi/g	1.68E-01	nCi/g	8.08E-04	Ci	4.83E+06	g
Am-243	1.56E-06	10	nCi/g	2.06E-01	nCi/g	9.92E-04	Ci	4.83E+06	g
Cm-242	5.25E-09	2,000	nCi/g	1.38E-01	nCi/g	6.67E-04	Ci	4.83E+06	g
Cm-243	5.83E-11	10	nCi/g	7.69E-06	nCi/g	3.71E-08	Ci	4.83E+06	g
Cm-244	2.84E-12	10	nCi/g	3.74E-07	nCi/g	1.80E-09	Ci	4.83E+06	g
Cm-245	5.54E-09	10	nCi/g	7.30E-04	nCi/g	3.52E-06	Ci	4.83E+06	g
Cm-246	5.80E-10	10	nCi/g	7.64E-05	nCi/g	3.69E-07	Ci	4.83E+06	g
Cm-247	9.46E-16	10	nCi/g	1.25E-10	nCi/g	6.02E-13	Ci	4.83E+06	g
Cm-248	1.33E-15	10	nCi/g	1.75E-10	nCi/g	8.44E-13	Ci	4.83E+06	g
Cf-249	3.76E-15	10	nCi/g	4.96E-10	nCi/g	2.39E-12	Ci	4.83E+06	g
Cf-250	2.42E-24	10	nCi/g	3.19E-19	nCi/g	1.54E-21	Ci	4.83E+06	g
Cf-251	1.09E-16	10	nCi/g	1.43E-11	nCi/g	6.92E-14	Ci	4.83E+06	g
Sum of Fractions	**1.53E-02**								

Table 6-19. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	3.37E-06	3.5	Ci/m^3	1.56E-01	Ci/m^3	5.82E-01	Ci	3.74E+00	m^3
Sr-90	1.64E-04	0.04	Ci/m^3	8.66E-02	Ci/m^3	3.24E-01	Ci	3.74E+00	m^3
Cs-137	1.27E-05	1	Ci/m^3	1.68E-01	Ci/m^3	6.27E-01	Ci	3.74E+00	m^3
Sum of Fractions	***1.80E-04***								

Table 6-20. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 4 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	8.41E-14	0.8	Ci/m^3	8.87E-10	Ci/m^3	1.42E-09	Ci	1.60E+00	m^3
Ni-59	2.49E-07	22	Ci/m^3	7.22E-02	Ci/m^3	1.16E-01	Ci	1.60E+00	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	9.20E-02	Ci	1.60E+00	m^3
Tc-99	2.23E-04	0.30	Ci/m^3	8.80E-01	Ci/m^3	1.41E+00	Ci	1.60E+00	m^3
I-129	1.38E-07	0.008	Ci/m^3	1.46E-05	Ci/m^3	2.33E-05	Ci	1.60E+00	m^3
Np-237	2.26E-04	10	nCi/g	2.98E+01	nCi/g	6.49E+07	nCi	2.18E+06	g
Pu-238	8.04E-03	10	nCi/g	1.06E+03	nCi/g	2.31E+09	nCi	2.18E+06	g
Pu-239	5.24E-03	10	nCi/g	6.90E+02	nCi/g	1.50E+09	nCi	2.18E+06	g
Pu-240	3.26E-03	10	nCi/g	4.30E+02	nCi/g	9.36E+08	nCi	2.18E+06	g
Pu-241	2.13E-10	350	nCi/g	9.83E-04	nCi/g	2.14E+03	nCi	2.18E+06	g
Pu-242	9.51E-06	10	nCi/g	1.25E+00	nCi/g	2.73E+06	nCi	2.18E+06	g
Pu-244	5.44E-15	10	nCi/g	7.17E-10	nCi/g	1.56E-03	nCi	2.18E+06	g
Am-241	1.09E-02	10	nCi/g	1.44E+03	nCi/g	3.14E+09	nCi	2.18E+06	g
Am-242m	1.62E-06	10	nCi/g	2.14E-01	nCi/g	4.65E+05	nCi	2.18E+06	g
Am-243	2.16E-06	10	nCi/g	2.85E-01	nCi/g	6.20E+05	nCi	2.18E+06	g
Cm-242	6.69E-09	2,000	nCi/g	1.76E-01	nCi/g	3.84E+05	nCi	2.18E+06	g
Cm-243	7.43E-11	10	nCi/g	9.80E-06	nCi/g	2.13E+01	nCi	2.18E+06	g
Cm-244	3.84E-12	10	nCi/g	5.06E-07	nCi/g	1.10E+00	nCi	2.18E+06	g

Table 6-20. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-245	7.05E-09	10	nCi/g	9.29E-04	nCi/g	2.02E+03	nCi	2.18E+06	g
Cm-246	7.40E-10	10	nCi/g	9.75E-05	nCi/g	2.12E+02	nCi	2.18E+06	g
Cm-247	1.21E-15	10	nCi/g	1.59E-10	nCi/g	3.46E-04	nCi	2.18E+06	g
Cm-248	1.69E-15	10	nCi/g	2.23E-10	nCi/g	4.86E-04	nCi	2.18E+06	g
Cf-249	4.79E-15	10	nCi/g	6.31E-10	nCi/g	1.37E-03	nCi	2.18E+06	g
Cf-250	3.08E-24	10	nCi/g	4.06E-19	nCi/g	8.84E-13	nCi	2.18E+06	g
Cf-251	1.39E-16	10	nCi/g	1.83E-11	nCi/g	3.98E-05	nCi	2.18E+06	g
Sum of Fractions	**2.79E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	5.06E-06	3.5	Ci/m^3	2.33E-01	Ci/m^3	3.74E-01	Ci	1.60E+00	m^3
Sr-90	2.39E-04	0.04	Ci/m^3	1.26E-01	Ci/m^3	2.02E-01	Ci	1.60E+00	m^3
Cs-137	1.71E-05	1	Ci/m^3	2.25E-01	Ci/m^3	3.60E-01	Ci	1.60E+00	m^3
Sum of Fractions	**2.61E-04**								

Table 6-21. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 5 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	8.74E-14	0.8	Ci/m^3	9.21E-10	Ci/m^3	3.06E-09	Ci	3.32E+00	m^3
Ni-59	2.58E-07	22	Ci/m^3	7.47E-02	Ci/m^3	2.48E-01	Ci	3.32E+00	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	1.98E-01	Ci	3.32E+00	m^3
Tc-99	2.34E-04	0.30	Ci/m^3	9.27E-01	Ci/m^3	3.08E+00	Ci	3.32E+00	m^3
I-129	1.45E-07	0.008	Ci/m^3	1.53E-05	Ci/m^3	5.07E-05	Ci	3.32E+00	m^3
Np-237	2.07E-04	10	nCi/g	2.73E+01	nCi/g	1.25E+08	nCi	4.58E+06	g
Pu-238	7.45E-03	10	nCi/g	9.82E+02	nCi/g	4.50E+09	nCi	4.58E+06	g
Pu-239	4.83E-03	10	nCi/g	6.37E+02	nCi/g	2.92E+09	nCi	4.58E+06	g

Table 6-21. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-240	3.30E-03	10	nCi/g	4.35E+02	nCi/g	1.99E+09	nCi	4.58E+06	g
Pu-241	2.18E-10	350	nCi/g	1.01E-03	nCi/g	4.61E+03	nCi	4.58E+06	g
Pu-242	8.08E-06	10	nCi/g	1.06E+00	nCi/g	4.88E+06	nCi	4.58E+06	g
Pu-244	5.56E-15	10	nCi/g	7.33E-10	nCi/g	3.36E-03	nCi	4.58E+06	g
Am-241	9.96E-03	10	nCi/g	1.31E+03	nCi/g	6.01E+09	nCi	4.58E+06	g
Am-242m	1.65E-06	10	nCi/g	2.18E-01	nCi/g	9.99E+05	nCi	4.58E+06	g
Am-243	1.80E-06	10	nCi/g	2.38E-01	nCi/g	1.09E+06	nCi	4.58E+06	g
Cm-242	6.83E-09	2,000	nCi/g	1.80E-01	nCi/g	8.25E+05	nCi	4.58E+06	g
Cm-243	7.61E-11	10	nCi/g	1.00E-05	nCi/g	4.59E+01	nCi	4.58E+06	g
Cm-244	3.42E-12	10	nCi/g	4.51E-07	nCi/g	2.06E+00	nCi	4.58E+06	g
Cm-245	7.21E-09	10	nCi/g	9.51E-04	nCi/g	4.36E+03	nCi	4.58E+06	g
Cm-246	7.55E-10	10	nCi/g	9.95E-05	nCi/g	4.56E+02	nCi	4.58E+06	g
Cm-247	1.23E-15	10	nCi/g	1.63E-10	nCi/g	7.45E-04	nCi	4.58E+06	g
Cm-248	1.74E-15	10	nCi/g	2.29E-10	nCi/g	1.05E-03	nCi	4.58E+06	g
Cf-249	4.90E-15	10	nCi/g	6.46E-10	nCi/g	2.96E-03	nCi	4.58E+06	g
Cf-250	3.14E-24	10	nCi/g	4.15E-19	nCi/g	1.90E-12	nCi	4.58E+06	g
Cf-251	1.41E-16	10	nCi/g	1.86E-11	nCi/g	8.52E-05	nCi	4.58E+06	g
Sum of Fractions	**2.60E-02**								
Table 2 of 10 CFR 61.55									
Ni-63	4.30E-06	3.5	Ci/m^3	1.98E-01	Ci/m^3	6.58E-01	Ci	3.32E+00	m^3
Sr-90	2.48E-04	0.04	Ci/m^3	1.31E-01	Ci/m^3	4.33E-01	Ci	3.32E+00	m^3
Cs-137	1.77E-05	1	Ci/m^3	2.33E-01	Ci/m^3	7.75E-01	Ci	3.32E+00	m^3
Sum of Fractions	**2.70E-04**								

Table 6-22. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 6 bins.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	3.32E-14	0.8	Ci/m³	3.50E-10	Ci/m³	1.43E-09	Ci	4.08E+00	m³
Ni-59	9.77E-08	22	Ci/m³	2.83E-02	Ci/m³	1.16E-01	Ci	4.08E+00	m³
Nb-94	0.00E+00	0.02	Ci/m³	0.00E+00	Ci/m³	9.24E-02	Ci	4.08E+00	m³
Tc-99	8.86E-05	0.30	Ci/m³	3.50E-01	Ci/m³	1.43E+00	Ci	4.08E+00	m³
I-129	5.47E-08	0.008	Ci/m³	5.76E-06	Ci/m³	2.35E-05	Ci	4.08E+00	m³
Np-237	4.79E-05	10	nCi/g	6.32E+00	nCi/g	3.17E+07	nCi	5.02E+06	g
Pu-238	1.83E-03	10	nCi/g	2.41E+02	nCi/g	1.21E+09	nCi	5.02E+06	g
Pu-239	2.86E-03	10	nCi/g	3.76E+02	nCi/g	1.89E+09	nCi	5.02E+06	g
Pu-240	1.49E-03	10	nCi/g	1.97E+02	nCi/g	9.87E+08	nCi	5.02E+06	g
Pu-241	9.33E-11	350	nCi/g	4.30E-04	nCi/g	2.16E+03	nCi	5.02E+06	g
Pu-242	3.25E-06	10	nCi/g	4.28E-01	nCi/g	2.15E+06	nCi	5.02E+06	g
Pu-244	2.37E-15	10	nCi/g	3.13E-10	nCi/g	1.57E-03	nCi	5.02E+06	g
Am-241	2.59E-03	10	nCi/g	3.42E+02	nCi/g	1.71E+09	nCi	5.02E+06	g
Am-242m	7.05E-07	10	nCi/g	9.30E-02	nCi/g	4.66E+05	nCi	5.02E+06	g
Am-243	9.34E-07	10	nCi/g	1.23E-01	nCi/g	6.17E+05	nCi	5.02E+06	g
Cm-242	2.91E-09	2,000	nCi/g	7.67E-02	nCi/g	3.85E+05	nCi	5.02E+06	g
Cm-243	3.24E-11	10	nCi/g	4.28E-06	nCi/g	2.14E+01	nCi	5.02E+06	g
Cm-244	1.64E-12	10	nCi/g	2.16E-07	nCi/g	1.08E+00	nCi	5.02E+06	g
Cm-245	3.07E-09	10	nCi/g	4.05E-04	nCi/g	2.03E+03	nCi	5.02E+06	g
Cm-246	3.21E-10	10	nCi/g	4.23E-05	nCi/g	2.12E+02	nCi	5.02E+06	g
Cm-247	5.25E-16	10	nCi/g	6.92E-11	nCi/g	3.47E-04	nCi	5.02E+06	g
Cm-248	7.36E-16	10	nCi/g	9.70E-11	nCi/g	4.87E-04	nCi	5.02E+06	g
Cf-249	2.09E-15	10	nCi/g	2.76E-10	nCi/g	1.38E-03	nCi	5.02E+06	g
Cf-250	1.34E-24	10	nCi/g	1.76E-19	nCi/g	8.84E-13	nCi	5.02E+06	g
Cf-251	6.04E-17	10	nCi/g	7.96E-12	nCi/g	3.99E-05	nCi	5.02E+06	g
Sum of Fractions	**8.91E-03**								

Table 6-22. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	1.11E-06	3.5	Ci/m^3	5.13E-02	Ci/m^3	2.09E-01	Ci	4.08E+00	m^3
Sr-90	8.77E-05	0.04	Ci/m^3	4.62E-02	Ci/m^3	1.89E-01	Ci	4.08E+00	m^3
Cs-137	6.72E-06	1	Ci/m^3	8.86E-02	Ci/m^3	3.61E-01	Ci	4.08E+00	m^3
Sum of Fractions	**9.56E-05**								

Table 6-23. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 2 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	2.12E-12	0.8	Ci/m^3	1.01E-06	Ci/m^3	3.58E-11	Ci	3.56E-05	m^3
Ni-59	1.83E-09	22	Ci/m^3	2.39E-02	Ci/m^3	8.51E-07	Ci	3.56E-05	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	6.78E-07	Ci	3.56E-05	m^3
Tc-99	5.07E-06	0.30	Ci/m^3	9.03E-01	Ci/m^3	3.21E-05	Ci	3.56E-05	m^3
I-129	3.11E-09	0.008	Ci/m^3	1.47E-05	Ci/m^3	5.24E-10	Ci	3.56E-05	m^3
Np-237	3.04E-07	10	nCi/g	1.80E+00	nCi/g	7.88E-08	Ci	4.37E+01	g
Pu-238	5.56E-05	10	nCi/g	3.30E+02	nCi/g	1.44E-05	Ci	4.37E+01	g
Pu-239	2.90E-05	10	nCi/g	1.72E+02	nCi/g	7.53E-06	Ci	4.37E+01	g
Pu-240	2.22E-05	10	nCi/g	1.32E+02	nCi/g	5.77E-06	Ci	4.37E+01	g
Pu-241	1.75E-12	350	nCi/g	3.63E-04	nCi/g	1.59E-11	Ci	4.37E+01	g
Pu-242	5.42E-08	10	nCi/g	3.22E-01	nCi/g	1.41E-08	Ci	4.37E+01	g
Pu-244	8.51E-17	10	nCi/g	5.05E-10	nCi/g	2.21E-17	Ci	4.37E+01	g
Am-241	1.08E-04	10	nCi/g	6.42E+02	nCi/g	2.81E-05	Ci	4.37E+01	g
Am-242m	1.38E-08	10	nCi/g	8.18E-02	nCi/g	3.58E-09	Ci	4.37E+01	g
Am-243	1.30E-08	10	nCi/g	7.73E-02	nCi/g	3.38E-09	Ci	4.37E+01	g
Cm-242	5.69E-11	2,000	nCi/g	6.75E-02	nCi/g	2.95E-09	Ci	4.37E+01	g
Cm-243	6.16E-13	10	nCi/g	3.65E-06	nCi/g	1.60E-13	Ci	4.37E+01	g

Table 6-23. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-244	3.22E-14	10	nCi/g	1.91E-07	nCi/g	8.35E-15	Ci	4.37E+01	g
Cm-245	5.82E-11	10	nCi/g	3.45E-04	nCi/g	1.51E-11	Ci	4.37E+01	g
Cm-246	6.02E-12	10	nCi/g	3.57E-05	nCi/g	1.56E-12	Ci	4.37E+01	g
Cm-247	9.83E-18	10	nCi/g	5.83E-11	nCi/g	2.55E-18	Ci	4.37E+01	g
Cm-248	1.38E-17	10	nCi/g	8.17E-11	nCi/g	3.57E-18	Ci	4.37E+01	g
Cf-249	3.91E-17	10	nCi/g	2.32E-10	nCi/g	1.01E-17	Ci	4.37E+01	g
Cf-250	2.51E-26	10	nCi/g	1.49E-19	nCi/g	6.51E-27	Ci	4.37E+01	g
Cf-251	1.13E-18	10	nCi/g	6.70E-12	nCi/g	2.93E-19	Ci	4.37E+01	g
Sum of Fractions	**2.21E-04**								
Table 2 of 10 CFR 61.55									
Ni-63	3.88E-08	3.5	Ci/m³	8.05E-02	Ci/m³	2.86E-06	Ci	3.56E-05	m³
Sr-90	4.54E-06	0.04	Ci/m³	1.08E-01	Ci/m³	3.83E-06	Ci	3.56E-05	m³
Cs-137	3.41E-07	1	Ci/m³	2.02E-01	Ci/m³	7.19E-06	Ci	3.56E-05	m³
Sum of Fractions	**4.92E-06**								

Table 6-24. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 3 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	1.40E-15	0.8	Ci/m³	6.62E-10	Ci/m³	2.35E-14	Ci	3.56E-05	m³
Ni-59	4.12E-09	22	Ci/m³	5.38E-02	Ci/m³	1.91E-06	Ci	3.56E-05	m³
Nb-94	0.00E+00	0.02	Ci/m³	0.00E+00	Ci/m³	1.52E-06	Ci	3.56E-05	m³
Tc-99	3.71E-06	0.30	Ci/m³	6.59E-01	Ci/m³	2.34E-05	Ci	3.56E-05	m³
I-129	2.30E-09	0.008	Ci/m³	1.09E-05	Ci/m³	3.88E-10	Ci	3.56E-05	m³
Np-237	9.63E-07	10	nCi/g	5.71E+00	nCi/g	2.62E-07	Ci	4.59E+01	g
Pu-238	8.25E-05	10	nCi/g	4.89E+02	nCi/g	2.24E-05	Ci	4.59E+01	g
Pu-239	5.15E-05	10	nCi/g	3.05E+02	nCi/g	1.40E-05	Ci	4.59E+01	g

Table 6-24 (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-240	3.58E-05	10	nCi/g	2.13E+02	nCi/g	9.75E-06	Ci	4.59E+01	g
Pu-241	3.65E-12	350	nCi/g	7.57E-04	nCi/g	3.47E-11	Ci	4.59E+01	g
Pu-242	9.53E-08	10	nCi/g	5.65E-01	nCi/g	2.59E-08	Ci	4.59E+01	g
Pu-244	9.50E-17	10	nCi/g	5.64E-10	nCi/g	2.58E-17	Ci	4.59E+01	g
Am-241	1.66E-04	10	nCi/g	9.88E+02	nCi/g	4.53E-05	Ci	4.59E+01	g
Am-242m	2.82E-08	10	nCi/g	1.68E-01	nCi/g	7.68E-09	Ci	4.59E+01	g
Am-243	3.47E-08	10	nCi/g	2.06E-01	nCi/g	9.43E-09	Ci	4.59E+01	g
Cm-242	1.17E-10	2,000	nCi/g	1.38E-01	nCi/g	6.34E-09	Ci	4.59E+01	g
Cm-243	1.30E-12	10	nCi/g	7.69E-06	nCi/g	3.53E-13	Ci	4.59E+01	g
Cm-244	6.30E-14	10	nCi/g	3.74E-07	nCi/g	1.71E-14	Ci	4.59E+01	g
Cm-245	1.23E-10	10	nCi/g	7.30E-04	nCi/g	3.35E-11	Ci	4.59E+01	g
Cm-246	1.29E-11	10	nCi/g	7.64E-05	nCi/g	3.50E-12	Ci	4.59E+01	g
Cm-247	2.10E-17	10	nCi/g	1.25E-10	nCi/g	5.72E-18	Ci	4.59E+01	g
Cm-248	2.95E-17	10	nCi/g	1.75E-10	nCi/g	8.02E-18	Ci	4.59E+01	g
Cf-249	8.36E-17	10	nCi/g	4.96E-10	nCi/g	2.28E-17	Ci	4.59E+01	g
Cf-250	5.37E-26	10	nCi/g	3.19E-19	nCi/g	1.46E-26	Ci	4.59E+01	g
Cf-251	2.42E-18	10	nCi/g	1.43E-11	nCi/g	6.57E-19	Ci	4.59E+01	g
Sum of Fractions	**3.41E-04**								
Table 2 of 10 CFR 61.55									
Ni-63	7.50E-08	3.5	Ci/m^3	1.56E-01	Ci/m^3	5.53E-06	Ci	3.56E-05	m^3
Sr-90	3.65E-06	0.04	Ci/m^3	8.66E-02	Ci/m^3	3.08E-06	Ci	3.56E-05	m^3
Cs-137	2.82E-07	1	Ci/m^3	1.68E-01	Ci/m^3	5.96E-06	Ci	3.56E-05	m^3
Sum of Fractions	**4.01E-06**								

Table 6-25. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 4 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	1.87E-15	0.8	Ci/m^3	8.87E-10	Ci/m^3	3.15E-14	Ci	3.56E-05	m^3
Ni-59	5.53E-09	22	Ci/m^3	7.22E-02	Ci/m^3	2.57E-06	Ci	3.56E-05	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	2.04E-06	Ci	3.56E-05	m^3
Tc-99	4.94E-06	0.30	Ci/m^3	8.80E-01	Ci/m^3	3.13E-05	Ci	3.56E-05	m^3
I-129	3.07E-09	0.008	Ci/m^3	1.46E-05	Ci/m^3	5.18E-10	Ci	3.56E-05	m^3
Np-237	5.02E-06	10	nCi/g	2.98E+01	nCi/g	1.44E-06	Ci	4.84E+01	g
Pu-238	1.79E-04	10	nCi/g	1.06E+03	nCi/g	5.12E-05	Ci	4.84E+01	g
Pu-239	1.16E-04	10	nCi/g	6.90E+02	nCi/g	3.34E-05	Ci	4.84E+01	g
Pu-240	7.25E-05	10	nCi/g	4.30E+02	nCi/g	2.08E-05	Ci	4.84E+01	g
Pu-241	4.74E-12	350	nCi/g	9.83E-04	nCi/g	4.75E-11	Ci	4.84E+01	g
Pu-242	2.11E-07	10	nCi/g	1.25E+00	nCi/g	6.06E-08	Ci	4.84E+01	g
Pu-244	1.21E-16	10	nCi/g	7.17E-10	nCi/g	3.47E-17	Ci	4.84E+01	g
Am-241	2.43E-04	10	nCi/g	1.44E+03	nCi/g	6.97E-05	Ci	4.84E+01	g
Am-242m	3.60E-08	10	nCi/g	2.14E-01	nCi/g	1.03E-08	Ci	4.84E+01	g
Am-243	4.80E-08	10	nCi/g	2.85E-01	nCi/g	1.38E-08	Ci	4.84E+01	g
Cm-242	1.49E-10	2,000	nCi/g	1.76E-01	nCi/g	8.52E-09	Ci	4.84E+01	g
Cm-243	1.65E-12	10	nCi/g	9.80E-06	nCi/g	4.74E-13	Ci	4.84E+01	g
Cm-244	8.53E-14	10	nCi/g	5.06E-07	nCi/g	2.45E-14	Ci	4.84E+01	g
Cm-245	1.57E-10	10	nCi/g	9.29E-04	nCi/g	4.49E-11	Ci	4.84E+01	g
Cm-246	1.64E-11	10	nCi/g	9.75E-05	nCi/g	4.71E-12	Ci	4.84E+01	g
Cm-247	2.68E-17	10	nCi/g	1.59E-10	nCi/g	7.69E-18	Ci	4.84E+01	g
Cm-248	3.76E-17	10	nCi/g	2.23E-10	nCi/g	1.08E-17	Ci	4.84E+01	g
Cf-249	1.06E-16	10	nCi/g	6.31E-10	nCi/g	3.05E-17	Ci	4.84E+01	g
Cf-250	6.85E-26	10	nCi/g	4.06E-19	nCi/g	1.96E-26	Ci	4.84E+01	g
Cf-251	3.08E-18	10	nCi/g	1.83E-11	nCi/g	8.83E-19	Ci	4.84E+01	g
Sum of Fractions	**6.21E-04**								

Table 6-25. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 2 of 10 CFR 61.55									
Ni-63	1.12E-07	3.5	Ci/m^3	2.33E-01	Ci/m^3	8.30E-06	Ci	3.56E-05	m^3
Sr-90	5.31E-06	0.04	Ci/m^3	1.26E-01	Ci/m^3	4.48E-06	Ci	3.56E-05	m^3
Cs-137	3.80E-07	1	Ci/m^3	2.25E-01	Ci/m^3	8.00E-06	Ci	3.56E-05	m^3
Sum of Fractions	**5.80E-06**								

Table 6-26. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 5 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	1.94E-15	0.8	Ci/m^3	9.21E-10	Ci/m^3	3.28E-14	Ci	3.56E-05	m^3
Ni-59	5.73E-09	22	Ci/m^3	7.47E-02	Ci/m^3	2.66E-06	Ci	3.56E-05	m^3
Nb-94	0.00E+00	0.02	Ci/m^3	0.00E+00	Ci/m^3	2.12E-06	Ci	3.56E-05	m^3
Tc-99	5.21E-06	0.30	Ci/m^3	9.27E-01	Ci/m^3	3.29E-05	Ci	3.56E-05	m^3
I-129	3.22E-09	0.008	Ci/m^3	1.53E-05	Ci/m^3	5.43E-10	Ci	3.56E-05	m^3
Np-237	4.60E-06	10	nCi/g	2.73E+01	nCi/g	1.34E-06	Ci	4.91E+01	g
Pu-238	1.66E-04	10	nCi/g	9.82E+02	nCi/g	4.82E-05	Ci	4.91E+01	g
Pu-239	1.07E-04	10	nCi/g	6.37E+02	nCi/g	3.12E-05	Ci	4.91E+01	g
Pu-240	7.34E-05	10	nCi/g	4.35E+02	nCi/g	2.14E-05	Ci	4.91E+01	g
Pu-241	4.85E-12	350	nCi/g	1.01E-03	nCi/g	4.94E-11	Ci	4.91E+01	g
Pu-242	1.79E-07	10	nCi/g	1.06E+00	nCi/g	5.22E-08	Ci	4.91E+01	g
Pu-244	1.24E-16	10	nCi/g	7.33E-10	nCi/g	3.60E-17	Ci	4.91E+01	g
Am-241	2.21E-04	10	nCi/g	1.31E+03	nCi/g	6.44E-05	Ci	4.91E+01	g
Am-242m	3.68E-08	10	nCi/g	2.18E-01	nCi/g	1.07E-08	Ci	4.91E+01	g
Am-243	4.01E-08	10	nCi/g	2.38E-01	nCi/g	1.17E-08	Ci	4.91E+01	g
Cm-242	1.52E-10	2,000	nCi/g	1.80E-01	nCi/g	8.83E-09	Ci	4.91E+01	g
Cm-243	1.69E-12	10	nCi/g	1.00E-05	nCi/g	4.92E-13	Ci	4.91E+01	g

Table 6-26. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Cm-244	7.60E-14	10	nCi/g	4.51E-07	nCi/g	2.21E-14	Ci	4.91E+01	g
Cm-245	1.60E-10	10	nCi/g	9.51E-04	nCi/g	4.66E-11	Ci	4.91E+01	g
Cm-246	1.68E-11	10	nCi/g	9.95E-05	nCi/g	4.88E-12	Ci	4.91E+01	g
Cm-247	2.74E-17	10	nCi/g	1.63E-10	nCi/g	7.98E-18	Ci	4.91E+01	g
Cm-248	3.86E-17	10	nCi/g	2.29E-10	nCi/g	1.12E-17	Ci	4.91E+01	g
Cf-249	1.09E-16	10	nCi/g	6.46E-10	nCi/g	3.17E-17	Ci	4.91E+01	g
Cf-250	6.99E-26	10	nCi/g	4.15E-19	nCi/g	2.03E-26	Ci	4.91E+01	g
Cf-251	3.14E-18	10	nCi/g	1.86E-11	nCi/g	9.13E-19	Ci	4.91E+01	g
Sum of Fractions	**5.78E-04**								
Table 2 of 10 CFR 61.55									
Ni-63	9.55E-08	3.5	Ci/m³	1.98E-01	Ci/m³	7.05E-06	Ci	3.56E-05	m³
Sr-90	5.50E-06	0.04	Ci/m³	1.31E-01	Ci/m³	4.64E-06	Ci	3.56E-05	m³
Cs-137	3.93E-07	1	Ci/m³	2.33E-01	Ci/m³	8.30E-06	Ci	3.56E-05	m³
Sum of Fractions	**5.99E-06**								

Table 6-27. Class C limit comparison to Tables 1 and 2 of 10 CFR 61.55 based on the chronic intruder drilling scenario for the Calcined Solids Storage Facility 6 transport lines.

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Table 1 of 10 CFR 61.55									
C-14	7.39E-16	0.8	Ci/m³	3.50E-10	Ci/m³	1.25E-14	Ci	3.56E-05	m³
Ni-59	2.17E-09	22	Ci/m³	2.83E-02	Ci/m³	1.01E-06	Ci	3.56E-05	m³
Nb-94	0.00E+00	0.02	Ci/m³	0.00E+00	Ci/m³	8.06E-07	Ci	3.56E-05	m³
Tc-99	1.97E-06	0.30	Ci/m³	3.50E-01	Ci/m³	1.25E-05	Ci	3.56E-05	m³
I-129	1.21E-09	0.008	Ci/m³	5.76E-06	Ci/m³	2.05E-10	Ci	3.56E-05	m³
Np-237	1.07E-06	10	nCi/g	6.32E+00	nCi/g	2.76E-07	Ci	4.37E+01	g
Pu-238	4.07E-05	10	nCi/g	2.41E+02	nCi/g	1.06E-05	Ci	4.37E+01	g
Pu-239	6.35E-05	10	nCi/g	3.76E+02	nCi/g	1.65E-05	Ci	4.37E+01	g

Table 6-27. (continued).

Radionuclide	Fraction of Class C Limit	Class A Table Value	Unit	Decayed Inventory Concentration (C_{Ri})	Unit	Decayed Inventory (I_{Ri})	Unit	Waste Volume or Mass (V_w or M_w)	Unit
Pu-240	3.32E-05	10	nCi/g	1.97E+02	nCi/g	8.61E-06	Ci	4.37E+01	g
Pu-241	2.07E-12	350	nCi/g	4.30E-04	nCi/g	1.88E-11	Ci	4.37E+01	g
Pu-242	7.22E-08	10	nCi/g	4.28E-01	nCi/g	1.87E-08	Ci	4.37E+01	g
Pu-244	5.28E-17	10	nCi/g	3.13E-10	nCi/g	1.37E-17	Ci	4.37E+01	g
Am-241	5.76E-05	10	nCi/g	3.42E+02	nCi/g	1.49E-05	Ci	4.37E+01	g
Am-242m	1.57E-08	10	nCi/g	9.30E-02	nCi/g	4.07E-09	Ci	4.37E+01	g
Am-243	2.07E-08	10	nCi/g	1.23E-01	nCi/g	5.38E-09	Ci	4.37E+01	g
Cm-242	6.47E-11	2,000	nCi/g	7.67E-02	nCi/g	3.36E-09	Ci	4.37E+01	g
Cm-243	7.21E-13	10	nCi/g	4.28E-06	nCi/g	1.87E-13	Ci	4.37E+01	g
Cm-244	3.64E-14	10	nCi/g	2.16E-07	nCi/g	9.45E-15	Ci	4.37E+01	g
Cm-245	6.83E-11	10	nCi/g	4.05E-04	nCi/g	1.77E-11	Ci	4.37E+01	g
Cm-246	7.13E-12	10	nCi/g	4.23E-05	nCi/g	1.85E-12	Ci	4.37E+01	g
Cm-247	1.17E-17	10	nCi/g	6.92E-11	nCi/g	3.03E-18	Ci	4.37E+01	g
Cm-248	1.64E-17	10	nCi/g	9.70E-11	nCi/g	4.24E-18	Ci	4.37E+01	g
Cf-249	4.65E-17	10	nCi/g	2.76E-10	nCi/g	1.21E-17	Ci	4.37E+01	g
Cf-250	2.97E-26	10	nCi/g	1.76E-19	nCi/g	7.71E-27	Ci	4.37E+01	g
Cf-251	1.34E-18	10	nCi/g	7.96E-12	nCi/g	3.48E-19	Ci	4.37E+01	g
Sum of Fractions	**1.98E-04**								
Table 2 of 10 CFR 61.55									
Ni-63	2.47E-08	3.5	Ci/m^3	5.13E-02	Ci/m^3	1.82E-06	Ci	3.56E-05	m^3
Sr-90	1.95E-06	0.04	Ci/m^3	4.62E-02	Ci/m^3	1.64E-06	Ci	3.56E-05	m^3
Cs-137	1.49E-07	1	Ci/m^3	8.86E-02	Ci/m^3	3.15E-06	Ci	3.56E-05	m^3
Sum of Fractions	**2.12E-06**								

6.4 Conclusion

Stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are expected to meet concentration limits for Class C LLW as set out in 10 CFR 61.55. While DOE believes there is a reasonable basis to conclude that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein will not exceed the Class C concentration limits in 10 CFR 61.55, DOE is consulting with the NRC on DOE's disposal plans, as described in this Draft CSSF Basis Document, to take full advantage of the consultation process in NDAA Section 3116(a)(3)(B)(iii).

7. WASTE WILL BE DISPOSED OF IN ACCORDANCE WITH 10 CFR 61, SUBPART C, PERFORMANCE OBJECTIVES

Section 7 Purpose

The purpose of this section is to demonstrate that the stabilized residuals in the CSSF bins (including integral equipment), and transport lines will be disposed of in compliance with the performance objectives for land disposal of LLW found in 10 CFR 61, Subpart C, and 10 CFR 61.41 through 61.44.

Section 7 Contents

This section describes key parameters and results from the CSSF PA/CA that demonstrate compliance with the performance objectives in 10 CFR 61.41 and 61.42; DOE regulatory and contractual requirements that ensure compliance with 10 CFR 61.43; and relevant factors of CSSF siting, design, use, operation, and closure that ensure compliance with 10 CFR 61.44.

Section 7 Key Points

- Stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein will be disposed of (in situ) in compliance with the performance objectives for land disposal of LLW found in 10 CFR 61.41 through 61.44 and, therefore, meet Criterion (3)(A)(i) of NDAA Section 3116(a).
- The CSSF PA/CA provides the technical basis and results to demonstrate reasonable assurance that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet the 10 CFR 61, Subpart C, performance objectives.
- Results of the peak groundwater annual all-pathways dose to a hypothetical member of the public, shown in the CSSF PA/CA base case, are orders of magnitude below the 25-mrem annual all-pathways dose performance objective set forth in 10 CFR 61.41, even without credit for some existing barriers.
- The CSSF PA/CA hypothetical inadvertent intruder analysis demonstrates reasonable assurance of compliance with the performance objective in 10 CFR 61.42 based on exposure scenarios assumed to occur at the CSSF bins and transport lines at 500 years post-closure of the CSSF.
- The DOE regulatory and contractual requirements establish dose limits based on 10 CFR 835 and relevant DOE orders. These dose limits correspond to the radiation protection standards set out in 10 CFR 20, as cross-referenced in 10 CFR 61.43.
- Radiation protection during operations and closure of the CSSF will be maintained using ALARA principles as implemented by the ICP Radiation Protection Program. Current INTEC CSSF operations are conducted in compliance with the standards for radiation protection as set out in 10 CFR 835 and relevant DOE orders, and CSSF closure operations will comply with applicable dose limits and with ALARA provisions in the ICP Radiation Protection Program. CSSF closure will support long-term stability consistent with objectives set forth in 10 CFR 61.44. Calcine will be removed from the CSSF to the maximum extent practical, and any remaining residual waste, equipment, and structures will be stabilized with grout. Engineered barriers (grouted residual waste and void spaces, stainless-steel bins and pipes, and reinforced-concrete vaults) in combination with natural site features (arid environment, low rainfall, remote location, groundwater depth, and geologically stable region) provide long-term safety and structural stability for the closed facility.

This section demonstrates that residual waste remaining in the CSSF will be disposed of in compliance with the performance objectives for land disposal of LLW found in 10 CFR 61.41 through 61.44, in accordance with NDAA Section 3116(a)(3)(A)(i) (Public Law 108-375).

The NDAA Section 3116(a) provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy..., in consultation with the Nuclear Regulatory Commission…, determines –
>
> (3)(A)(i) [Will be disposed of] in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations.

10 CFR 61, Subpart C, § 61.40, "General Requirement" (10 CFR 61.40) states:

> Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in §§61.41 through 61.44.

10 CFR 61.40 requires "reasonable assurance" that exposures are within the limits of the subsequent performance objectives at 10 CFR 61.41 through 61.44 for licensed disposal facilities. Similarly, DOE M 435.1-1 Chg 3 requires the "reasonable expectation" (analogous to "reasonable assurance") that the performance objectives and performance measures set forth in DOE M 435.1-1 Chg 3 will be met. For convenience, this Draft CSSF 3116 Basis Document uses the phrase "reasonable assurance." As explained later in this document, DOE performance objectives and performance measures in Chapter IV.P.(1) of DOE M 435.1-1 Chg 3 established safety requirements comparable to the NRC performance objectives in 10 CFR 61, Subpart C. A detailed comparison of DOE and NRC safety requirements for disposal of radioactive waste is provided in Appendix A.

DOE has developed a CSSF PA/CA (DOE-ID 2022a) that provides the technical basis and results demonstrating there is reasonable assurance that 10 CFR 61.41 and 61.42 performance objectives will be met at the time of CSSF closure. These analyses were performed using a variety of modeling codes, including the GWSCREEN, CAP88-PC, and RESRAD-ONSITE computer models.[62] As required by DOE M 435.1-1 Chg 3, maintenance of the CSSF PA/CA will include future revisions as needed (e.g., to incorporate new information and update model codes).

The CSSF PA/CA groundwater modeling consisted of a hybrid approach using both deterministic modeling as well as probabilistic modeling for certain sensitivity and uncertainty analyses. The CSSF PA/CA includes deterministic and probabilistic groundwater analyses for 1 million years after CSSF closure. This approach envelopes both the 1,000-year post-closure period, as described in DOE M 435.1-1 Chg 3 for DOE disposal facility PAs, and the 10,000-year period suggested in NUREG-1854 (NRC 2007).

The CSSF PA/CA details analyses performed to provide reasonable assurance that stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will be disposed of in compliance with 10 CFR 61.41 and 61.42 performance objectives. The CSSF PA/CA provides details on the development and calculation of the following doses:

- Potential radiological doses to a hypothetical member of the public
- Potential radiological doses to a hypothetical inadvertent intruder.

62. References to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise do not necessarily constitute or imply its endorsement, recommendation, or favoring by the U.S. Government, any agency thereof, or any company affiliated with the Idaho Cleanup Project.

These calculations were performed to provide information regarding potential peak doses from the closed CSSF. In addition, uncertainty and sensitivity analyses were used to ensure that adequate information is available to support decisions related to the closure of the CSSF.

7.1 Dose to Representative Member of the Public

10 CFR 61.41 provides the following:

> Concentrations of radioactive material which may be released to the general environment in groundwater, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

7.1.1 General Approach

To demonstrate compliance with the NRC performance objective in 10 CFR 61.41, a 25-mrem peak annual all-pathways ED is used rather than individual organ doses. NRC guidance in NUREG-1854 (NRC 2007) states that the 25-mrem annual all-pathways ED is used by the NRC in making the assessment for compliance with the whole body, thyroid, and any other organ limits in 10 CFR 61.41 and is protective of human health and the environment.

In addition, NUREG-1854 (NRC 2007) states:

> … incidental waste determinations may use total effective dose equivalent (TEDE) without specific consideration of individual organ doses. Intruder calculations should be based on 5 mSv [500 mrem] TEDE limit, without specific consideration of individual organ doses, to ensure consistency between 10 CFR 61.41 and 10 CFR 61.43. Because of the tissue weighting factors and the magnitude of the TEDE limit, specific organ dose limits are not necessary for protection from deterministic effects.

The hypothetical future member of the public is assumed to be located at the boundary of the DOE-controlled area until the assumed active institutional control period ends (i.e., 100 years after closure of CSSF), at which time the receptor is assumed to move to the point of maximum exposure 100 m (328 ft) downgradient from the CSSF. For the purposes of demonstrating reasonable assurance that the 10 CFR 61.41 performance objective will be met, the peak annual all-pathways ED 100 m (328 ft) downgradient from the CSSF is used. This is another example of added pessimism being applied, because doses beyond 100 m from the CSSF as a whole would be lower.

Pathways for release to a member of the public considered in the CSSF PA/CA analyses are discussed in the following subsections. The scenarios are not assumed to occur until after the assumed 100-year institutional control period ends.[63]

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63. As part of the efforts related to the end state vision for the INL Site, planning assumptions for land use within, and adjacent to, the INL Site have moved toward the assumption that key areas of the INL Site, including INTEC, will remain under government control until at least 2095 and portions of the INL Site will remain under government control in perpetuity (INL 2016; DOE-ID 2022d).

7.1.2 Public Release All Pathways Dose Analysis

The total annual all-pathways ED for the public in the CSSF PA/CA (DOE-ID 2022a) is a combination of the dose from the groundwater pathway and air pathway.[64] The receptor is considered to be an MEI who is assumed to be located along the centerline of the air pathway plume and using water from a well located at the highest concentration point in the SRPA 100 m (328 ft) downgradient from the CSSF. Groundwater concentrations are used as the concentrations at the wellhead. This approach has been applied to maintain consistency between groundwater protection performance objectives and the annual all-pathways dose performance objective in the CSSF PA/CA, but this approach does not take into account any dilution that may occur in the well as it is pumped.

For the all-pathways scenario, the individual who receives the dose is a representative person (from "ICRP Publication 101a: Assessing Dose of the Representative Person for the Purpose of the Radiation Protection of the Public" [ICRP 2006]) who resides near the CSSF and draws contaminated water from a downgradient well. The all-pathways representative person is assumed to use the water to drink, shower, irrigate crops, and water livestock. The exposed representative person is assumed to receive dose by the exposure pathways shown in Figure 7-1.

Current DOE and International Commission on Radiological Protection guidance recommends the use of a representative person for describing the hypothetical member of the public for use in projecting future doses (DOE O 458.1 Chg 4; ICRP 2006, 2007). The representative person is described as a person who is representative of the more highly exposed individuals in the population. The concept of the representative person replaces the concept of an average member of the critical group used in older radiation protection guidance.

Internal doses to the representative person are calculated using dose factors provided in DOE-STD-1196-2011, "Derived Concentration Technical Standard," and external doses are calculated using dose factors in EPA's *Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil, Office of Radiation and Indoor Air* (EPA 1993). These dose factors represent ED coefficients calculated to a reference person in the manner of "ICRP Publication 72: Age-Dependent Doses to the Members of the Public from Intake of Radionuclides – Part 5 Compilation of Ingestion and Inhalation Coefficients" (ICRP 1996). The reference person is a hypothetical aggregation of human (male and female) physical and physiological characteristics established by international consensus for the purpose of standardizing radiation dose calculations (DOE-STD-1196-2011; Jannik 2014).

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64. Under DOE M 435.1-1 Chg 3 requirements, the air pathway excludes the dose from radon and its progeny in air. The air pathway has an annual ED limit of 10 mrem, excluding radon and its progeny. Doses from radon are discussed later in this Draft CSSF 3116 Basis Document.

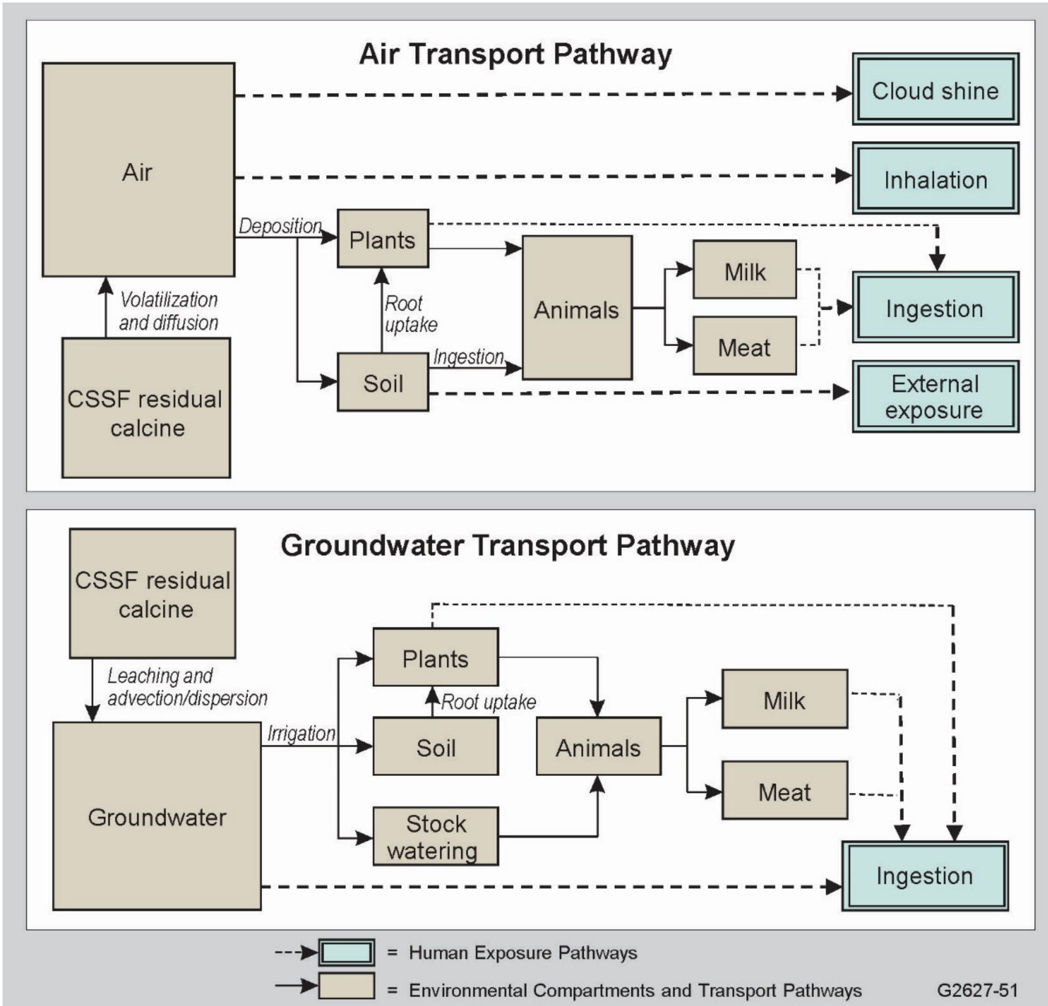

Figure 7-1. Overview of the transport pathways for the analysis of performance of the Calcined Solids Storage Facility.

The source of contamination for the all-pathways scenario is the portion of the inventory transported by groundwater to the well location and drawn through the well. The exposed individual is assumed to use the water to drink, shower, irrigate crops, and water livestock. Exposure occurs through the following pathways:

- Ingestion of water
- Ingestion of fruits and vegetables grown on the farm
- Ingestion of beef raised on the farm
- Ingestion of milk from cows raised on fodder grown on the farm
- Ingestion of eggs from poultry fed with fodder grown on the farm
- Ingestion of poultry fed with fodder grown on the farm
- Ingestion of contaminated soil
- Inhalation of contaminated soil in the air
- External exposure to radiation.

An atmospheric pathway scenario is considered in which an individual is exposed to radionuclides that are diffused to the surface from the wastes disposed of at the CSSF and transported 100 m (328 ft) downwind. Three exposure mechanisms are considered for the atmospheric pathway:

- Air immersion
- Inhalation
- External exposure to the contaminated ground surface.

External exposure results from a fraction of the waste in the air that settles on the ground via dry and wet depositions as they are transported by wind.

In addition to the deterministic peak annual all-pathways ED for the CSSF PA/CA base-case analysis, analyses are provided in the CSSF PA/CA to characterize the context of uncertainty and sensitivity surrounding the peak annual all-pathways ED results. These evaluations focused on key uncertainties and sensitivities identified during calculation of the dose to a member of the public. The uncertainty analyses provide information regarding how collective uncertainty in model input parameters is propagated through the model to the various model results. The sensitivity analyses provide information regarding how various individual input parameters affect dose results. Together, the uncertainty and sensitivity analyses provide assurance that the impacts of variability and uncertainty in analyses of doses to the member of the public are understood and addressed.

Results of the annual all-pathways analysis for the deterministic and uncertainty analyses are provided in the following subsections.

7.1.3 Groundwater All-Pathways Analysis Results

This subsection summarizes results of the groundwater-based component of the all-pathways dose analysis (i.e., not including air-based pathways) in the CSSF PA/CA (DOE-ID 2022a). The air-pathway-based component of the all-pathways dose analysis is addressed separately in Subsection 7.1.4. CSSF PA/CA modeling was used to determine an annual all-pathways ED to a member of the public for comparison with the performance objectives in 10 CFR 61.41. The deterministic base-case analysis in the CSSF PA/CA projected the peak groundwater annual all-pathways dose to the public receptor 100 m (328 ft) downgradient from CSSF to be less than the annual ED limit of 25 mrem.

All-pathways doses for CSSF receptors were all considerably less than the 25-mrem annual dose performance objective for the entire evaluation period (1,000,000 years) and essentially zero for the 1,000-year period after CSSF closure. All-pathways doses for the time thereafter were highest at the CSSF 1 receptor followed by the CSSF 2 receptor (see Table 5-1). The peak annual dose of 1.9E-01 mrem (rounded to two significant digits) occurred at 19,500 years and in the 10,000- to 50,000-year time frame (see Table 5-1). This dose is more than 12 orders of magnitude greater than the peak dose during the 1,000-year post-closure period (6.79E-14 mrem, shown as <1E-10 in Table 5-1). An isopleth map of the doses from the CSSF at the time of peak dose (see Figure 7-2) shows that essentially only CSSFs 1 and 2 contribute to the dose at the time of peak dose at approximately 19,500 years. All-pathways doses at each of the CSSF 100-m (328-ft) receptors are shown in Figure 7-3. Again, the dose at each receptor includes contributions from each CSSF, and the highest dose is at the CSSF 1 receptor. The early maximum doses (around simulation year 19,500) for the receptors 100 m (328 ft) downgradient of CSSFs 1, 2, and 3 are all the result of radionuclide transport from CSSF 1, which fails from corrosion long before the other CSSFs due to the CSSF 1 bins having the thinnest stainless steel of all the bins. The later maximum doses (around simulation year 78,750) result primarily from radionuclide transport from CSSF 2.

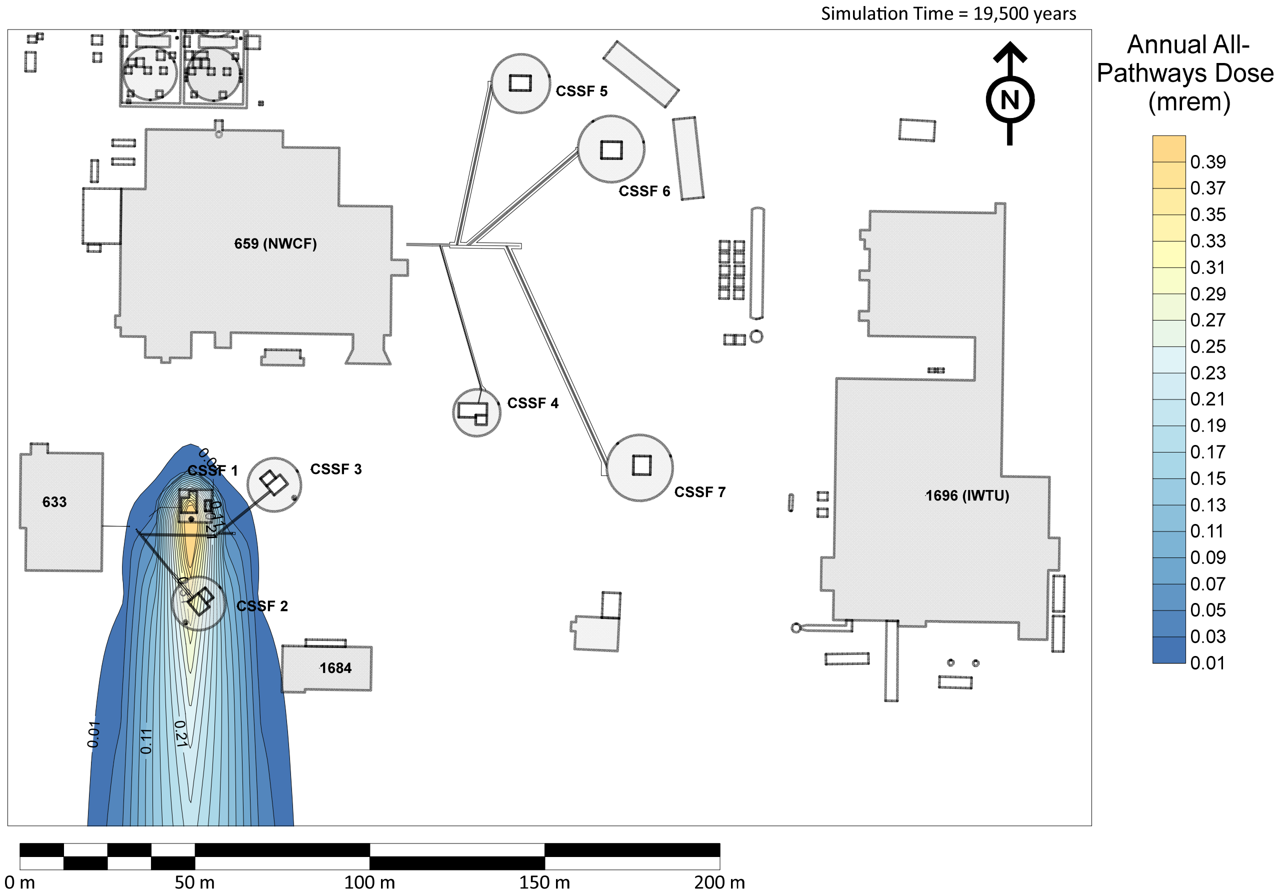

Figure 7-2. Isopleth map of the groundwater annual all-pathways effective dose from all Calcined Solids Storage Facility sources at 19,500 years from the start of the simulation.

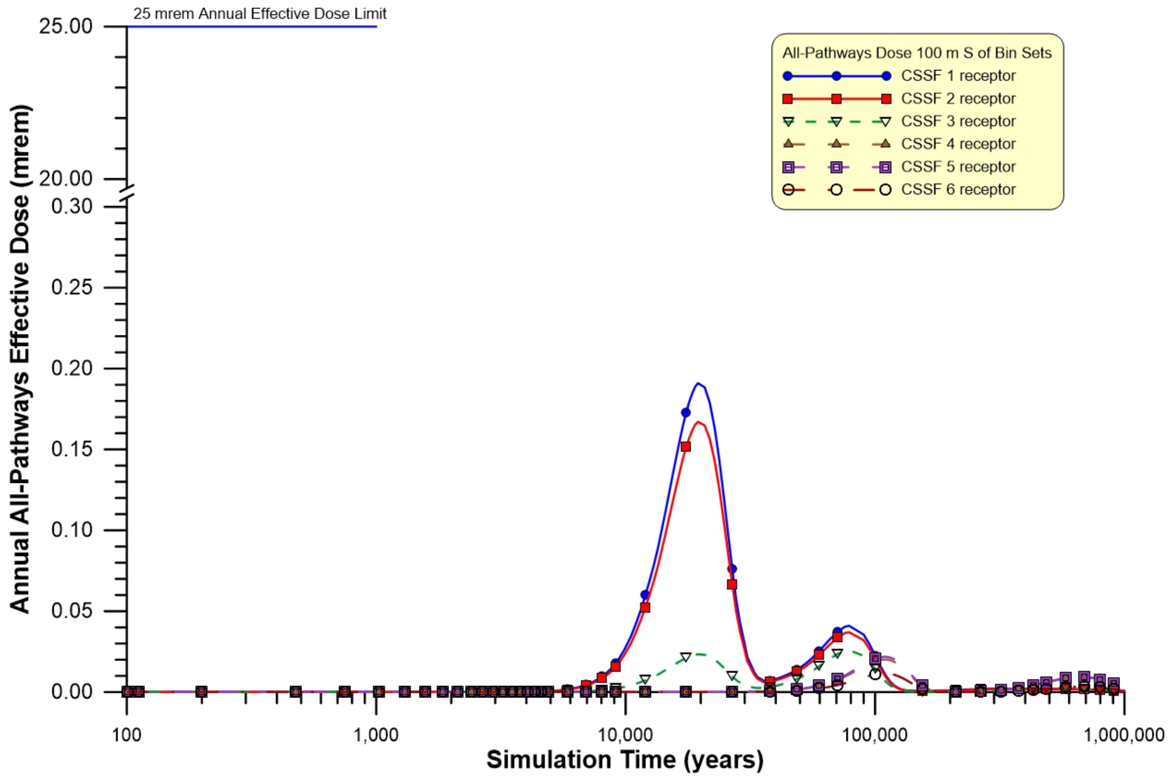

Figure 7-3. Groundwater annual all-pathways effective dose at the 100-m (328-ft) receptors for each Calcined Solids Storage Facility. (Note that (1) in the model, each receptor is placed 100 m (328 ft) downgradient from the Calcined Solids Storage Facility source and along the plume centerline and (2) the y-axis in the figure is broken between 0.3 and 20 mrem.)

The contribution of each CSSF source to the total annual all-pathways dose at the receptor 100 m (328 ft) downgradient from the CSSF 1 is shown in Table 7-1. At the time of peak dose, 19,500 years following closure, CSSF 1 contributes 100% of the total dose. The remaining sources contribute little to the total dose for the CSSF 1 receptor at the time of peak because they have not fully failed at that time. At the time of the second peak at 78,750 years, CSSF 2 contributes 91% of the total dose, with CSSFs 1 and 3 contributing the remaining 9%. The peak times are different for each CSSF; consequently, the doses listed in Table 7-1 do not add to the total (peak dose) because the total is a function of time at the specified location.

Groundwater annual all-pathways EDs are dominated by Tc-99 (1.91E-01 mrem annual dose) (Table 5-2 and Figure 5-1) followed by Se-79 (2.19E-03 mrem annual dose) between 10,000 and 50,000 years and by Np-237 and progeny (2.03E-03 mrem annual dose) after 100,000 years. The actinides (Pu-239, Pu-240, and Pu-242) and Cs-135 contribute little to the annual all-pathways dose until >500,000 years after the start of the simulation because these radionuclides sorb strongly in the vadose zone and, consequently, have long unsaturated transit times.

Table 7-1. Maximum groundwater annual all-pathways effective dose contribution by source at the receptor 100 m (328 ft) south of Calcined Solids Storage Facility 1 for various time periods.

Time Period (yr)	Maximum Groundwater Annual All-Pathways Effective Dose at Receptor 100-m (328 ft) South of CSSF 1 (mrem)[a]						
	CSSF 1	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6	Total[b]
0–100	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
100–500	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
500–1,000	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10	<1E-10
1,000–5,000	6.30E-04	1.26E-05	1.23E-06	<1E-10	<1E-10	<1E-10	6.34E-04
5,000–10,000	3.45E-02	4.47E-06	4.34E-07	<1E-10	<1E-10	<1E-10	3.45E-02
10,000–50,000	1.91E-01	1.25E-02	1.21E-03	6.47E-10	4.18E-09	<1E-10	1.91E-01
50,000–100,000	1.05E-03	3.73E-02	3.63E-03	6.95E-09	4.49E-08	<1E-10	4.10E-02
100,000–500,000	1.83E-03	1.47E-02	1.43E-03	7.36E-09	4.75E-08	<1E-10	1.61E-02
>500,000	6.33E-04	9.96E-04	3.82E-04	3.65E-09	2.08E-08	<1E-10	1.94E-03

a. Doses presented as less than 1E-10 are essentially zero.

b. The total is the maximum total dose during that time period and not the sum of the maximum doses shown for each CSSF. The maximums for each CSSF can be at different times during the time period. Total values include the time-dependent contributions from each of the CSSF sources.

CSSF Calcined Solids Storage Facility

7.1.3.1 Uncertainty and Sensitivity Results for the Groundwater All-Pathways Dose

This subsection presents results of an uncertainty and sensitivity analysis performed for the groundwater pathway annual all-pathways dose analysis at the CSSF. The analysis included six OFAT sensitivity cases and a parametric uncertainty and sensitivity analysis. OFAT sensitivity analyses examined the sensitivity of the output variable (annual all-pathways dose) to various assumptions and parameters. A separate parametric uncertainty analysis using Monte Carlo sampling methods was applied to the significant radionuclides and evaluated uncertainty in the all-pathways dose in relation to uncertainty in infiltration rates, stainless-steel corrosion rates, radionuclide inventories, aquifer parameters, and radionuclide transport parameters. The following six OFAT sensitivity analysis cases, described more fully in Section 5 of this document and the CSSF PA/CA, were examined:

- Case 1: Failure of the containment structure due to a seismic event
- Case 2: No containment provided by the stainless-steel bins
- Case 3: 25.4 cm (10 in.) of calcine remains in the bins
- Case 4: Higher infiltration as a result of climate change
- Case 5: No credit for beneficial geochemical conditions of grout
- Case 6: Sensitivity of the grouted waste layer thickness.

Figure 7-4 (Cases 1 through 5) and Table 7-2 (Cases 1 through 6) summarize the results of the OFAT sensitivity analysis. All doses during the institutional control period, 1,000-year post-closure period, and beyond the 1,000-year post-closure period were less than the 25-mrem annual all-pathways ED performance objective.

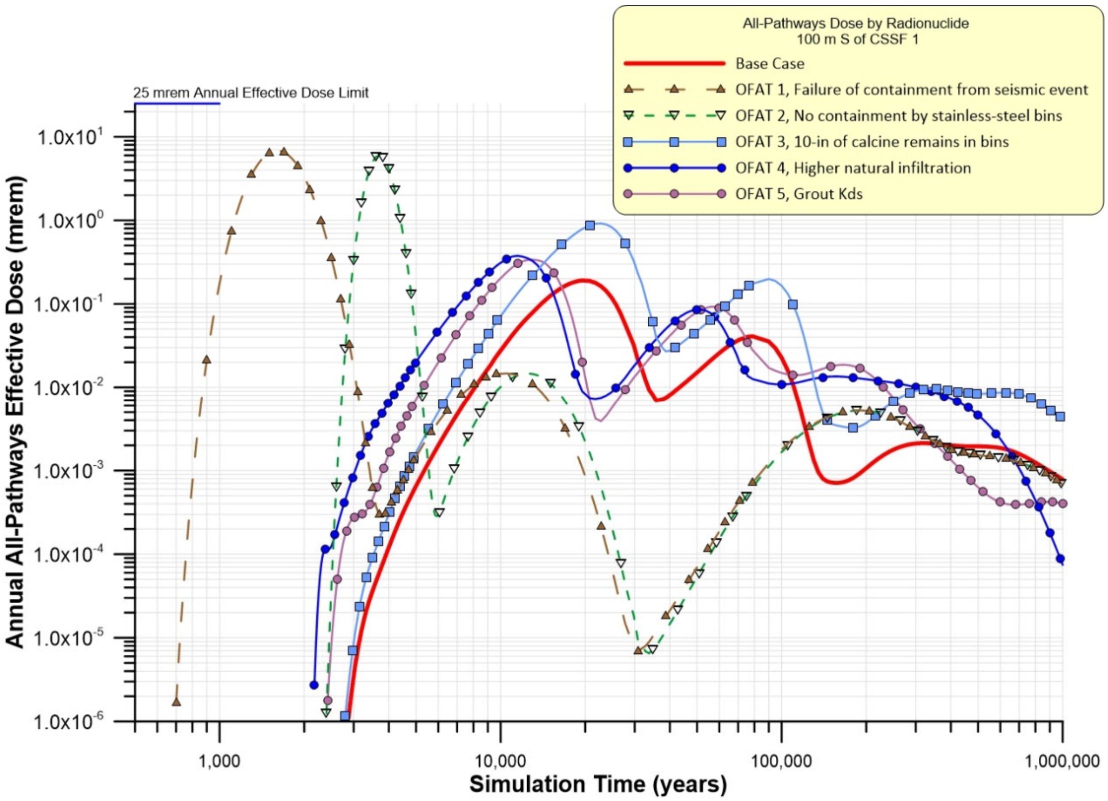

Figure 7-4. Groundwater annual all-pathways effective dose as a function of time for the base case and one-factor-at-a-time Cases 1 through 5.

Table 7-2. Summary of peak groundwater annual all-pathways effective dose for the six one-factor-at-a-time cases and the ratio of the peak one-factor-at-a-time effective dose to the base-case effective dose.

OFAT Case and Description	Peak Groundwater Annual All-Pathways Effective Dose During 1,000-yr Period After Closure (mrem)[a]	Peak Groundwater Annual All-Pathways Effective Dose for All Times (mrem)	Ratio of Peak Effective Dose to Base Case Effective Dose
Base case	<1.0E-10	1.91E-01	1.0
1. Failure of the containment structure due to a seismic event	6.93E-02	6.82E+00	35.7
2. No containment provided by the stainless-steel bins	<1.0E-10	5.96E+00	31.2
3. 25.4 cm (10 in.) of calcine remains in the bins	<1.0E-10	9.19E-01	4.81
4. Higher infiltration as a result of climate change	<1.0E-10	3.77E-01	1.97
5. No credit for beneficial geochemical conditions of grout	<1.0E-10	3.38E-01	1.77
6. Sensitivity of the grouted waste layer thickness	b	b	1.35[c]

a. Doses presented as less than 1.0E-10 are essentially zero.

b. The all-pathways dose was not calculated for this OFAT case, only grouted waste layer Tc-99 flux from CSSF 1.

c. The grout in CSSF 1 was discretized into fifteen 0.1016-m (4-in.) layers, and the entire Tc-99 inventory was assigned to the lowest layer. The peak annual flux was 4.2E-04 Ci, whereas the base-case peak annual flux was 3.1E-04 Ci. Thus, a decrease in the grouted waste layer thickness by a factor of 3 resulted in an increase in the peak flux by a factor of 1.35.

CSSF Calcined Solids Storage Facility OFAT one factor at a time

7.1.4 Cumulative Atmospheric Pathway Dose Analysis Results

Dose results from atmospheric releases from the CSSF were substantially below the 10-mrem performance objective (see Table 5-3 and Figure 7-5). The annual ED for an MEI located off the INL Site during the period of assumed institutional control was 1.35E-04 mrem. Tritium dominates the dose during the institutional control period, primarily because it is assumed to be released rapidly. Notably, if H-3 was actually released at the rate assumed, most of the H-3 would be gone before the CSSF would close. After institutional control and during the 1,000-year post-closure period, the maximum annual ED of 6.66E-06 mrem at the 100-m (328-ft) receptor also is substantially less than the 10-mrem annual performance objective. The chemical forms assumed for the radionuclides were selected to maximize release and transport. Thus, this assessment represents a conservative estimate of doses from atmospheric releases. The atmospheric all-pathways dose calculations and results are presented in the CSSF PA/CA (DOE-ID 2022a).

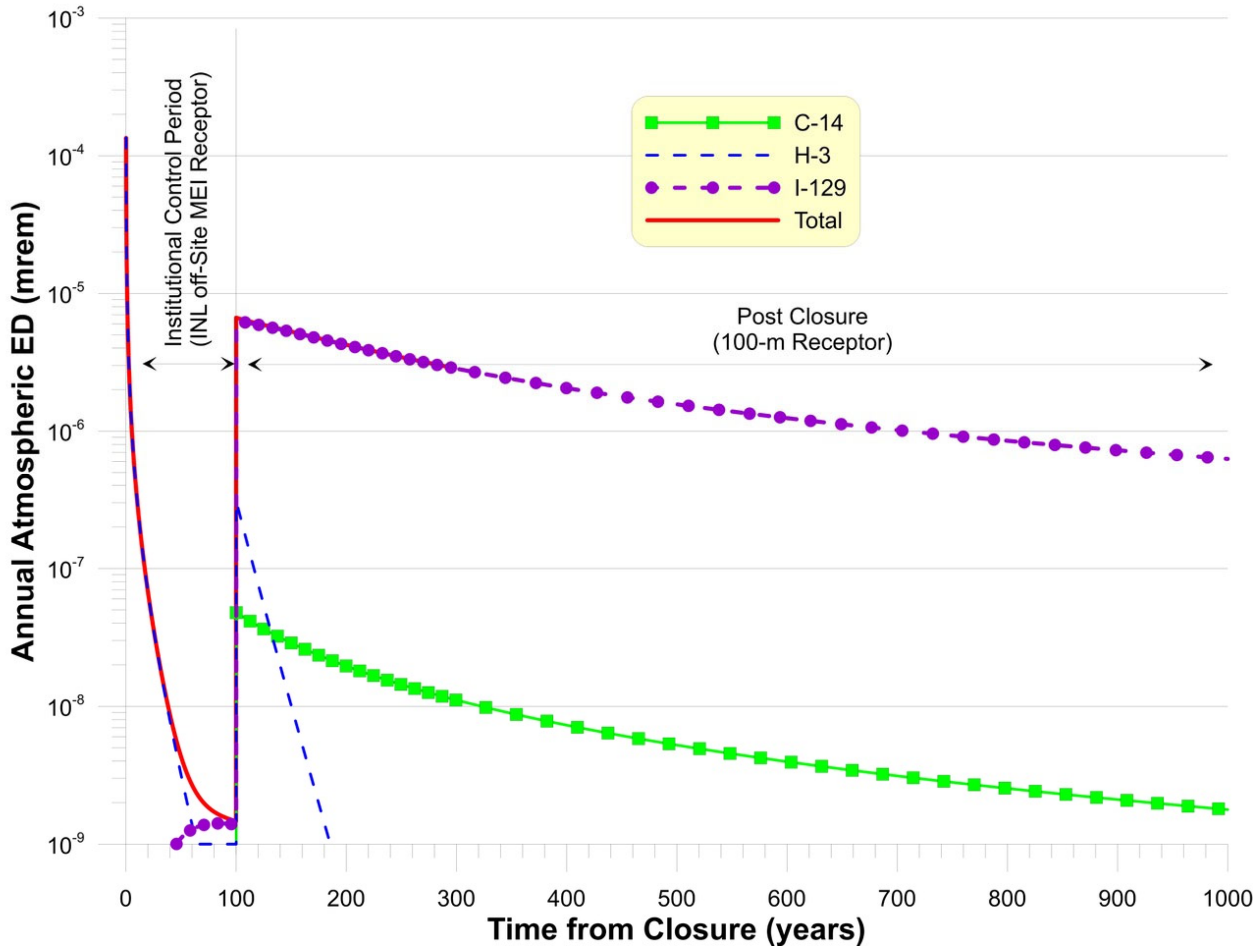

Figure 7-5. Atmospheric annual all-pathways effective dose as a function of time from closure of the Calcined Solids Storage Facility. Dose during the institutional control period is for a maximally exposed individual located off the Idaho National Laboratory Site. After the institutional control period, the dose is calculated for a receptor 100 m (328 ft) from the source. (Note: Concentrations in the air continue to decrease after the 1,000-year post-closure period shown in the figure.)

7.1.5 Total All-Pathway Dose Results

This subsection summarizes results of the total annual all-pathways dose (combined groundwater and atmospheric pathways). The groundwater pathway is addressed in Subsection 7.1.3, and the air-pathway-based component of the all-pathways dose analysis is addressed in Subsection 7.1.4. The results of the all-pathways (combined groundwater and atmospheric) are provided in Figure 7-6.

All-pathways doses for CSSF receptors were all less than the 25-mrem annual dose performance objective for the 10,000-year post-closure period, with a maximum annual dose of 3.00E-02 mrem at 10,000 years. The peak annual dose of 9.00E-01 mrem occurs at 19,500 years.

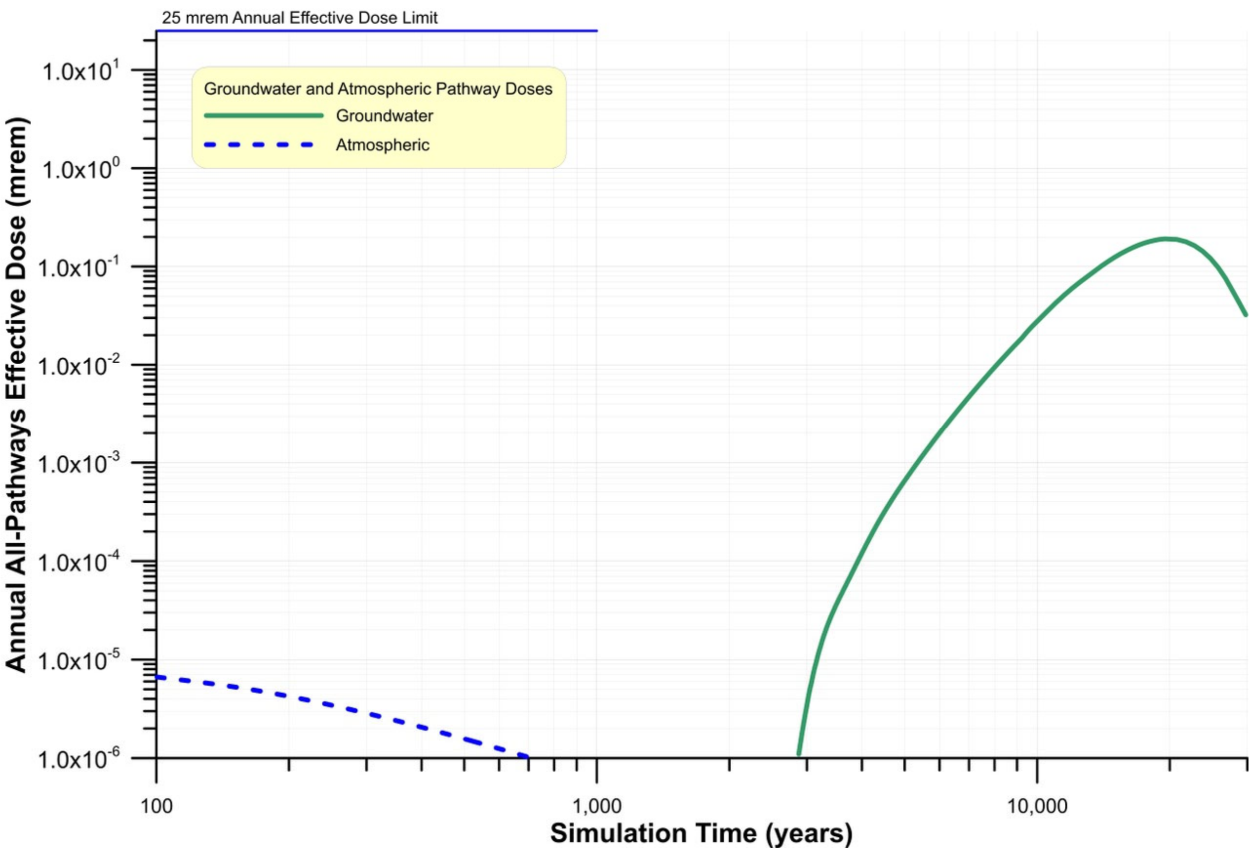

Figure 7-6. Total annual all-pathways effective dose (combined groundwater and atmospheric pathways) as a function of time from closure of the Calcined Solids Storage Facility.

7.1.6 Radon Flux Screening Analysis Results

For additional information, the radon transport from the residual waste at the CSSF is compared to the DOE M 435.1-1 Chg 3 radon flux standard of 20 pCi/m^2/second. The residual waste in the CSSF bins is a minimum of 13.7 m (45 ft) bls. This depth precludes the release of Rn-222 from the bins due to the decay of the radon gas before reaching the ground surface. Therefore, only the residual waste in the transport lines, which are approximately 3 m (10 ft) bls or deeper, are considered in this radon analysis.

A total of 613.3 m (2,012 ft) of piping is in place at the CSSF; however, only 23.8 m (78 ft) of the transport lines have potentially accumulated radioactive calcine (DOE-ID 2022a) (see details in Subsection 2.11.1.7). The transport lines where radioactive calcine may have accumulated were used in the analysis to bound (i.e., pessimistic analysis) potential impacts from radon from the residual waste.

The screening analysis for the radon flux was evaluated for a simplified, degraded system consisting of only the residual waste in the transport lines and an overlying 3 m (10 ft) of soil material (Figure 7-7). In reality, residual waste in the transport lines is at least approximately 3 m (10 ft) bls. The transport lines, the pipe walls, and concrete encasements were not considered in the analysis.

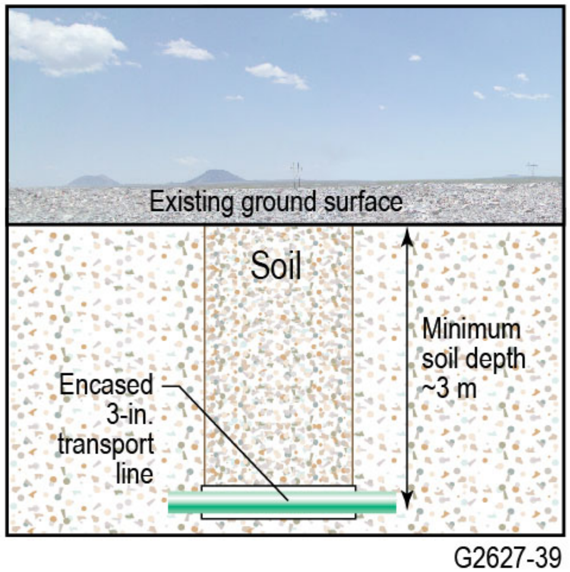

Figure 7-7. Conceptualization of the Calcined Solids Storage Facility transport lines for the radon pathway screening model.

The transport lines were evaluated for their radon flux contributions. The largest radon flux for the transport lines occurred for the CSSF 1 radionuclide inventory. The CSSF 1 source term was included for this bounding, screening analysis even though EDF-11119 indicates that no potential accumulation of radioactive calcine in the transport lines is composed of waste from CSSF 1. The source term for a partially filled 1-m-long (3.3-ft-long), 3-in.-diameter transport line was assumed to be spread over a 1-m^2 (10.76-ft^2) area. The results for a partially filled transport line are shown in Figure 7-8 and Table 7-3.

The maximum peak radon flux for a CSSF transport line assumed to be partially filled with radioactive calcine, within the 1,000-year post-closure period, was estimated to be 4.35E-02 pCi/m^2/second at the surface above the pipe, assuming the pipe was partially filled to capacity with CSSF 1 inventory. This flux is below the DOE M 435.1-1 Chg 3 performance standard of 20 pCi/m^2/second. The result, given the pessimistic bias in the screening analysis, which is intended to overpredict radon flux estimates, provides confidence in the long-term performance of the CSSF from the perspective of the radon flux performance objective.

7.1.7 Biotic Pathway Screening Analysis Results

Biointrusion into the CSSF bins is not expected due to the depth of the residual waste. The residual waste in the bins is a minimum of 13.7 m (45 ft) bls, which is much greater than the depth of roots and animal burrows at the INL Site (i.e., a maximum of 2.25 and 2.70 cm [89 and 106 in.], respectively). Details of the CSSF biointrusion pathway screening analysis are documented in EDF-11173.

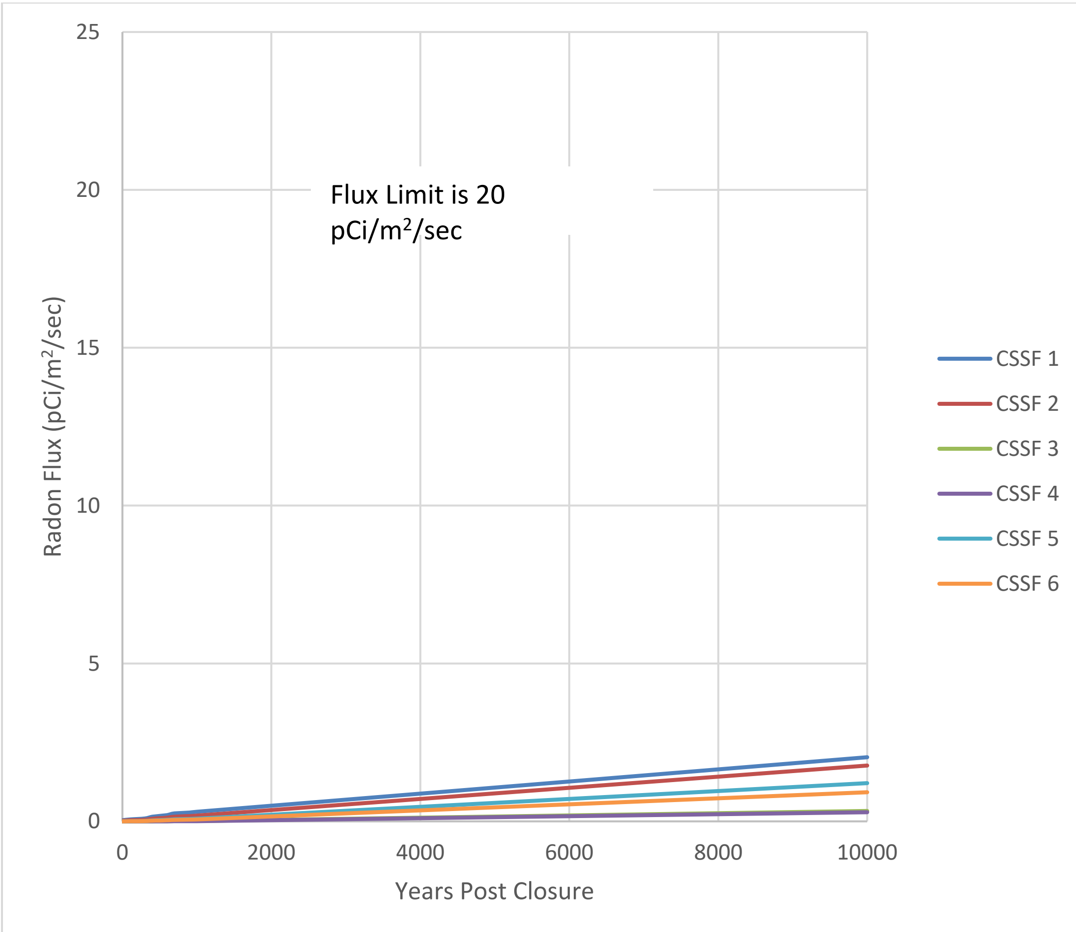

Figure 7-8. Radon flux at the surface of a transport line assumed to be partially filled with radioactive calcine.

Table 7-3. Results for the radon flux at the surface of a transport line assumed to be partially filled with radioactive calcine.

Time Post Closure (yr)	Radon Flux ($pCi/m^2/sec$)					
	CSSF 1	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
0	3.6E-02	2.1E-02	2.5E-04	9.7E-05	1.5E-03	1.0E-03
100	6.4E-02	3.7E-02	9.8E-04	3.2E-04	9.1E-03	6.3E-03
300	9.3E-02	5.3E-02	1.9E-03	7.0E-04	1.7E-02	1.2E-02
400	1.5E-01	8.5E-02	4.1E-03	1.9E-03	3.3E-02	2.4E-02
500	1.7E-01	1.0E-01	5.3E-03	2.7E-03	4.2E-02	3.0E-02
600	2.0E-01	1.2E-01	6.8E-03	3.7E-03	5.1E-02	3.6E-02
700	2.5E-01	1.5E-01	1.0E-02	5.9E-03	6.9E-02	4.9E-02
900	2.8E-01	1.7E-01	1.2E-02	7.2E-03	7.8E-02	5.6E-02
1,000	3.0E-01	1.8E-01	1.4E-02	8.6E-03	8.8E-02	6.3E-02
10,000	2.0E+00	1.8E+00	3.3E-01	2.9E-01	1.2E+00	9.2E-01

CSSF Calcined Solids Storage Facility

7.1.8 ALARA Analysis Results

The NRC performance objective in 10 CFR 61.41 also provides that "Reasonable effort should be made to maintain releases of radioactivity in effluents to the environment as low as reasonably achievable." The CSSF PA/CA (DOE-ID 2022a) was developed in accordance with the comparable requirement in DOE M 435.1-1 Chg 3, which states:

> Performance assessments shall include a demonstration that projected releases of radionuclides to the environment shall be maintained as low as reasonably achievable (ALARA).

As discussed previously, the CSSF PA/CA provides information to demonstrate compliance with the 25-mrem annual all-pathways ED performance objective, including stabilization of residual waste using grout to minimize releases to the environment. Section 5 of this Draft CSSF 3116 Basis Document provides information to show that HRRs in the CSSF will have been removed to the maximum extent practical at closure.

In addition to removal of HRRs to the maximum extent practical, other CSSF closure design features support meeting the ALARA objectives set forth in 10 CFR 61.41. Closure of the CSSF will stabilize remaining structures, and void spaces will be filled with grout. Grout is the most commonly used material for stabilizing radioactive waste (DOE-ID 2022a). The CSSF closure design will stabilize residual waste, minimize infiltration of water through the bins and transport lines, and provide long-term stability. These features will serve to impede release of stabilized contaminants into the general environment. Specifically, engineered barriers (reinforced-concrete vaults, grouted void space inside and outside of the bins, and stainless-steel bins and transfer lines) will minimize infiltration of water to prevent migration.

For the base-case groundwater analysis, the annual all-pathways EDs were predicted to be at least a factor of 132 below the 25-mrem annual standard for the 1 million years simulated in the CSSF PA/CA. These results show that the estimated individual dose from release of radionuclides from the CSSF would be less than the reference dose of 1 mrem in a year,[65] which is established as a threshold for a quantitative ALARA analysis (DOE-HDBK-1215-2014). Based on the low values of estimated individual and collective doses relative to reference values, only a qualitative ALARA analysis is required in accordance with DOE-HDBK-1215-2014, "DOE Handbook Optimizing Radiation Protection of the Public and the Environment for use with DOE O 458.1, ALARA Requirements." Closure of the CSSF was considered as part of the NEPA process for HLW management at the INL Site (DOE 2002). The Final EIS (DOE 2002) and additional documentation were used as the basis for the 2005 ROD (DOE 2005). This information included cost- and impact-based evaluations of different alternatives, which are also considerations for an ALARA analysis. The Final EIS evaluated several options for facility disposition, and performance-based closure was selected as the preferred alternative (DOE 2005). Given that (1) the magnitude of the projected doses at levels for which a qualitative ALARA analysis is sufficient and (2) the formal process that has been followed to consider alternatives, the ALARA requirement will be satisfied.

Subsections 7.3.10 and 7.3.11 provide additional discussion relative to compliance with the ALARA performance objective set forth in 10 CFR 61.43.

7.1.9 Conclusion for Dose Representative Member of the Public

The peak groundwater annual all-pathways ED for the CSSF PA/CA base case during the 1,000-year post-closure period was 6.79E-14 mrem (less than 1E-10 and thus essentially zero) and is dominated by H-3. Thereafter, the peak annual ED is 1.91E-01 mrem in simulation year 19,500, primarily due to Tc-99. The groundwater sensitivity analysis for OFAT Case 1 also provided a maximum dose of 6.93E-02 mrem during the 1,000-year post-closure period and a maximum dose of 6.82E+00 mrem after the 1,000-year

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65. As noted in DOE-HDBK-1215-2014, if the dose to the MEI, or the representative person of the critical group, is much less than 1 mrem (0.01 mSv) in a year, only a qualitative ALARA analysis is warranted.

post-closure period. In addition, the combined total annual all-pathways doses (combined groundwater and atmospheric) were less than the 25-mrem annual ED, with a dose of 1.4E-04 mrem during the institutional control period and a dose of 6.7E-06 mrem during the 1,000-year post-closure period. Therefore, reasonable assurance is provided that the performance objective of 25 mrem from 10 CFR 61.41 will not be exceeded. The additional performance requirements in DOE M 435.1-1 Chg 3 stipulating (1) the air pathway annual ED limit of 10 mrem, excluding the dose from radon and its progeny, and (2) that the release of radon shall be less than an average flux of 20 pCi/m^2/second at the surface of the disposal facility also will not be exceeded. CSSF closure will support long-term stability and minimize contaminant release to the environment consistent with the ALARA objective set forth in 10 CFR 61.41.

7.2 Protection of Inadvertent Intruders

Provisions in 10 CFR 61.42 require the following:

> Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

For additional information, Chapter IV.P(2)(h) of DOE M 435.1-1 Chg 3 for protection of individuals from inadvertent intrusion reads as follows:

> For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air.

10 CFR 61.42 shows NRC's intent to protect persons who inadvertently intrude into the waste. While the performance objective does not establish quantitative limits on exposure, the Final EIS for 10 CFR 61 (NUREG-0945) (NRC 1982) suggests a dose limit of 500 mrem for the waste classification scheme in 10 CFR 61.55. By way of guidance, NRC uses a 500-mrem dose limit for evaluating impacts to an inadvertent intruder for purposes of 10 CFR 61.42 (NRC 1982, 2007). To demonstrate reasonable assurance that the 10 CFR 61.42 performance objective will be met, NRC's peak intruder dose limit of 500 mrem is used.

Neither DOE M 435.1-1 Chg 3 nor 10 CFR 61.42 specify use of a particular scenario to demonstrate compliance. In developing intruder scenarios, DOE assumes that humans will continue land use activities that are consistent with past (e.g., recent) decades and present regional practices after the end of the assumed active institutional control period. The following subsections describe the inadvertent intruder analysis in the CSSF PA/CA (DOE-ID 2022a).

7.2.1 Inadvertent Intruder Analysis

The intruder dose assessment in the CSSF PA/CA assumes that a hypothetical inadvertent intruder at the CSSF could be exposed in two stylized scenarios. The first scenario involves acute (i.e., short-term) exposure during an assumed initial contact with radioactive waste, and the second scenario involves chronic (i.e., long-term) exposure to waste after an intruder is assumed to take up permanent residence on the disposal site. The assumed base case intruder scenarios are identified as an acute drilling scenario and a chronic post-drilling agriculture scenario. Intruder scenarios were evaluated for each of the CSSF bins and transport lines.

The intruder analysis was conducted based on the assumption that a closure cover will not be constructed. Therefore, the residual waste in the bins is a minimum of 13.7 m (45 ft) bls, which is much greater than the depth of the excavation of 3 m (10 ft) for a basement. The intruder analysis assumes that transport lines located above the ground surface and down to 3 m (10 ft) bls will be removed during closure activities. Therefore, based on these depths and assumptions, the basement excavation scenario is not considered applicable.

Additionally, this inadvertent intruder dose assessment assumed that one-twenty-fifth (3.9 %) of the transport line volume was filled with residual waste based on the information provided in EDF-11119. The transport line contamination is based on an estimated 23.8 m (78 ft) of piping containing residual waste, averaged over the total length of 613.3 m (2,012 ft) of piping at the CSSF. Though some of the transport lines may potentially have areas that have some residual accumulations of waste (see Subsections 2.11.1.7 and 7.2), the probability of drilling into one of these areas is small in comparison to the overall length of lines at the CSSF (see Subsection 7.2.3). The inventory used for the transport line analysis was based on the residual inventory for each CSSF as provided in Subsection 2.11.3. The use of the CSSF inventories (except for CSSF 1) for the transport line intruder assessment ensures that the range of residual waste volumes potentially contained in the transport lines is captured in the analysis. The CSSF 1 inventory was excluded from the transport line intruder assessment because the section of line that was partially filled with CSSF 2 cold startup material was removed during the HWMA/RCRA closure and the remaining empty transport line to CSSF 1 was grouted. A summary of inadvertent intruder exposure scenarios is presented in Table 7-4.

Each CSSF consists of several stainless-steel storage bins housed within a concrete vault. The reinforced-concrete vaults range in thickness from 0.53 to 2 m (1.75 to 6.5 ft). The bins are constructed of Type 405, 304, and 304L stainless-steel that has a minimum thickness of 3.18 mm (0.125 in.), 6.4 mm (0.25 in.), and 9.53 mm (0.375 in.), respectively. The fact that the bins are all constructed of stainless steel and contained in reinforced-concrete vaults provides assurance that any calcine left in the storage bins after retrieval operations has a reduced potential for inadvertent intrusion for a very long time. However, it is pessimistically assumed that inadvertent intrusion into the bins occurs at 500 years in this analysis. Subsection 7.2.3 presents a qualitative discussion of the likelihood of inadvertent human intrusion after closure of the CSSF.

The transport lines used to transfer solids (calcine) from WCF and NWCF to the CSSF storage bins vary in size, length, and depth for each line, as described in EDF-11119. The transport lines travel in a larger containment pipe that is encased in reinforced concrete for shielding. The configuration of the stainless-steel transport lines surrounded by reinforced concrete is considered to provide a robust barrier that precludes inadvertent intrusion directly into the residual waste until 500 years post-closure. The 3-in. transport lines are Schedule 40 stainless-steel pipe with a thickness of 0.216 in. (5.49 mm). The average corrosion rate for Type 304 stainless steel for CSSF calcine-specific corrosion coupons was found to be 2.06E-04 mm/year (see Table 4-4 of the CSSF PA/CA [DOE-ID 2022a]). At this rate, it would take 26,650 years to corrode through the total thickness of the pipe wall. This corrosion rate is considered to be bounding, since corrosion rates in INL Site soil and concrete are expected to be much lower than the corrosion rates based on corrosion coupons in calcine.

Table 7-4. Summary of exposure scenarios for inadvertent intrusion into the Calcined Solids Storage Facility bins and transport lines.

Exposure Scenario	Description
Drilling (acute exposure)	Assumed to occur any time after 500 years post-closure for the bins and transport lines. Exposure to the residual radioactive waste is assumed to occur as a result of drilling an agricultural (i.e., large-diameter) well through the bins or transport lines. Evaluated exposure pathways include external, inhalation, and soil ingestion for a time period required to complete the well.
Post-drilling agriculture (chronic exposure)	Assumed to occur any time after 500 years post-closure for the bins and transport lines. Exposure to the residual radioactive waste is assumed to occur as a result of drilling a residential water supply well through the bins or transport lines, mixing exhumed drill cuttings and waste with garden soil, and using the soil for growing crops and beef. Assumed exposure pathways include direct exposure to contaminated soil, inhalation of contaminated soil, ingestion of contaminated garden soil, ingestion of vegetables grown in contaminated garden soil, and ingestion of contaminated beef and milk.

7.2.1.1 Background on 500-Year Timing for Inadvertent Intrusion

The screening level intruder analysis used for the Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory (DOE-ID 2003b) conservatively assumed that drilling occurred at 100 years immediately after loss of institutional control. In addition, for the purposes of screening in that PA, a bounding assumption was made essentially assuming that the reinforced concrete, stainless-steel bin, and grout were not present to deter the intruder. For the CSSF bins and transport lines, it has been decided to move away from the use of a screening intruder analysis and use an assumption that the reinforced concrete will reasonably be expected to provide an extended deterrence against inadvertent drilling. This is consistent with DOE's technical standard (DOE-STD-5002-2017), which states that reinforced concrete can serve as a robust barrier against intrusion while it maintains its integrity. For the CSSF inadvertent intrusion analysis, the reinforced concrete is assumed to deter inadvertent drilling for 500 years.

DOE's minimum 500-year time frame is consistent with the minimum generic time frame that was considered in the context of 10 CFR 61. NRC used 500 years as the generically applicable time of effectiveness for robust reinforced-concrete barriers when defining the waste classification system and limits for Class C LLW disposal in 10 CFR 61, and the 500-year time period is routinely accepted for disposal of Class C waste at commercial disposal facilities. The *Update of Part 61 Impacts Analysis Methodology* (Oztunali and Roles 1986) provides the description of the generically applicable drilling scenario that is used for inadvertent intrusion. The applicability of the generic intruder scenarios as a function of time and structural stability as presented in Table 5-2 of Oztunali and Roles (1986) is shown in Table 7-5. Subsection 4.2.1 of Oztunali and Roles (1986) also includes specific considerations for disposal methods involving extensive use of reinforced concrete. Namely, it states:

> It is assumed that drilling through reinforced concrete would be sufficiently difficult, or sufficiently out of the ordinary, that the drilling crew would stop and shift to a different drilling location. No, or comparatively little, impacts from the drilling scenario would occur. This assumption is assumed to be applicable for only so long as the reinforced concrete structure can be assumed to be structurally stable. This limiting time period, as in the following Section 4.2.3 for the intruder discovery scenario, is assumed to be 500 years following the end of the surveillance period. Following this 500 year period, the scenario is assumed to be fully applicable.

Notably, the assumption identified in the NRC scenario that a drilling crew would try moving to a different location if reinforced concrete is encountered is particularly relevant for the CSSF bins, where moving several feet or, in the case of the transport lines, simply moving 1 ft one way or the other could avoid the reinforced concrete and waste.

Table 7-5. Matrix of intrusion scenario generic applicability as a function of structural stability and time (Table 5-2 in Oztunali and Roles [1986]).

Condition	Intrusion Scenarios		
	Drilling	Construction	Agriculture
TDEL[a] <500 years			
Unstable waste[b]	Yes	Yes	Yes
Stable waste	Yes	No	No
Grouted waste	Yes	No	No
RC disposal	No	No	No
TDEL[a] >500 years			
Unstable waste[b]	Yes	Yes	Yes
Stable waste	Yes	Yes	Yes
Grouted waste	Yes	Yes	Yes
RC disposal	Yes	Yes	Yes

a. TDEL = time after the end of surveillance period (e.g., institutional control period)

b. Unsegregated, stable waste is considered unstable.

RC reinforced concrete

7.2.2 Inadvertent Intruder Drilling Scenario Analyses Results

Figure 7-9 shows that acute intruder doses between 500 and 10,000 years after closure of the facility for the CSSF bins are well below the dose limit of 500 mrem in a year. Figure 7-10 shows that acute intruder doses between 500 and 10,000 years after closure for the transport lines based on each CSSF inventory are also well below the dose limit of 500 mrem in a year. The largest acute intruder doses for both the CSSF bins and transport lines were found for the CSSF 4 residual waste profile at 500 years post-closure.

The acute intruder drilling scenario yielded a peak dose of 7.1E+00 mrem 500 years post-closure for the bins (Table 7-6). The acute intruder drilling scenario also yielded a peak dose of 5.6E-02 mrem 500 years post-closure for the transport lines (Table 7-7). These doses capture the peak intruder doses because the doses continue to decrease during the 10,000-year post-closure period due to radioactive decay. Am-241, Pu-239, Pu-240, Pu-238, Nb-94, Pu-241, Sn-126, and Np-237 provide most of the intruder doses for the bins and transport lines at 500 years post-closure for the CSSF 4 residual waste profile. Details of the acute intruder doses by radionuclide are provided in EDF-11132, Appendix A-4.

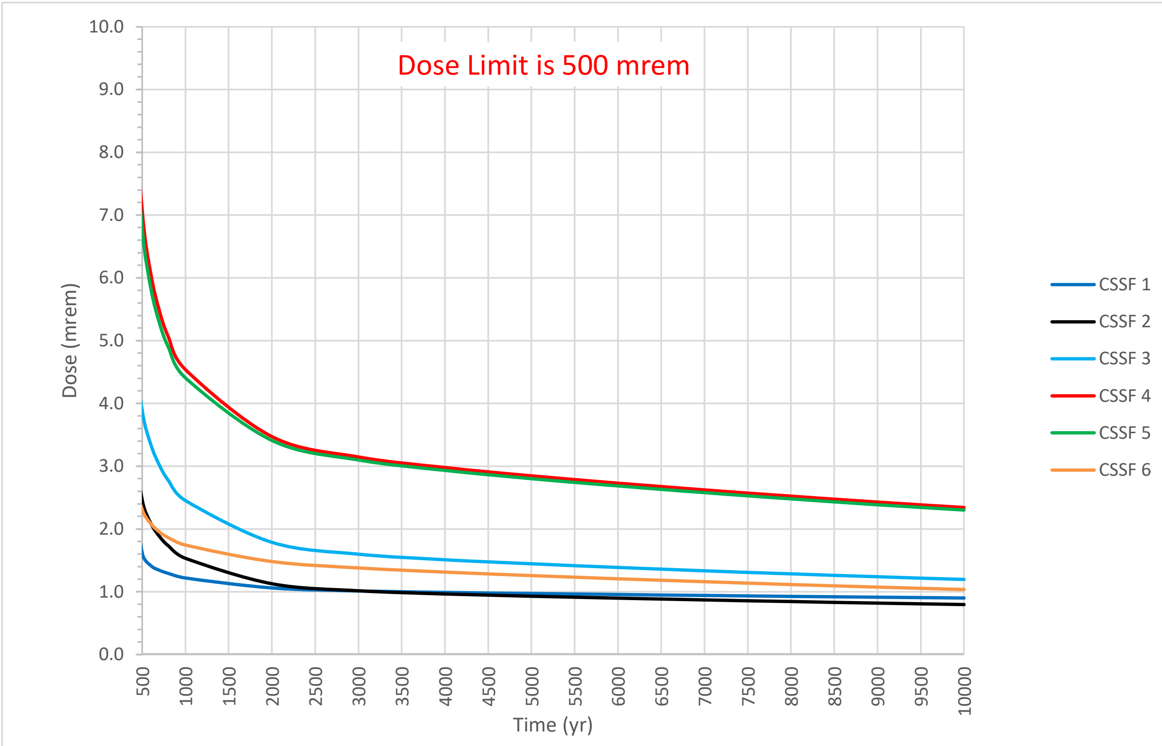

Figure 7-9. Acute intruder scenario dose results for the Calcined Solids Storage Facility bins.

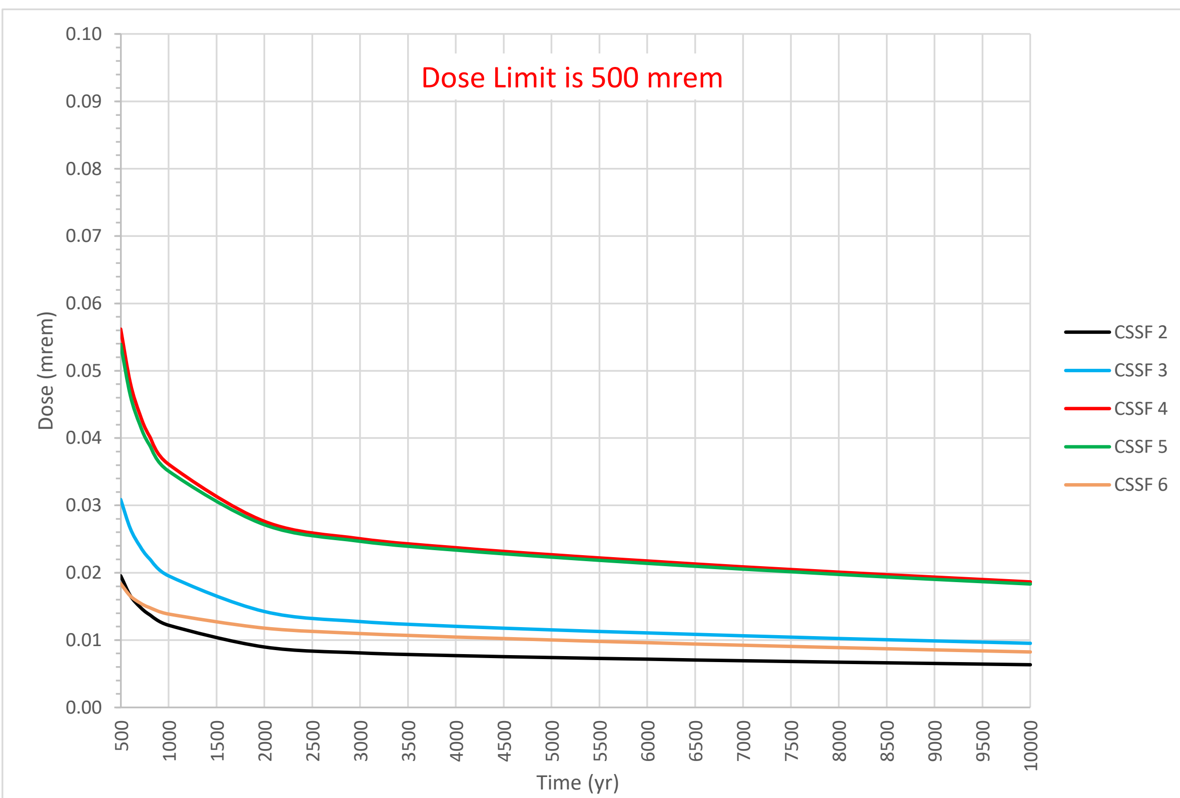

Figure 7-10. Acute intruder scenario dose results for the transport lines to each Calcined Solids Storage Facility (except Calcined Solid Storage Facility 1).

Table 7-6. Acute intruder scenario dose results for the Calcined Solids Storage Facility bins.

Time (yr)	Acute Intruder Doses (mrem)					
	CSSF 1	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	1.6E+00	2.5E+00	3.9E+00	7.1E+00	6.8E+00	2.3E+00
600	1.4E+00	2.1E+00	3.3E+00	6.1E+00	5.8E+00	2.1E+00
700	1.3E+00	1.9E+00	3.0E+00	5.5E+00	5.3E+00	2.0E+00
800	1.3E+00	1.7E+00	2.8E+00	5.1E+00	4.9E+00	1.9E+00
1,000	1.2E+00	1.5E+00	2.5E+00	4.5E+00	4.4E+00	1.7E+00
2,000	1.1E+00	1.1E+00	1.8E+00	3.5E+00	3.4E+00	1.5E+00
3,000	1.0E+00	1.0E+00	1.6E+00	3.1E+00	3.1E+00	1.4E+00
4,000	9.9E-01	9.6E-01	1.5E+00	3.0E+00	2.9E+00	1.3E+00
5,000	9.7E-01	9.3E-01	1.4E+00	2.8E+00	2.8E+00	1.3E+00
6,000	9.5E-01	9.0E-01	1.4E+00	2.7E+00	2.7E+00	1.2E+00
7,000	9.4E-01	8.7E-01	1.3E+00	2.6E+00	2.6E+00	1.2E+00
8,000	9.3E-01	8.4E-01	1.3E+00	2.5E+00	2.5E+00	1.1E+00
9,000	9.1E-01	8.2E-01	1.2E+00	2.4E+00	2.4E+00	1.1E+00
10,000	9.0E-01	8.0E-01	1.2E+00	2.3E+00	2.3E+00	1.0E+00

CSSF Calcined Solids Storage Facility

Table 7-7. Acute intruder scenario dose results for the transport lines to each Calcined Solids Storage Facility (except Calcined Solid Storage Facility 1).

Time (yr)	Acute Intruder Doses (mrem)[a]				
	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	2.0E-02	3.1E-02	5.6E-02	5.4E-02	1.8E-02
600	1.7E-02	2.7E-02	4.8E-02	4.6E-02	1.7E-02
700	1.5E-02	2.4E-02	4.3E-02	4.2E-02	1.6E-02
800	1.4E-02	2.2E-02	4.0E-02	3.9E-02	1.5E-02
1,000	1.2E-02	2.0E-02	3.6E-02	3.5E-02	1.4E-02
2,000	9.0E-03	1.4E-02	2.8E-02	2.7E-02	1.2E-02
3,000	8.1E-03	1.3E-02	2.5E-02	2.5E-02	1.1E-02
4,000	7.7E-03	1.2E-02	2.4E-02	2.3E-02	1.0E-02
5,000	7.4E-03	1.2E-02	2.3E-02	2.2E-02	1.0E-02
6,000	7.1E-03	1.1E-02	2.2E-02	2.1E-02	9.6E-03
7,000	6.9E-03	1.1E-02	2.1E-02	2.1E-02	9.2E-03
8,000	6.7E-03	1.0E-02	2.0E-02	2.0E-02	8.9E-03
9,000	6.5E-03	9.9E-03	1.9E-02	1.9E-02	8.5E-03
10,000	6.3E-03	9.5E-03	1.9E-02	1.8E-02	8.2E-03

a. The CSSF 1 inventory was excluded from the transport line intruder assessment because the section of line that was partially filled with CSSF 2 cold startup material was removed during the HWMA/RCRA closure and the remaining empty transport line to CSSF 1 was grouted.

CSSF Calcined Solids Storage Facility
HWMA Hazardous Waste Management Act
RCRA Resource Conservation and Recovery Act

Figure 7-11 shows that chronic intruder doses between 500 and 10,000 years after closure of the CSSF bins are well below the DOE dose limit of 100 mrem in a year. Figure 7-12 shows that chronic intruder doses between 500 and 10,000 years after closure for the transport lines to each CSSF (except CSSF 1) are also well below the DOE dose limit of 100 mrem in a year. The largest chronic intruder doses for the CSSF bins were found for the CSSF 1 residual waste profile at 500 years post-closure, while the largest chronic doses for the transport lines were found for the CSSF 5 residual waste profile at 500 years post-closure. As noted earlier in this subsection, the CSSF 1 waste profile was not evaluated for the transport lines because the partially filled portion of the CSSF 1 lines had previously been removed during closure of the WCF. Details of the chronic intruder doses by radionuclide are provided in EDF-11132, Appendix A-5.

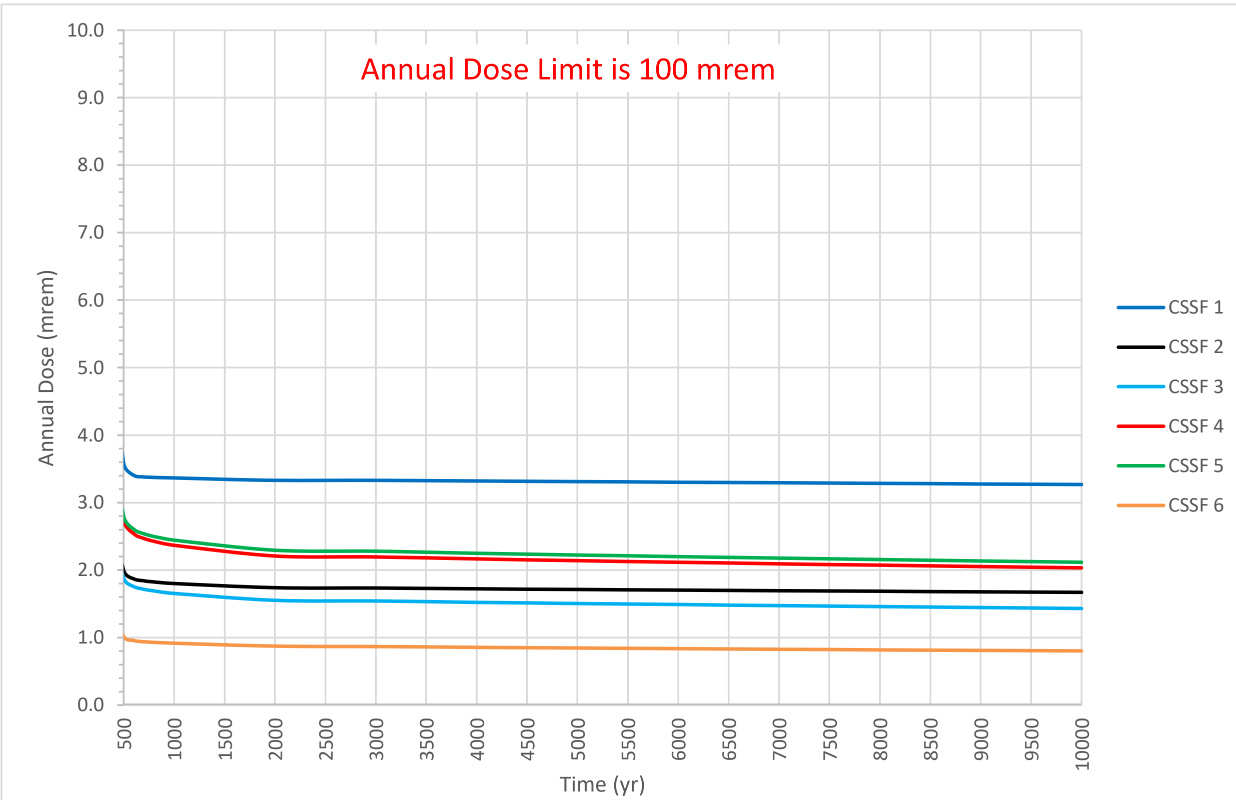

Figure 7-11. Chronic intruder scenario dose results for the Calcined Solids Storage Facility bins.

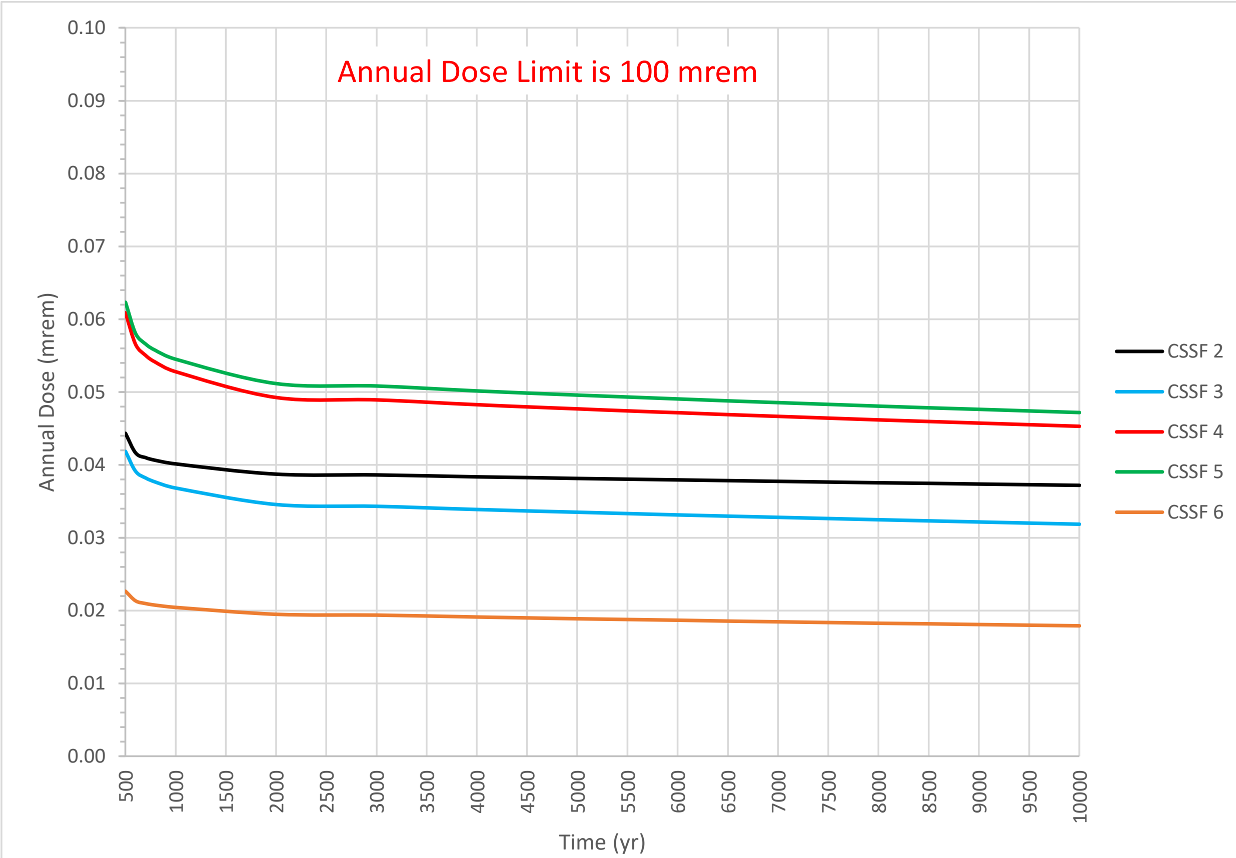

Figure 7-12. Chronic intruder scenario dose results for the transport lines to each Calcined Solids Storage Facility (except Calcined Solids Storage Facility 1).

The chronic intruder post-drilling agriculture scenario yielded a peak dose of 3.6E+00 mrem in a year 500 years post-closure for the CSSF bins (Table 7-8). The chronic intruder post-drilling agriculture scenario yielded a peak dose of 6.2E-02 mrem in a year 500 years post-closure for the transport lines (Table 7-9). These doses capture the peak intruder doses because the doses continue to decrease during the 10,000-year modeling period due to radioactive decay. Tc-99, Sn-126, Cs-137, Sr-90, Am-241, Pu-239, Pu-240, and Np-237 provide most of the contributions to the intruder doses for the CSSF bins, based on the CSSF 1 waste profile. Tc-99, Nb-94, Sn-126, Am-241, Pu-239, Pu-240, Pu-238, Cs-137, Np-237, Sr-90, and Pu-241 provide most of the contributions to the intruder doses for the transport lines, based on the CSSF 5 waste profile. It is noted that Cs-137 and Sr-90 continue to provide doses in the chronic intruder scenarios even after 500 years. This is because the initial inventories for Cs-137 (7.6E+03 Ci) and Sr-90 (7.0E+03 Ci) are the largest at the CSSF such that after 500 years of decay, they provide sufficient inventories (i.e., 8.0E-02 Ci for Cs-137 and 3.6E-02 for Sr-90), resulting in chronic intruder dose contributions.

The results of the intruder analysis should be considered within the context of the actual likelihood of inadvertent intrusion at the CSSF. The assumption that drilling a well at the CSSF will result in inadvertent penetration of a bin or transport line, and that the residual waste will not be recognized as different from soil and will be brought to the surface for subsequent human exposure, is considered to be pessimistic. Additionally, the likelihood of drilling into contamination at the CSSF is considered extremely low, as discussed in Subsection 7.2.3.

Table 7-8. Chronic intruder scenario dose results for the Calcined Solids Storage Facility bins.

Time (yr)	Chronic Intruder Annual Dose (mrem)					
	CSSF 1	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	3.6E+00	2.0E+00	1.9E+00	2.7E+00	2.8E+00	1.0E+00
600	3.4E+00	1.9E+00	1.8E+00	2.5E+00	2.6E+00	9.6E-01
700	3.4E+00	1.8E+00	1.7E+00	2.5E+00	2.5E+00	9.4E-01
800	3.4E+00	1.8E+00	1.7E+00	2.4E+00	2.5E+00	9.3E-01
1,000	3.4E+00	1.8E+00	1.7E+00	2.4E+00	2.4E+00	9.2E-01
2,000	3.3E+00	1.7E+00	1.6E+00	2.2E+00	2.3E+00	8.7E-01
3,000	3.3E+00	1.7E+00	1.5E+00	2.2E+00	2.3E+00	8.7E-01
4,000	3.3E+00	1.7E+00	1.5E+00	2.2E+00	2.3E+00	8.6E-01
5,000	3.3E+00	1.7E+00	1.5E+00	2.1E+00	2.2E+00	8.5E-01
6,000	3.3E+00	1.7E+00	1.5E+00	2.1E+00	2.2E+00	8.4E-01
7,000	3.3E+00	1.7E+00	1.5E+00	2.1E+00	2.2E+00	8.3E-01
8,000	3.3E+00	1.7E+00	1.5E+00	2.1E+00	2.2E+00	8.2E-01
9,000	3.3E+00	1.7E+00	1.4E+00	2.1E+00	2.1E+00	8.1E-01
10,000	3.3E+00	1.7E+00	1.4E+00	2.0E+00	2.1E+00	8.0E-01

CSSF Calcined Solids Storage Facility

Table 7-9. Chronic intruder scenario dose results for the transport lines to each Calcined Solids Storage Facility (except CSSF 1).

Time (yr)	Chronic Intruder Annual Dose (mrem)[a]				
	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	4.4E-02	4.2E-02	6.1E-02	6.2E-02	2.3E-02
600	4.2E-02	3.9E-02	5.7E-02	5.8E-02	2.1E-02
700	4.1E-02	3.8E-02	5.5E-02	5.7E-02	2.1E-02
800	4.1E-02	3.8E-02	5.4E-02	5.6E-02	2.1E-02
1,000	4.0E-02	3.7E-02	5.3E-02	5.4E-02	2.0E-02
2,000	3.87E-02	3.46E-02	4.92E-02	5.12E-02	1.95E-02
3,000	3.86E-02	3.43E-02	4.89E-02	5.08E-02	1.94E-02
4,000	3.84E-02	3.39E-02	4.83E-02	5.02E-02	1.91E-02
5,000	3.81E-02	3.35E-02	4.77E-02	4.96E-02	1.89E-02
6,000	3.79E-02	3.32E-02	4.72E-02	4.91E-02	1.87E-02
7,000	3.77E-02	3.28E-02	4.67E-02	4.85E-02	1.85E-02
8,000	3.75E-02	3.25E-02	4.62E-02	4.81E-02	1.83E-02
9,000	3.74E-02	3.22E-02	4.57E-02	4.76E-02	1.81E-02
10,000	3.72E-02	3.19E-02	4.53E-02	4.72E-02	1.79E-02

a. The CSSF 1 inventory was excluded from the transport line intruder assessment because the section of line that was partially filled with CSSF 2 cold startup material was removed during the HWMA/RCRA closure and the remaining empty transport line to CSSF 1 was grouted.

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7.2.3 Likelihood of Drilling a Well into Residual Contamination at the CSSF

Inadvertent human intrusion analyses are hypothetical constructs used to identify wastes appropriate for near-surface disposal and to establish waste classifications (DOE G 435.1-1; NCRP [2015]; NRC [2007]). An intrusion calculation is not intended to represent a realistic calculation of doses to a future member of the public; rather, it is intended as a stylized representation of hypothetical doses to people who may occupy the area of the disposal site[66] or, in this case, the CSSF closure site. The analyses are generally carried out as deterministic calculations with some averaging of waste concentrations intended to overstate the expected consequences by assuming the occurrence of the sequence of events necessary for a full scenario without regard to the likelihood of each event's occurrence. The only credible potential intrusion event at the CSSF is a drilling intrusion, owing to the depth of the residual wastes in the closed facility (greater than 3 m [10 ft] for residual waste in bins and transport lines, even without a cover at the surface). Although an inadvertent drilling intrusion scenario is assumed to occur at some time in the future at the CSSF, it is important to provide perspective on the true likelihood of the events necessary for such a scenario as part of a risk-informed approach to PA.

The following is a qualitative discussion on the likelihood of inadvertent human intrusion after closure of the CSSF. This discussion is intended solely to provide risk-informed context and perspective for the inadvertent intruder results; the specific likelihoods of occurrence are not considered explicitly in the CSSF intruder analysis. Intrusion is pessimistically assumed to occur into transport lines and into the bins with a probability of 1 at 500 years following closure.

For a drilling event to intersect the waste, exhume contamination, and lead to exposures to that contamination, a series of necessary events must occur, as shown in Figure 7-13. These events can generally be regarded as independent, and each can only occur if all of the previous necessary events have occurred. If at any stage of the sequence the necessary event does not occur, the overall intrusion event will not occur. Also shown on the figure are a set of precursor conditions that support understanding of the likelihood of occurrence of each necessary event. The precursor conditions relate to issues or circumstances such as societal change and the motivations and actions of the intruder. As such, they are not readily amenable to rigorous probabilistic calculation, but the evolution of each over the post-closure performance period can be qualitatively assessed, supported by logical arguments.

The initial necessary event leading to intrusion is an assumed loss of institutional control. It is reasonable to assume that as long as DOE and the U.S. Government exist, some form of governmental control over the INTEC area will be maintained as required under DOE O 458.1 Chg 4 and DOE P 454.1 Chg 1,[67] as well as institutional control agreements under CERCLA. Therefore, a precursor condition to the loss of control would be an unforeseen major change in the governance of the United States to the extent that previously established administrative controls would be forgotten or deliberately disregarded. Such possibilities are regarded as low over the 100 years after CSSF closure but could increase over time.

66. 10 CFR 61 defines an inadvertent intruder as a person who might occupy the disposal site after closure and engage in normal activities, such as agriculture, dwelling construction, or other pursuits in which the person might be unknowingly exposed to radiation from the waste. The disposal site is defined as that portion of a land disposal facility which is used for disposal of waste. It consists of disposal units and a buffer zone.

67. This policy specifically states the following: "DOE will maintain the institutional controls as long as necessary to perform their intended protective purposes and seek sufficient funds."

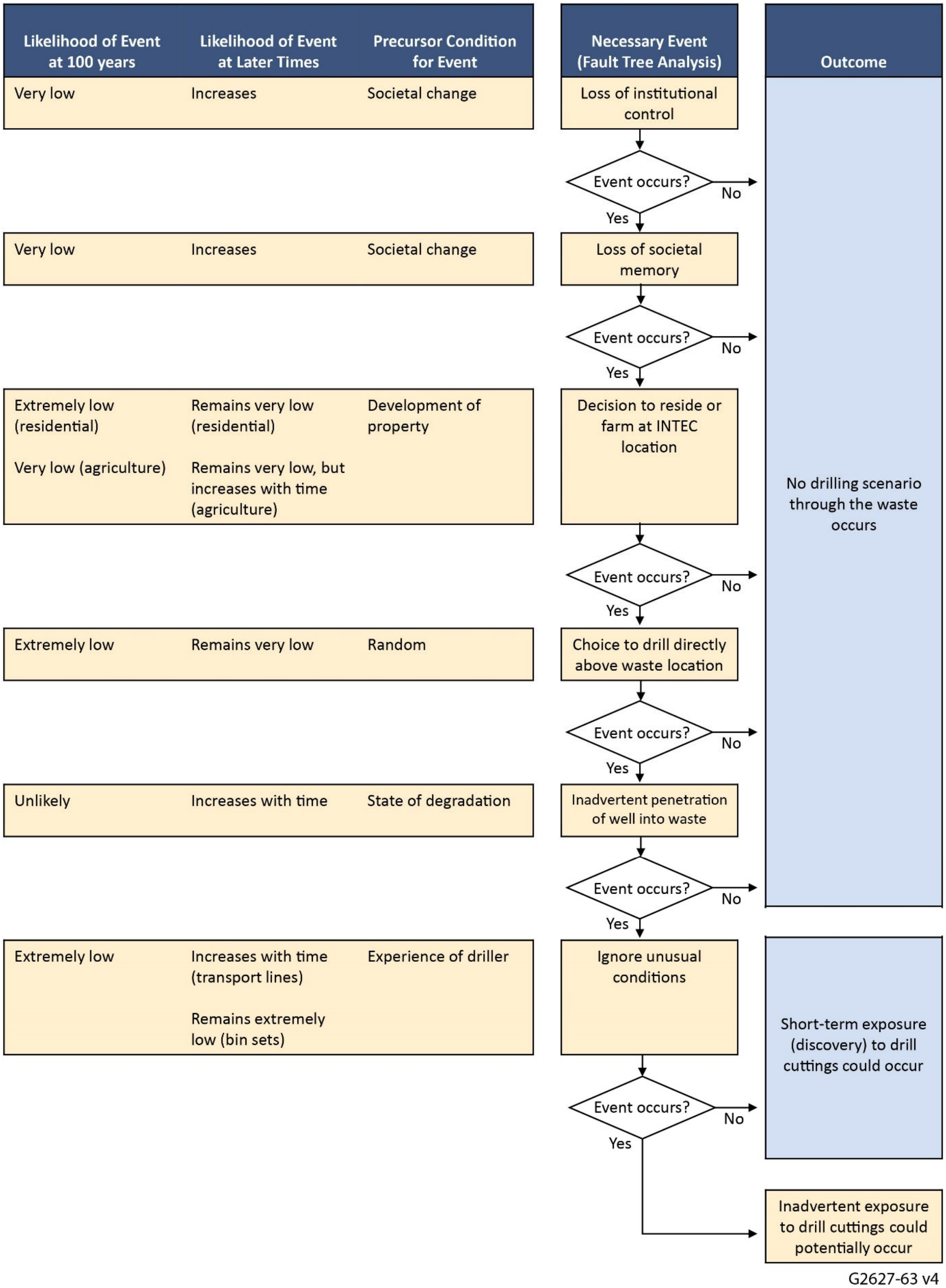

Figure 7-13. Illustration of likelihood and events necessary for drill cuttings from waste to be brought to the surface in an inadvertent human intrusion scenario.

The second necessary event in the chain is the loss of societal memory of the existence of the INL Site and of waste disposal activities and tank closure activities in the INTEC area. This would involve both loss of individuals' knowledge of the existence of the INL Site and the loss of relevant records and deed restrictions that would warn future inhabitants of the residual hazards that exist. In Idaho, deep wells must be drilled by licensed professionals, and a permit to drill must be obtained (Idaho Code 42-235 et seq.). The precursor condition for this event is similar to the loss of institutional control: a profound societal change with a loss of memory of the activities at the INL Site and failure of the permitting process

The third necessary event would be a decision to drill a well in the area of INTEC. The likelihood that this would occur is dependent on the motivation for drilling for water (or potentially other resources). The precursor condition is a desire for some sort of property development on the INL Site that needs a source of water, such as a housing project or farm. This event is also conditional based on the loss of institutional control and memory of the INL Site. The lack of significant residential development around the boundary of the existing INL Site, with the thousands of jobs in the area that have existed for more than 60 years, highlights the generally inhospitable conditions and suggests a limited expectation of any significant future residential development when the employment opportunities are gone. However, there is farming in some boundary areas, primarily around the northern half of the INL Site. So there is some possibility that potential farming could extend into the boundaries of the INL Site if controls and memory of the INL Site are assumed to be lost. The potential for farming activities on the INL Site is expected to be low in the near term but could increase over time.

The fourth necessary event would be a decision to specifically drill where the CSSF bins or transport lines are located. In the absence of notable distinguishing features to modify the likelihood across the INL Site, this would reasonably be regarded as a random decision (the presence of the CSSF above grade is likely to be a strong indicator to limit the potential for development). Therefore, assuming a random event, in this case, one could in principle evaluate the probability by comparing the area of the contaminated bins and transport lines at the CSSF with the typical number of wells in areas surrounding the INL Site. This event can also be further refined to distinguish between the likelihood of intrusion into bins, transport lines, and—more specifically—potential residual waste deposits in the transport lines.

An example of the density of wells installed since 1987 in an area near the INL Site is provided in Figure 7-14. This figure involves an area outside the western boundary of the INL Site (see the "Area of Interest" in the figure legend for perspective regarding the location). The red square on the figure represents 2 mi^2 (~1,280 acres or ~5.2E6 m^2), specifically, where the highest density of wells are present. Focusing a square on an area of the map with a higher frequency of wells suggests that for the worst case, there are roughly 35 wells for a 2-mi^2 area, which translates to ~1 well per 37 acres (1.5E5 m^2). Even if it is assumed that future drilling at the CSSF occurs at a frequency consistent with the higher-density areas on the map, the likelihood of drilling in an area directly above a bin is very low (~3 in 1,000), drilling directly above a transport line is lower (~2 in 10,000), and drilling directly above a suspected deposit of waste in a transport line is exceedingly small (~1 in 100,000).[68] It could be argued that more wells may be drilled over time, but even if 10 times as many wells were drilled as are shown inside the 2-mi square in the figure, the likelihood of hitting a suspect waste deposit in a transport line would still be ~1 in 10,000 if farming was occurring with a high number of wells similar to agricultural areas near the INL Site.

68. Estimated probabilities are based on density of wells (6.76E-06 wells/m^2) times the areal footprint of the bin sets (394 m^2), transport lines (23.4 m^2), and potential waste in the transport lines (1.83 m^2), respectively.

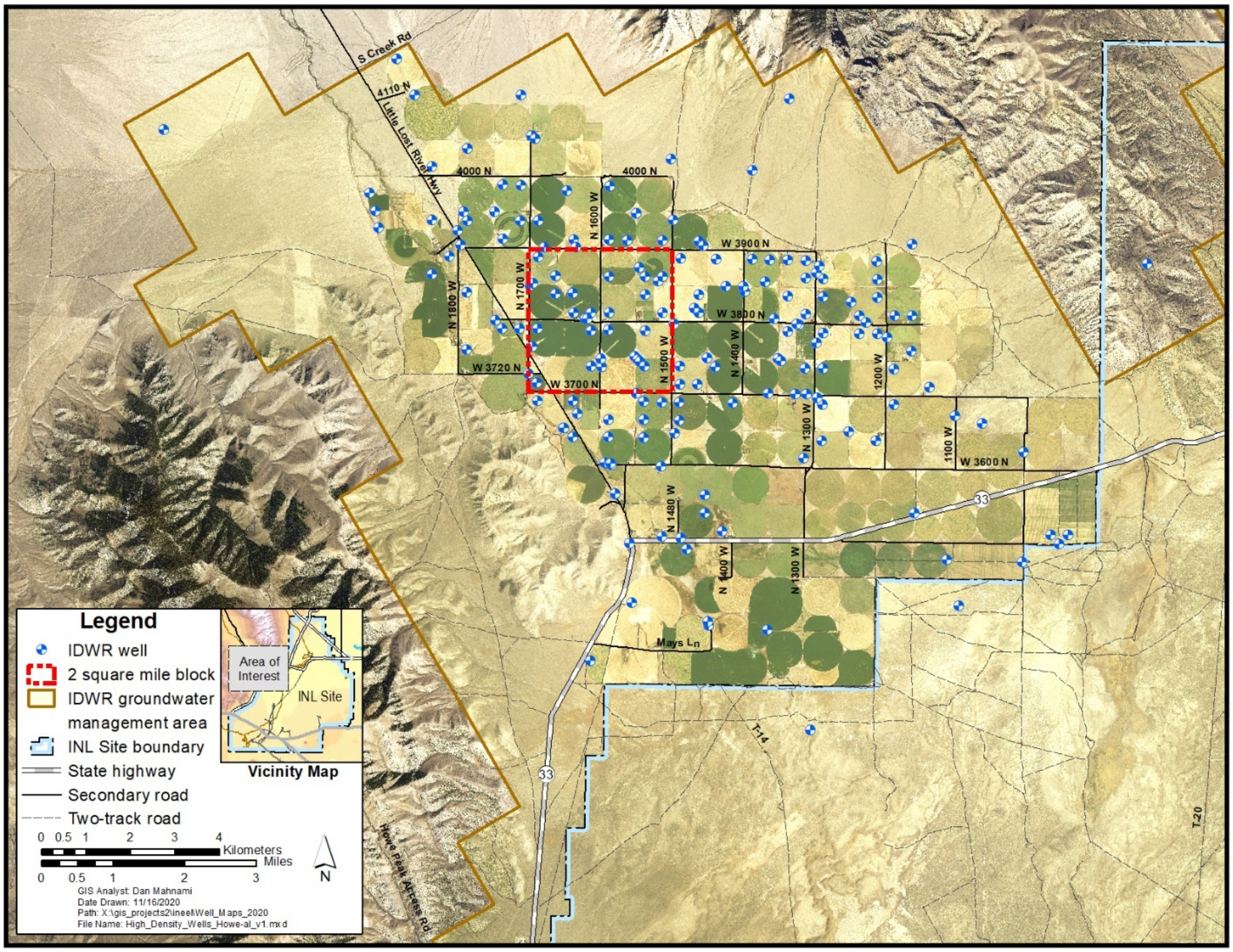

Figure 7-14. Screenshot of wells installed since 1987 in an area near Howe, Idaho, on the western boundary of the Idaho National Laboratory Site (see the "Area of Interest" in the figure legend for perspective regarding the location) (screenshot source: IDWR [2020]).

The fifth necessary event after assuming drilling occurs directly above the waste is the penetration of the drill bit into waste. The conditional likelihood of drilling into the waste at 100 years, assuming all the previous events have occurred, is low for the bins and transport lines, owing to the reinforced concrete, grout, and steel barriers present above and around the contamination. It would generally be expected to take extra effort beyond normal drilling activities, even if it is possible, to drill through the robust barriers. The NRC generally considers reinforced concrete to be an effective barrier to inadvertent drilling for at least 500 years (NRC 2007), and DOE PAs have reflected similar assumptions (Oztunali and Roles 1986). In actuality, it is anticipated that the barriers would potentially be effective for significantly longer time frames, especially if an event is considered "inadvertent." The default time frame of 500 years, which is consistent with NRC assumptions and DOE recommendations regarding drilling into reinforced concrete, was adopted for the CSSF PA/CA (for example, see DOE-STD-5002-2017 and Oztunali and Roles [1986]).

The sixth and final necessary event assumes the driller would not recognize that the drill had encountered unusual conditions as they drill to the depth of the waste. In developing the inadvertent intruder scenarios, the NRC used the term "indistinguishable from soil" as one consideration for the viability of an inadvertent intrusion event. The likelihood of a driller reaching residual waste at the bottom of a vault and bin without recognizing it was not soil is exceedingly small (and would remain small for times well beyond the 1,000-year post-closure period). The precursor condition for this necessary event to occur would be that the driller would have to be inexperienced or inattentive to not recognize the unusual nature of the bins and grout materials or transport line components in the well bore. In Idaho, it is required to obtain a drilling permit and use a licensed driller (Idaho Code 42-235 et seq.). So even if the permitting process missed the land-use restrictions for the INL Site, it is regarded as extremely unlikely that someone would not notice the difference from soil while drilling through roughly 9 m (30 ft) or more of reinforced concrete, grout, and stainless steel bins to the depth of the residual waste. Furthermore, the conditional likelihood of this event would not increase even at very long times when the CSSF materials may be structurally degraded, because the texture and color of the grout and concrete will continue to be easily distinguishable from the surrounding soils for several thousand years. This specific argument is more limited for the transport lines because there is less manmade material above the waste.

Because these events are independent and sequential, one could in principle assign numerical conditional probabilities to each and calculate a joint probability of occurrence of the sequence as the product of the probabilities. Given the speculative and judgmental nature of any assessment of the precursor conditions, such numerical probabilities were not used in the CSSF PA/CA. However, from a qualitative viewpoint, one can say that multiplying six small likelihoods together would give a small joint probability of the entire sequence occurring, regardless of the specific assignment of numerical values. It can therefore be reasonably concluded that the likelihood of all of the events required for drilling into waste residuals at the CSSF is expected to be very small.

Note that consideration of the likelihood of residential-type exposures after a hypothetical drilling event and specific exposure assumptions as considered in the intruder analysis is not addressed in this discussion other than noting the general absence of large population centers developing near the INL Site boundary even with the existence of thousands of jobs at the INL Site for more than 60 years (Stacey 2000).

7.2.4 Sensitivity Analysis for an Inadvertent Intruder Drilling into Transport Lines with Residual Material

As described in Subsection 2.11.1.7, the pneumatic transport system operated with an air velocity high enough to prevent solids from falling out (salting) into the transport line, and generally, the last material sent through each transport line was transport air (EDF-11119). During both closures of the calcining facilities (WCF and NWCF), the systems (calcining, processing, and transfer equipment) were scoured with high-velocity air or nonradioactive material to remove residual waste (EDF-11119).

In some instances during startup operations, nonradioactive material was sent through the calcining and transport systems, and in other instances, operations switched to the next CSSF without shutting down the calcining operations or sending nonradioactive material through the system. The air transport system operated in such a way that deposits or "residual accumulation" of material may have developed in dead space in the transport lines, such as dead legs or solids transport lines no longer in use. Potential deposits or residual accumulation locations are identified in Figure 2-44 and Tables 2-4 and 2-5.

Based on the information provided in Subsection 2.11.1.7, which is in turn based on EDF-11119, radioactive calcine may have accumulated in the transport lines at three potential locations near CSSFs 2, 3, and 4 (see Figure 2-44). Though there are areas where calcine may have accumulated in the transport lines, the probability of drilling into one of these areas is small in comparison to the overall length of lines at the CSSF (see discussion in Subsection 7.2.3). At CSSF 1, a portion of the transport lines (approximately a 6.1- to 9.1-m [20- to 30-ft] section) was removed during HWMA/RCRA closure of WCF because that portion of the lines had deposits from CSSF 2 cold startup. Therefore, no material from calcining operations remains in the CSSF 1 transport lines, and the remaining CSSF 1 transport lines were grouted in place.

This subsection presents a sensitivity analysis based on the assumption that an intruder drills into one of the three locations with potential accumulations of radioactive calcine in the transport lines. The intruder drilling scenario assumptions and parameter values are the same as those presented in previous sections, with the only exception being that the volume of waste intercepted by the drill is assumed to be a completely filled 3-in. transport line. The inventory used for the transport line analysis was based on the residual inventory for each CSSF, as provided in Subsection 2.11.3. The use of the CSSF inventories (except for CSSF 1) for the transport line inadvertent intruder sensitivity analysis ensures the range of residual waste volumes potentially contained in the transport lines is captured. The CSSF 1 inventory was excluded from the transport line inadvertent intruder analysis because the section of line that contained nonradioactive material from CSSF 2 cold startup was removed during the HWMA/RCRA closure and the remaining empty transport lines to CSSF 1 were grouted in place.

7.2.4.1 Inadvertent Intruder Drilling Sensitivity Analysis Results

Figure 7-15 shows dose results for the acute intruder sensitivity analysis between 500 and 10,000 years after closure for the transport lines. Results, which are based on the inventory for each CSSF, are well below the dose limit of 500 mrem in a year. The sensitivity analysis for the acute intruder scenario (drilling into a transport line with radioactive calcine) yielded a peak dose of 1.4E+00 mrem for the 500-year post-closure period (see Table 7-10). These results capture the peak intruder doses because the doses continue to decrease during the 10,000-year post-closure period due to radioactive decay. Details of the intruder drilling sensitivity results are provided in EDF-11455, "Supplemental Inadvertent Intruder Pathway Dose Assessment Calculations for the CSSF 3116 Analysis."

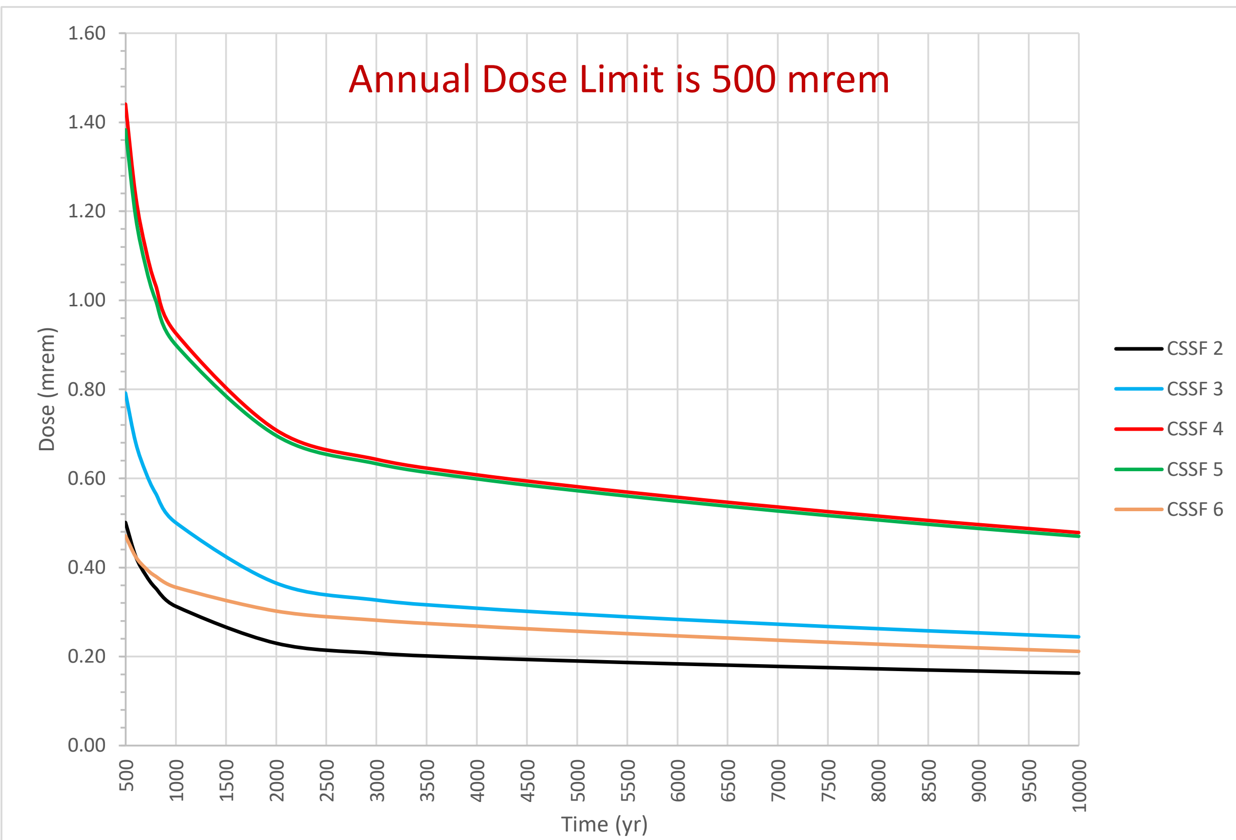

Figure 7-15. Dose results for the acute intruder scenario sensitivity analysis for the transport lines to each Calcined Solids Storage Facility (except Calcined Solids Storage Facility 1).

Table 7-10. Dose results for the acute intruder scenario sensitivity analysis for the transport lines to each Calcined Solids Storage Facility (except Calcined Solids Storage Facility 1).

Time (yr)	Acute Intruder Doses (mrem)[a]				
	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	5.0E-01	7.9E-01	1.4E+00	1.4E+00	4.7E-01
600	4.3E-01	6.8E-01	1.2E+00	1.2E+00	4.2E-01
700	3.8E-01	6.1E-01	1.1E+00	1.1E+00	4.0E-01
800	3.5E-01	5.7E-01	1.0E+00	1.0E+00	3.8E-01
1,000	3.1E-01	5.0E-01	9.3E-01	9.0E-01	3.6E-01
2,000	2.3E-01	3.6E-01	7.1E-01	7.0E-01	3.0E-01
3,000	2.1E-01	3.3E-01	6.4E-01	6.3E-01	2.8E-01
4,000	2.0E-01	3.1E-01	6.1E-01	6.0E-01	2.7E-01
5,000	1.9E-01	3.0E-01	5.8E-01	5.7E-01	2.6E-01
6,000	1.8E-01	2.8E-01	5.6E-01	5.5E-01	2.5E-01
7,000	1.8E-01	2.7E-01	5.4E-01	5.3E-01	2.4E-01
8,000	1.7E-01	2.6E-01	5.1E-01	5.1E-01	2.3E-01
9,000	1.7E-01	2.5E-01	5.0E-01	4.9E-01	2.2E-01
10,000	1.6E-01	2.4E-01	4.8E-01	4.7E-01	2.1E-01

a. The CSSF 1 inventory was excluded from the transport line acute intruder sensitivity analysis because the section of line that was partially filled with CSSF 2 cold startup material was removed during the HWMA/RCRA closure and the remaining empty transport lines to CSSF 1 were grouted in place.

CSSF Calcined Solids Storage Facility
HWMA Hazardous Waste Management Act
RCRA Resource Conservation and Recovery Act

Figure 7-16 shows dose results for the chronic intruder post-drilling agriculture sensitivity analysis between 500 and 10,000 years after closure for the transport lines. Results, which are based on the inventory for each CSSF, are well below the dose limit of 500 mrem in a year. The sensitivity analysis for the chronic intruder post-drilling agriculture scenario (drilling into a transport line filled with radioactive calcine) yielded a peak dose of 1.6E+00 mrem for the 500-year post-closure period (see Table 7-11). These doses capture the peak intruder doses because the doses continue to decrease during the 10,000-year modeling period due to radioactive decay. Details of the intruder drilling sensitivity results are provided in EDF-11455.

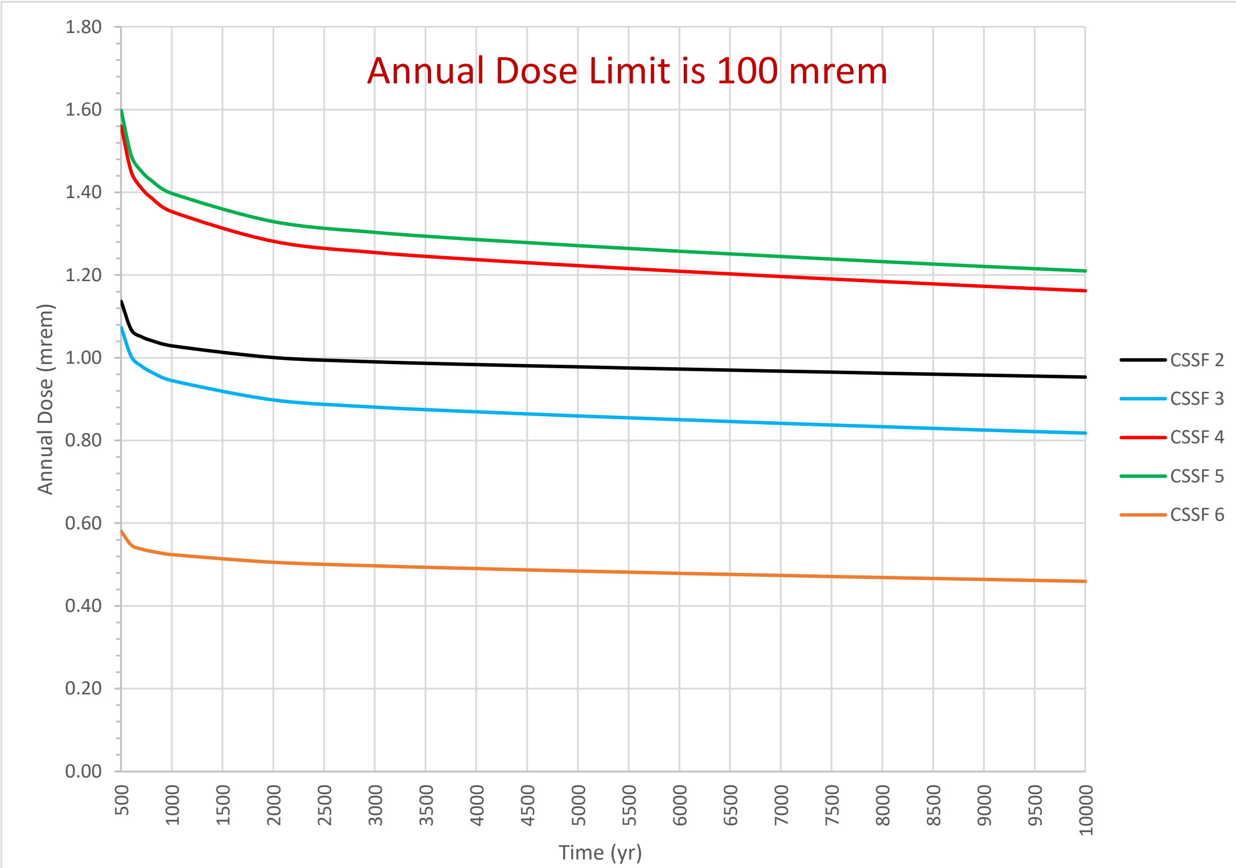

Figure 7-16. Dose results for the chronic intruder scenario sensitivity analysis for the transport lines to each Calcined Solids Storage Facility (except Calcined Solids Storage Facility 1).

Table 7-11. Dose results for the chronic intruder scenario sensitivity analysis for the transport lines to each Calcined Solids Storage Facility (except Calcined Solids Storage Facility 1).

Time (yr)	Chronic Intruder Annual Dose (mrem)[a]				
	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
500	1.1E+00	1.1E+00	1.6E+00	**1.6E+00**	5.8E-01
600	1.1E+00	1.0E+00	1.5E+00	1.5E+00	5.5E-01
700	1.1E+00	9.8E-01	1.4E+00	1.5E+00	5.4E-01
800	1.0E+00	9.7E-01	1.4E+00	1.4E+00	5.3E-01
1,000	1.0E+00	9.4E-01	1.4E+00	1.4E+00	5.2E-01
2,000	1.0E+00	9.0E-01	1.3E+00	1.3E+00	5.1E-01
3,000	9.9E-01	8.8E-01	1.3E+00	1.3E+00	5.0E-01
4,000	9.8E-01	8.7E-01	1.2E+00	1.3E+00	4.9E-01
5,000	9.8E-01	8.6E-01	1.2E+00	1.3E+00	4.8E-01
6,000	9.7E-01	8.5E-01	1.2E+00	1.3E+00	4.8E-01
7,000	9.7E-01	8.4E-01	1.2E+00	1.2E+00	4.7E-01
8,000	9.6E-01	8.3E-01	1.2E+00	1.2E+00	4.7E-01

Table 7-11. (continued).

Time (yr)	Chronic Intruder Annual Dose (mrem)[a]				
	CSSF 2	CSSF 3	CSSF 4	CSSF 5	CSSF 6
9,000	9.6E-01	8.3E-01	1.2E+00	1.2E+00	4.6E-01
10,000	9.5E-01	8.2E-01	1.2E+00	1.2E+00	4.6E-01

a. The CSSF 1 inventory was excluded from the transport line intruder sensitivity analysis because the section of line that was partially with CSSF 2 cold startup material was removed during the HWMA/RCRA closure and the remaining empty transport line to CSSF 1 was grouted in place.

CSSF Calcined Solids Storage Facility
HWMA Hazardous Waste Management Act
RCRA Resource Conservation and Recovery Act

7.2.5 Conclusion for Intruder Analysis Results

Based on dose results for the hypothetical inadvertent human intruder, there is reasonable assurance that the 10 CFR 61.42 inadvertent intruder performance objective of 500 mrem will not be exceeded after CSSF closure. For additional information, there is also reasonable assurance (reasonable expectation) that the DOE M 435.1-1 Chg 3 performance measures of 100 mrem annual ED and 500 mrem total ED, excluding radon in air, for chronic and acute exposure scenarios, respectively, will not be exceeded. In addition, the sensitivity analysis for a transport line with residual accumulation of calcine shows that there is reasonable assurance that the 10 CFR 61.42 inadvertent intruder performance objective of 500 mrem will not be exceeded after CSSF closure.

The results of the intruder analysis should be considered within the context of the actual likelihood of inadvertent intrusion at the CSSF and the pessimistic assumptions that (1) drilling a well at the CSSF will result in inadvertent penetration of a bin or transport line and (2) the residual waste will be unrecognized as different from soil and brought to the surface for subsequent human exposure. The likelihood of drilling into contamination at the CSSF is considered extremely low, as discussed in Subsection 7.2.3. Additionally, the intruder scenarios did not include more material, such as a closure cap, that would reduce concentrations of radionuclides in the soil.

As part of the efforts related to the end state vision for the INL Site and based on planning assumptions for land use within, and adjacent to, the INL Site, key areas of the INL Site, including INTEC, will remain under government control until at least 2095 and portions of the INL Site will remain under government control in perpetuity (INL 2016; DOE-ID 2022d). No new major, private developments (residential or nonresidential) are expected in areas adjacent to the INL Site. Future land use at the CSSF and INTEC for at least a 200-year period (and more likely in perpetuity) is expected to remain essentially the same as the current use: a research facility or controlled access within INL Site boundaries, especially in the major working areas.

7.3 Radiation Protection During Operations

Provisions in 10 CFR 61.43 state the following:

> Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by § 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.

This requirement references 10 CFR 20, "Standards for Protection Against Radiation," which contains radiological protection standards for workers and the public. DOE requirements for occupational radiological protection are provided in 10 CFR 835, "Occupational Radiation Protection," and those for radiological protection of the public and the environment are provided in DOE O 458.1 Chg 4.

Similarly, DOE requirements in Chapter I.D(13) of DOE M 435.1-1 Chg 3, for protection of individuals during operations, states:

> Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, Occupational Radiation Protection, and DOE [Order] 5400.5 [now DOE Order 458.1], Radiation Protection of the Public and the Environment.

The cross-referenced "standards for radiation protection" in 10 CFR 20 that are considered in detail in this Draft CSSF 3116 Basis Document are the dose limits for the public and the workers during disposal operations as set forth in:

- 10 CFR 20, Subpart B, "Radiation Protection Programs," § 20.1101, "Radiation protection programs," 10 CFR 20.1101(d)
- 10 CFR 20, Subpart C, "Occupational Dose Limits," § 20.1201, "Occupational dose limits for adults," 10 CFR 20.1201(a)(1)(i), 10 CFR 20.1201(a)(1)(ii), 10 CFR 20.1201(a)(2)(i), 10 CFR 20.1201(a)(2)(ii), 10 CFR 20.1201(e)
- 10 CFR 20, Subpart C, "Occupational Dose Limits," § 20.1208, "Dose equivalent to an embryo/fetus," 10 CFR 20.1301(a)(1)
- 10 CFR 20, Subpart D, "Radiation Dose Limits for Individual Members of the Public," § 20.1301, "Dose limits for individual members of the public," 10 CFR 20.1301(a)(2) and 10 CFR 20.1301(b).[69]

Consistent with NUREG-1854 (NRC 2007), the following subsections explain that these dose limits correspond to the dose limits in 10 CFR 835 and relevant DOE orders that establish DOE regulatory and contractual requirements for DOE facilities and activities. In addition, the following subsections show that the CSSF closure meets these dose limits and that doses will be maintained ALARA.[70] Table 7-12 provides a crosswalk between the standards set forth in 10 CFR 20 and the applicable DOE requirements.

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69. The introductory "notwithstanding" phrase in NDAA Section 3116 makes it clear that the provisions of NDAA Section 3116(a) are to apply in lieu of other laws that "define classes of radioactive waste." As is evident from the plain language of this introductory "notwithstanding" phrase, NDAA Section 3116(a) pertains to classification and disposal, and radiation protection standards for disposal, of certain waste at certain DOE sites. Thus, the factors for consideration set forth in NDAA Section 3116(a)(1) through NDAA Section 3116(a)(3) are those that pertain to classification and disposal of waste, and the radiation protection standards for disposal. The Joint Explanatory Statement of the Committee of Conference in Conference Report 108-767, accompanying H.R. 4200 (the NDAA), also confirms that NDAA Section 3116(a) concerns classification, disposal, and radiation protection standards associated with disposal and does not concern general environmental laws or laws regulating radioactive waste for purposes other than disposal. Moreover, in the plain language of NDAA Section 3116, Congress directed that the Secretary of Energy consult with the NRC but did not mandate that DOE obtain a license or any other authorization from NRC and did not grant NRC any general regulatory, administrative, or enforcement authority for disposal of the DOE wastes covered by NDAA Section 3116. As such, the "standards for radiation protection" in 10 CFR 20 (as cross-referenced in the performance objective in 10 CFR 61.43), which are relevant in the context of NDAA Section 3116, are the dose limits for radiation protection of the public and workers during disposal operations, and not those that address general licensing, administrative, programmatic, or enforcement matters administered by NRC for NRC licensees. Accordingly, this Draft CSSF 3116 Basis Document addresses in detail the radiation dose limits for the public and workers during disposal operations that are contained in the provisions of 10 CFR 20 referenced above. Although 10 CFR 20.1206 contains limits for planned special exposures for adult workers, there will be no such planned special exposures for closure operations at the CSSF. Therefore, this limit is not discussed further in this Draft CSSF 3116 Basis Document. Likewise, 10 CFR 20.1207 specifies occupational dose limits for minors. However, no minors will be working at CSSF. Therefore, this limit is not discussed further in this Draft CSSF 3116 Basis Document.

70. In addition, 10 CFR 835, like 10 CFR 20 for NRC licensees, includes requirements that do not set dose limits, such as requirements for radiation protection programs, monitoring, entrance controls for radiation areas, posting, records, reporting, or training.

Table 7-12. Crosswalk between applicable 10 CFR 20 standards and U.S. Department of Energy requirements.

10 CFR 20 Standard	DOE Requirement	Draft CSSF 3116 Basis Document Section	Section Title
10 CFR 20.1101(d)	DOE O 458.1 Chg 4	7.3.1	Air Emissions Limit for Individual Member of the Public
10 CFR 20.1201(a)(1)(i)	10 CFR 835.202 (a)(1)	7.3.2	Total Effective Dose Equivalent Limit for Adult Workers
10 CFR 20.1201(a)(1)(ii)	10 CFR 835.202 (a)(2)	7.3.3	Any Individual Organ or Tissue Dose Limit for Adult Workers
10 CFR 20.1201(a)(2)(i)	10 CFR 835.202 (a)(3)	7.3.4	Annual Dose Limit to the Lens of the Eye for Adult Workers
10 CFR 20.1201(a)(2)(ii)	10 CFR 835.202 (a)(4)	7.3.5	Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers
10 CFR 20.1201(e)	10 CFR 851.23(a)(3) and (9)	7.3.6	Occupational Limit on Soluble Uranium Intake
10 CFR 20.1208(a)	10 CFR 835.206(a)	7.3.7	Dose Equivalent to an Embryo/Fetus
10 CFR 20.1301(a)(1)	DOE O 458.1 Chg 4	7.3.8	Total Effective Dose Equivalent Limit for Individual Members of the Public
10 CFR 20.1301(a)(2)	10 CFR 835.602 10 CFR 835.603	7.3.9	Dose Limits for Individual Members of the Public in Unrestricted Areas
10 CFR 20.1301(b)	10 CFR 835.208	7.3.10	Dose Limits for Individual Members of the Public in Controlled Areas

CSSF Calcined Solids Storage Facility
DOE U.S. Department of Energy

7.3.1 Air Emissions Limit for Individual Member of the Public [NRC 10 CFR 20.1101(d); DOE O 458.1 Chg 4]

NRC regulation at 10 CFR 20.1101(d) provides, in relevant part, the following:

> [A] constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established … such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions.

DOE O 458.1 Chg 4 similarly limits the effective dose from air emissions to the public at 10 mrem in a year to comply with the EPA requirement in 40 CFR 61, Subpart H, "National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities," § 61.92, "Standard" (40 CFR 61.92), which has the same limit. The estimated dose per year from airborne emissions to the MEI member of the public located at or beyond the INL Site boundary from all operations at the INL Site ranged from 9.30E-02 to 8.00E-03 mrem from 2007 through 2017 (DOE-ID 2018a). These doses are for all INL Site operations, not only INTEC CSSF closure operations, and are well below the dose limit specified in 10 CFR 20.1101(d) of 10 mrem (0.1 mSv) in a year. The MEI is a hypothetical member of the public who could receive the maximum possible dose from INL Site releases. This person was assumed to live just south of the INL Site boundary. For comparison, the dose from natural background radiation was estimated in 2017 to be 3.83E+02 mrem (3.83E+00mSv) to an individual living on the Snake River Plain (DOE-ID 2018a).

7.3.2 Total Effective Dose Equivalent Limit for Adult Workers [NRC 10 CFR 20.1201(a)(1)(i); DOE 10 CFR 835.202(a)(1)]

NRC regulation at 10 CFR 20.1201(a) concerning occupational dose limits for adults provides, in relevant part, the following:

> (a) … [C]ontrol the occupational dose to individual adults, except for planned special exposures … to the following dose limits.
>
> (1) An annual limit, which is the more limiting of—
>
> (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv).

DOE regulation in 10 CFR 835, Subpart C, "Standards for Internal and External Exposure," § 835.202, "Occupational dose limits for general employees" (10 CFR 835.202), item (a)(1) has the same annual dose limit for the annual occupational dose to general employees. For the occupational dose to adults during CSSF closure, the total ED per year will be controlled using ALARA principles and will be below 5 rem in a year, as described in the ICP "Radiological Control Manual" (PRD-183) and "ICP Radiation Protection Program" (PLN-260). Occupational doses to workers have been well below annual limits specified in 10 CFR 20.1201(a)(1)(i) for CSSF work activities. Total ED to workers from the CSSF closure is expected to remain well below the DOE and NRC limit.

7.3.3 Any Individual Organ or Tissue Dose Limit for Adult Workers [NRC 10 CFR 20.1201(a)(1)(ii); DOE 10 CFR 835.202(a)(2)]

NRC regulation at 10 CFR 20.1201(a) concerning occupational dose limits for adults provides, in relevant part, the following:

> (a) … [C]ontrol the occupational dose to individual adults, except for planned special exposures … to the following dose limits.
>
> (1) An annual limit, which is the more limiting of—

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

The dose limit specified in 10 CFR 20.1201(a)(1)(ii) is similar to the dose limit specified in 10 CFR 835.202(a)(2). For the occupational dose to adults during CSSF closure, the sum of the equivalent dose to the whole body and the committed equivalent dose to any individual organ or tissue (other than the lens of the eye) will be controlled using ALARA principles and will be below a maximum of 50 rem in a year, as described in PRD-183 and PLN-260.

7.3.4 Annual Dose Limit to the Lens of the Eye for Adult Workers [NRC 10 CFR 20.1201(a)(2)(i); DOE 10 CFR 835.202(a)(3)]

NRC regulation at 10 CFR 20.1201(a) concerning occupational dose limits for adults provides, in relevant part, the following:

> (a) … [C]ontrol the occupational dose to individual adults, except for planned special exposures … to the following dose limits.
>
> (2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, which are:
>
> (i) A lens dose equivalent of 15 rems (0.15 Sv).

The dose limit specified in 10 CFR 20.1201(a)(2)(i) is the same as that specified in the DOE regulation at 10 CFR 835.202(a)(3). For the occupational dose to adults during CSSF closure, the annual dose limit to the lens of the eye will be controlled using ALARA principles and will be below 15 rem in a year, as described in PRD-183 and PLN-260.

7.3.5 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers [NRC 10 CFR 20.1201(a)(2)(ii); DOE 10 CFR 835.202(a)(4)]

NRC regulation at 10 CFR 20.1201(a) concerning occupational dose limits for adults provides, in relevant part, the following:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures … to the following dose limits.
>
> (2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are:
>
> (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

This NRC dose limit specified in 10 CFR 20.1201(a)(2)(ii) is the same as the DOE dose limit specified at 10 CFR 835.202(a)(4). For the occupational dose to adults during INTEC CSSF closure that involves limited, hands-on activity, the annual dose limit to the skin of the whole body or to the skin of any extremity will be controlled using ALARA principles and will be below a shallow-dose equivalent of 50 rem in a year, as described in PRD-183 and PLN-260.

7.3.6 Occupational Limit on Soluble Uranium Intake [NRC 10 CFR 20.1201(e)]

NRC regulation at 10 CFR 20.1201(e) concerning occupational dose limits for adults, provides in relevant part:

> (e) In addition to the annual dose limits, …limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity[.]

In addition to the adult annual dose limits during CSSF closure, the soluble uranium intake by an individual is controlled to less than 10 mg in a week. DOE regulations in 10 CFR 851.23(a)(3) and (9) state that contractors must comply with the Occupational Safety and Health Administration (OSHA) safety and health standards in 29 CFR 1910, "Occupational Safety and Health Standards," or the American Conference of Governmental Industrial Hygienists' *2022 Threshold Limit Values (TLVs) and Biological Exposure Indices (BEIs)* (ACGIH 2021) when the threshold limit values are lower (more protective) than permissible exposure limits established in 29 CFR 1910. The American Conference of Governmental Industrial Hygienists' threshold limit value for soluble uranium is 0.2 mg/m^3 (the same as noted in 10 CFR 20, Appendix B, Footnote 3). The OSHA permissible exposure limit is 0.05 mg/m^3. DOE-STD-1136-2017, "Good Practices for Occupational Radiological Protection in Uranium Facilities," suggests that the OSHA limit should be applied at 0.05 mg/m^3 for soluble uranium and at 0.25 mg/m^3 for insoluble uranium. The soluble uranium OSHA permissible exposure limit, which equates to a soluble uranium intake of 2.4 mg in a week, is the more restrictive of the two. The soluble uranium intake, if any, during CSSF closure will be controlled to 2.4 mg in a week, which is below the NRC limit in 10 CFR 20.1201(e).

7.3.7 Dose Equivalent to an Embryo/Fetus [NRC 10 CFR 20.1208(a); DOE 10 CFR 835.206(a)]

NRC regulation at 10 CFR 20.1208(a) concerning the dose equivalent to an embryo/fetus provides, in relevant part, the following:

> (a) … [E]nsure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv).

DOE regulation at 10 CFR 835, Subpart C, § 835.206, "Limits for the embryo/fetus" (10 CFR 835.206), Item (a) has the same dose limit. For the embryo/fetus occupational dose during CSSF closure, doses will be controlled so the dose equivalent to the embryo/fetus during the entire pregnancy for a declared pregnant worker will not exceed 0.5 rem. Furthermore, after pregnancy declaration, DOE provides a mutually agreeable assignment option of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. In addition, personnel dosimetry is provided and used to carefully track exposure, as controlled by PRD-183 and PLN-260.

7.3.8 Total Effective Dose Equivalent Limit for Individual Members of the Public [NRC 10 CFR 20.1301(a)(1); DOE O 458.1 Chg 4]

NRC regulation at 10 CFR 20.1301(a) concerning dose limits for individual members of the public provides, in relevant part, the following:

> (a) … [C]onduct operations so that—
>
> (1) The total effective dose equivalent to individual members of the public … does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released … from voluntary participation in medical research programs, and from the … disposal of radioactive material into sanitary sewerage[.]

Provisions in DOE O 458.1 Chg 4 similarly limit public doses to less than 100 mrem in a year. However, the DOE application of the limit is more restrictive in that it requires DOE to make a reasonable effort to ensure multiple sources (e.g., DOE sources and NRC-regulated sources) do not combine to cause the limit to be exceeded. For individual members of the public during CSSF closure, the total ED limit to an individual member of the public will be controlled to less than 25 mrem in a year per Section 4 e (1)(c) of DOE O 458.1 Chg 4.

7.3.9 Dose Limits for Individual Members of the Public in Unrestricted Areas [NRC 10 CFR 20.1301(a)(2); DOE 10 CFR 835.602 and 603]

NRC regulation at 10 CFR 20.1301(a) concerning dose limits for individual members of the public provides, in relevant part, the following:

> (a) … [C]onduct operations so that—
>
> (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released … does not exceed 0.002 rem (0.02 millisievert) in any one hour.

DOE regulation at 10 CFR 835, Subpart G, "Posting and Labeling," § 835.602, "Controlled areas" (10 CFR 835.602) establishes the expectation that total ED in controlled areas will be less than 0.1 rem/year. For individual members of the public during CSSF closure, operations will be conducted such that the dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material, will be less than 0.00005 rem/hour above background. PRD-183 also restricts the total ED in controlled areas to less than 0.1 rem/year. To ensure these dose limits are met, the following measures have been instituted within controlled areas:

Per 10 CFR 835, Subpart G, § 835.603, "Radiological areas and radioactive material areas" (10 CFR 835.603), radioactive materials areas have been established for radioactive material accumulation possibly resulting in a radiation dose of greater than or equal to 100 mrem in a year. In addition, INTEC has established radiological buffer areas (RBAs) around posted radiological areas. Standard INTEC practice is to assume a 2,000-hour/year continuous occupancy at the outer boundary of these areas; therefore, the dose rate at an RBA boundary is 0.05 mrem/hour (100 mrem/2,000 hour = 0.05 mrem/hour or 0.00005 rem/hour). Because the controlled area encompasses an RBA, it is ensured that the dose in the controlled area (but outside of radioactive material areas and RBA) will be less than 0.1 rem/year in accordance with PRD-183. Therefore, INTEC implementation of the provisions at 10 CFR 835.602 and 835.603 provides limits protective of the dose limit specified in 10 CFR 20.1301(a)(2). Training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem/year (PLN-260; PRD-183).

7.3.10 Dose Limits for Individual Members of the Public in Controlled Areas [NRC 10 CFR 20.1301(b); DOE 10 CFR 835.208]

NRC regulation at 10 CFR 20.1301(b) concerning dose limits for individual members of the public provides, in relevant part, the following:

> (b) If … members of the public [are permitted] to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

DOE regulation at 10 CFR 835, Subpart C, § 835.208, "Limits for members of the public entering a controlled area" (10 CFR 835.208) has the same dose limit. The total ED limit for an individual member of the public granted access to controlled areas during CSSF closure will be controlled to 0.1 rem/year. Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem/year (PLN-260; PRD-183).

7.3.11 As Low as Reasonably Achievable (NRC 10 CFR 20.1003; DOE 10 CFR 835.2)

NRC regulation at 10 CFR 20, Subpart A, "General Provisions," § 20.1003, "Definitions" (10 CFR 20.1003) defines ALARA, in relevant part, as follows:

> ALARA … means making every reasonable effort to maintain exposures to radiation as far below the dose limits … as is practical consistent with the purpose for which the … activity is undertaken …[.]

Measures that provide reasonable assurance that the CSSF closure will comply with the applicable dose limits and with the ALARA provisions include the documented radiation protection program, the CSSF Safety Analysis Report (SAR-105); design, regulatory, and contractual enforcement mechanisms; and access controls, training, and dosimetry. These measures are discussed in the following subsections.

7.3.11.1 CSSF Operations Contractor Radiation Protection Program

DOE regulates occupational radiation exposure at its facilities through 10 CFR 835, which establishes exposure limits and other requirements to ensure DOE facilities are operated in a manner such that occupational exposure to workers is maintained within acceptable limits and as far below these limits as is reasonably achievable. Requirements in 10 CFR 835, if violated, provide a basis for the assessment of civil penalties under Section 234A of the Atomic Energy Act (42 USC 2011 et seq.), as amended.

Pursuant to 10 CFR 835, INTEC activities, including CSSF closure operations, must be conducted in compliance with the "ICP Radiation Protection Program" (PLN-260), as approved by DOE. The ICP Radiation Protection Program is responsible for addressing each requirement in 10 CFR 835 and is generally based on functional elements contained in DOE G 441.1-C Chg 1, "Radiation Protection Programs Guide for Use with 10 CFR 835." Functional elements, described in PLN-260, include organization and administration, ALARA, external dosimetry, internal dosimetry, area monitoring and control, radiological controls in the workplace, emergency exposure situations, nuclear accident dosimetry, records, reports to individuals, and radiation safety training.

The 10 CFR 835 requirements, as contained in the radiation protection program, are incorporated in the standards/requirement identification document system. The system links the requirements of 10 CFR 835 to the company- and lower-level implementing policies and procedures that control radiological work activities conducted across the INL Site. These procedures control the planning of radiological work; the use of radiation monitoring devices by employees; the bioassay program; the air monitoring program; the contamination control program; the ALARA program; the training of general employees, radiological workers, and health physics professionals and technicians; and the other aspects of an occupational radiation protection program as required by 10 CFR 835.

7.3.11.2 Documented Safety Analysis

The CSSF is a Hazard Category 2 nuclear facility and is currently undergoing waste retrieval operations in preparation for closure. An existing approved safety basis (SAR-105) covers operational activities at the CSSF, including waste storage, monitoring, and retrieval. Operating procedures and work control documents are screened for compliance with the safety basis and technical safety requirements. This process ensures that (1) all credible hazards and accidents are analyzed and (2) controls are put in place to prevent or mitigate the hazards and accidents.

Post-retrieval closure activities (e.g., grouting, abovegrade demolition and decommissioning, and construction of an engineered barrier over the facility) are not specifically addressed in the current safety basis. As the CSSF transitions to closure, these activities will be evaluated through the process hazard analysis and unreviewed safety question processes. These processes will determine how the safety basis should be amended to support closure activities. It is expected that the safety basis hazards and controls will be reduced as closure activities progress and that no safety basis controls will be required at the completion of closure.

7.3.11.3 Radiological Design for Protection of Occupational Workers and the Public

The CSSF and facility modifications have been designed to meet the requirements of 10 CFR 835, Subpart K, "Design and Control." The ICP "ALARA Program and Implementation" procedure (MCP-91) provides instruction necessary to ensure compliance with the requirements of 10 CFR 835. The procedure refers to 10 CFR 835 and PRD-183 for implementing the requirements, while the ALARA design review process includes applicable DOE orders, DOE standards, DOE handbooks, national consensus standards, INTEC manuals, INTEC engineering standards, INTEC engineering guides, and INL Site operating experience to meet the 10 CFR 835 specific requirements. Compliance with MCP-91 also meets additional requirements to ensure the design provides for protection of the workers and the environment.

10 CFR 835 covers the full spectrum of radiological design requirements, not just radiation exposure limits. The following are the specific areas addressed in the regulation: radiation exposure limits, facility and equipment layout, area radiation levels, radiation shielding, internal radiation exposure, radiological monitoring, confinement, and ventilation.

Facility design at the CSSF incorporates radiation zoning criteria to ensure exposure limits are met by providing adequate radiation shielding. Areas in which nonradiological workers are present are assumed to have continuous occupancy (2,000 hours/year) and are designed to a dose rate less than 0.05 mrem/hour to ensure the annual dose is less than 100 mrem. Other zoning criteria are established to ensure radiological worker doses are ALARA and less than an average of 0.5 mrem/hour to meet the design requirements in 10 CFR 835, Subpart K, § 835.1002, "Facility design and modifications" (10 CFR 835.1002). The facility design also is required to provide necessary radiological monitoring or sampling for airborne and surface contamination to ensure the engineered controls are performing their function and, in the event of a failure or upset condition, workers are warned and exposures avoided.

Radiological protection personnel ensure applicable requirements of the standard are addressed and presented in design summary documentation. The incorporation of radiological design criteria in the engineering standard ensures that requirements of 10 CFR 835 are met and that the design provides for the radiological safety of the workers and environment.

7.3.11.4 Regulatory and Contractual Enforcement

Any violation of 10 CFR 835 requirements is subject to civil penalties pursuant to Section 234A of the Atomic Energy Act, as amended (42 USC 2011 et seq.), as implemented by DOE regulations in 10 CFR 820, "Procedural Rules for DOE Nuclear Activities." In addition, the requirements in 10 CFR 835 and all applicable DOE orders are incorporated into all contracts with DOE contractors. DOE enforces these contractual requirements through contract enforcement measures, including the reduction of contract fees (48 CFR 970).

7.3.11.5 Access Controls, Training, Dosimetry, and Monitoring

Training or an escort is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits (PLN-260; PRD-183), use of dosimetry is required if a member of the public is expected to enter a controlled area and exceed 0.05 rem/year.

In addition, worker radiation exposure monitoring is performed for all workers expected to receive 100 mrem/year from internal and external sources of radiation to provide assurance that no worker exceeds radiation exposure limits and all radiation doses are maintained as far below the limits as is reasonably achievable (PLN-260; PRD-183).

7.3.11.6 Occupational Radiation Exposure History for the CSSF Operations Contractor

The effectiveness of the radiation protection program, including the effectiveness of oversight programs, is demonstrated by the occupational radiation exposure results. INTEC quarterly radiological performance reports consistently demonstrate that the program is effective. For the period January 1, 2022, to December 31, 2022, the average whole body dose for an exposed worker was less than 100 mrem/year,[71] compared to the DOE maximum Administrative Control Limit of 2,000 mrem/year and the 10 CFR 835 limit of 5,000 mrem/year. INTEC strives to maintain doses well below the 2,000-mrem DOE annual administrative limit by using 700 mrem as the administrative control limit for each employee. This administrative limit is only increased following an analysis of the worker dose and subsequent actions to keep the worker dose ALARA.

7.3.12 Conclusion for Radiation Protection During Operations

Based on the previous discussion, operations at the CSSF are conducted in compliance with the standards for radiation protection set out in 10 CFR 20 and 10 CFR 835. Every reasonable effort continues to be made at the CSSF to maintain radiation exposures ALARA.

Measures that provide reasonable assurance that the CSSF closure will comply with the applicable dose limits and with the ALARA provisions include the documented radiation protection program (PLN-260; PRD-183); the CSSF Safety Analysis Report (SAR-105); design, regulatory, and contractual enforcement mechanisms; and access controls, training, and dosimetry.

7.4 Stability of the Disposal Site After Closure

10 CFR 61.44 states:

> The disposal facility must be sited, designed, used, operated and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

As discussed previously, all the bins in the CSSF have yet to be emptied of calcine to the maximum extent practicable in preparation for stabilization and closure of the CSSF. Once the calcine has been transferred from the CSSF for disposal, as applicable, the CSSF will be closed. Stabilization of residual waste in situ in the CSSF will meet the performance objectives of 10 CFR 61.44 for long-term stability.

7.4.1 Siting

A comprehensive review of site geology, seismology, hydrology, demography, meteorology, and environmental setting are presented in Section 2 of this Draft CSSF 3116 Basis Document and its references. This subsection briefly summarizes the comprehensive review.

The INL Site is located on the north-central part of the ESRP in southeastern Idaho. Included in its 2,305-km^2 (890-mi^2) area are portions of five Idaho counties (Bingham, Bonneville, Butte, Clark, and Jefferson). The nearest INL Site boundaries are 51 km (32 mi) west of Idaho Falls, 37 km (23 mi) northwest of Blackfoot, 71 km (44 mi) northwest of Pocatello, and 11 km (7 mi) east of Arco, Idaho.

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71. Source: The INL Health Physics Dosimetry Laboratory. The INL Health Physics Dosimetry Laboratory (managed by BEA) maintains exposure data for the ICP. Exposure data are not publicly available. For questions regarding exposure data, contact the INL Health Physics Dosimetry Laboratory.

The INL Site is approximately equidistant from the three larger metropolitan areas of Salt Lake City, Utah, 339 km (211 mi); Boise, Idaho, 413 km (257 mi); and Butte, Montana, 344 km (214 mi).

The ESRP is commonly divided into two regions: the western region of the plain is a northwest-trending depositional basin, and the eastern region is a northeastern-trending volcanic plain. The ESRP is the product of plains-style volcanism due to low-viscosity magma that flowed laterally from vents. Overlapping flows from one or more vents produced shield formations across the plain, followed by minor fissure-fed flows into low areas between shields. Underlying the western region of the ESRP is a sequence of Tertiary and Quaternary volcanic rocks and sedimentary interbeds that extend beyond the depth of 3,048 m (10,000 ft). The uppermost part of the volcanic rocks consists mainly of basalt flows, with rhyolitic ash-flow tuffs composing the lowermost part.

Under DOE O 420.1C Chg 3 requirements, the INL Site has a seismic network for monitoring earthquake activity on and around the ESRP to support DOE operations. Monitoring indicates that the ESRP is relatively seismically inactive when compared to surrounding Basin and Range regions. The seismically active Intermountain seismic belt and Centennial Tectonic seismic belts, which surround the ESRP, extend more than 1,287 km (800 mi) from southern Arizona through eastern Idaho to western Montana. This distribution of epicenters indicates that the Snake River Plain is devoid of earthquakes relative to the active areas surrounding it, and recent ongoing activity is likely associated with volcanic processes.

The INL Site is in a region of Pleistocene and Holocene volcanic activity typically characterized by nonviolent, effusive basalt lava flows. Explosive rhyolite volcanism occurred beneath the INL Site 4 million to 7 million years ago, forming calderas now buried beneath basalt lava flows. In the region immediately surrounding the INL Site, the youngest lava flow erupted about 4,100 years ago from Hell's Half Acre lava flow southeast of the INL Site. Within INL Site boundaries, the most recent lava flow—the Cerro Grande flow—occurred 13,000 years ago, near the southern boundary. Renewed explosive rhyolite volcanism at the INL Site is very unlikely.

Geological and geochronological data indicate an eastward progression of silicic volcanism. The mantle plume or hotspot assumed responsible for the volcanism now lies beneath Yellowstone National Park. Past patterns of volcanism suggest that future volcanism at the INL Site within the next 1,000 to 10,000 years is very improbable, and the two most likely sources of future basalt flows on the INL Site are the Arco-Big Southern Butte and Lava Ridge-Hell's Half Acre rift zones.

Rain and snowmelt periodically infiltrate into the gravelly alluvium in and around the CSSF and INTEC. Even though average annual precipitation (22 cm [8.66 in.]/year) is much less than the pan evaporation rate (109 cm [42.9 in.]/year), water from snowmelt or heavy rains can and does infiltrate into the ground to depths where it cannot evaporate. This water then continues to move downward until it recharges perched water and the SRPA. The OU 3-14 Remedial Investigation/Baseline Risk Assessment (Cahn et al. 2006) concluded that the recharge rate inside the INTEC security fence may be approximately 18 cm (7 in.)/year, which constitutes 85% of the average annual precipitation (22 cm [8.66 in.]/year).

Surface water sources at INTEC and near the CSSF include (1) the Big Lost River (when flowing), (2) ponded rain and snowmelt, (3) the CERCLA storm water evaporation pond (construction completed October 2003), (4) the ICDF evaporation ponds (operations began September 2003), (5) the INTEC Sewage Treatment Plant, and (6) the former INTEC percolation ponds. The CERCLA storm water evaporation pond, the ICDF evaporation ponds, and INTEC Sewage Treatment Plant are lined ponds managed by their respective programs and, as such, are not considered a likely source of infiltration. The former INTEC percolation ponds also are not considered a likely source of infiltration. They were relocated 3.2 km (2 mi) west of INTEC.

The Big Lost River is the major surface water feature on the INL Site. At its closest point, the channel of the Big Lost River lies within 30 m (100 ft) of the northwest corner of INTEC. The Big Lost River is an intermittent stream that flows north through the INL Site to its terminus at the Big Lost River sinks, where the water either infiltrates into the ground or evaporates. The stretch of the Big Lost River on the INL Site is ephemeral with no recreational or consumptive uses of the water (e.g., irrigation, manufacturing, or drinking).

When it is flowing, the Big Lost River constitutes a source of recharge to perched water and the SRPA. However, this recharge is limited to the immediate vicinity of the Big Lost River and is not a significant source of recharge near the CSSF. When the Big Lost River is flowing past INTEC, only one INTEC monitoring well—Well BLR-CH, which is the monitoring well located closest to the river and 152.4 m (500 ft) from the river channel—has consistently shown a significant water-level response to the river flow events. INTEC flood inundation maps, with various scenarios of flow infiltration and Lincoln Boulevard culvert flow, indicate the north–northwest end of INTEC to be more susceptible to flooding and that the CSSF is outside the 100-year floodplain boundary, based on the *Big Lost River Flood Hazard Study Idaho National Laboratory, Idaho* (Ostenaa and O'Connell 2005).

The Ostenna and O'Connell (2005) study represented the approximate 100-year historical discharge records of the Big Lost River and augmented the data with a paleoflow analysis; the study modeled precipitation-derived flows onto the INL Site and the potential to reach proposed facility locations. The 100-year flood peak flow is estimated to be 87 cms (3,072 cfs), with the 1,000-year flood peak flow estimated to be 131 cms (4,626 cfs) at the INL diversion dam. Ostenaa and O'Connell (2005) projected the river's flooding extent and water depth at the CSSF vicinity to have a water depth up to 0.5 m (1.64 ft) from a 40-hour flow of 150 cms (5,295 cfs). With an average riverbank elevation northwest of INTEC of 1,467 m (4,912 ft), the 500- and 2,000-year flood mean water elevation for the Big Lost River is calculated to be 1,497.54 m (4,913.18 ft) and 1,497.67 m (4,913.62 ft), respectively. The southeast corner of the NWCF, directly adjacent to the CSSF vicinity, is modeled to be dry through a 500-year flood and have a 2,000-year flood mean water elevation of 1,496.98 m (4,911.35 ft).

A 2003 PA for closure of the TFF, located near the CSSF, evaluated the impact of a Big Lost River flood on the TFF (DOE-ID 2003b). The flood bounding scenario was an extreme precipitation event within the drainage basin and above the Mackay Dam, causing the overtopping failure of the dam. One to two meters (3.3 to 6.5 ft) of water could cover INTEC, but this would occur only for a short duration. The evaluation concluded that the impact of this possible flooding condition on INTEC would be minimal.

7.4.2 Design

The CSSF design is similar for all the bin sets. The design includes vertical, stainless-steel bins inside a reinforced-concrete vault. The vault for CSSF 1 is rectangular and placed wholly underground. The vaults for CSSFs 2 and 3 are cylindrical, are located partially underground, and have had gravel berms placed around them. The CSSF 4, 5, and 6 storage vaults are cylindrical and located partially underground. In addition to housing the bins, each CSSF contains a cyclone cell (for calcine distribution) and an instrument room with CSSF monitoring equipment. Figure 2-27 provides cutaway views of the seven CSSF vaults.

Construction is different for each CSSF, with bin heights ranging from 6.1 to 20.7 m (20 to 68 ft), diameters ranging from 3.6 to 4.1 m (142 to 162 in.), and construction materials of Type 304, 304L, or 405 stainless-steel plate. The reinforced-concrete storage vaults also are different for each CSSF. The CSSF 1 storage vault has inside horizontal dimensions of 7.8 × 7.8 m (25.5 × 25.5 ft) and a height of 12.6 m (41.3 ft). The CSSF 2, 3, 4, and 6 storage vaults have an outer diameter that ranges from 15.2 to 18.6 m (50 to 61 ft) and a height of 18.8 to 34 m (61.8 to 112 ft). The concrete vault wall thickness ranges from 0.61 to 1.3 m (2 to 4 ft), and the floor thickness ranges from 0.61 to 2 m (2 to 6.5 ft).

The CSSF closure configuration is anticipated to include the concrete storage vaults, stainless-steel bins, and void spaces (storage vaults, bins, and piping) filled with grout, which will serve to provide long-term structural stability, limit the amount of water infiltration into the bins to mitigate contaminant migration, and provide a barrier against intrusion by burrowing animals, roots, or humans (see details in Subsection 2.11.4.3).

7.4.3 Use and Operation

The CSSF is used to safely store calcine, as described in Subsection 2.11.1. Calcine includes both radioactive constituents and constituents that are hazardous under HWMA/RCRA. The CSSF is operated as a permitted storage unit and regulated under authority of the Idaho HWMA.

7.4.4 Closure

As described in Subsection 2.11.4, closure of the CSSF will include removal of calcine from the bins using a pneumatic retrieval system. Equipment (such as the access risers) that cannot be removed after retrieval operations are complete will be stabilized. As each CSSF component is cleaned, and it is verified that performance objectives and measures have been met, any remaining residual waste will be stabilized by filling each of the applicable CSSF components and void spaces with grout. Grout is the most commonly used material for stabilizing and solidifying radioactive waste (DOE-ID 2022a), and it is expected that long-term stability will be achieved by grouting remaining residual waste, equipment, structures, and void spaces.

Long-term stability of the disposal site after closure means the waste maintains structural integrity under the expected disposal conditions. As such, the long-term stability of the closed facility is an important element of meeting the performance objectives. Stability prevents subsidence, water infiltration, and radionuclide release due to disintegration of the waste form and/or containment, and it minimizes the likelihood of intrusion into the waste. Removal of the waste, including HRRs to the maximum extent practical, minimizes the waste that will be stabilized at closure of the CSSF. The primary barriers relied upon at the CSSF to provide structural stability and reduce migration of contamination from residual calcine are grouted waste in the bins, stainless-steel bins, reinforced-concrete vaults, and grouted void spaces (equipment and structures, such as vaults). Similar to the CSSF bins, the primary barriers for potential residual waste in the transport lines are the stainless-steel transport lines, containment pipe, and reinforced-concrete shielding.

Grouted residual waste, bins, storage vaults, piping, and other remaining equipment provide a long-term, stable waste form. For guidance, DOE considered the *Standard Format and Content of a License Application for a Low-Level Radioactive Waste Disposal Facility* (NRC 1991), which notes that the concept of site stability is focused on reducing the contact of water with the waste and providing assurance that active maintenance will not be needed following closure. The long-term stability of the facility is evaluated in the CSSF PA/CA (DOE-ID 2022a), which provides a degradation analysis of the grouted bins, vaults, and piping and shows that the grout will likely remain intact for at least 2,000 years and likely much longer. The corrosion time for each CSSF will vary because the stainless-steel type and thickness and the corrosion rates will evolve over time as the grout fails; however, the mean geometric corrosion time to hydrological failure for CSSF 1 (worst case) is expected to occur after 141,000 years. As shown above, site conditions do not present hazards that impact the stability of the closed CSSF. In addition, the methods used to close the CSSF will result in a closed facility that does not require ongoing active maintenance following closure. As such, the performance of the closed CSSF will comply with the performance objective in 10 CFR 61.44.

7.4.5 Conclusion of Stability of the Disposal Site After Closure

Analyses provided in previous subsections demonstrate that disposal of stabilized residuals in the CSSF will meet the performance objective set out in 10 CFR 61.44. Calcine will be removed from the CSSF to the maximum extent practical, and any remaining residual waste, equipment, and structures will be stabilized with grout. These engineered barriers (grouted residual waste, stainless-steel bins and transport lines, reinforced-concrete vaults, and grouted void spaces) provide structural stability and reduce migration of contamination to the environment. This is in addition to natural features at the INL Site, such as an arid environment, low rainfall, depth to groundwater, remote location, and a geologically stable region, that also contribute to the overall safety and long-term site stability.

8. STATE-APPROVED CLOSURE PLANS

Section 8 Purpose

The purpose of this section is to demonstrate that removing the CSSF from service and stabilizing the CSSF bins (including integral equipment), transport lines, and any residual calcine therein, as appropriate, will be performed pursuant to a State-approved closure plan.

Section 8 Contents

This section discusses State of Idaho regulation of the CSSF and shows that removing CSSF from service and stabilizing the CSSF bins (including integral equipment), transport lines, and residual calcine therein will be pursuant to State-approved closure plans consistent with the *Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL*.

Section 8 Key Points

- DOE will conduct a performance-based closure of the CSSF pursuant to State-approved closure plans and, thereby, meet Criterion (3)(A)(ii) of NDAA Section 3116(a).
- DOE will prepare a series of closure plans to support CSSF closure. These closure plans will present the strategy for closure of the CSSF and will be submitted to each regulatory authority for approval, as required, prior to closure.

Section 3116(a) of the NDAA (Public Law 108-375) provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission ..., determines ...
>
> (3)(A)(ii) [will be disposed of] pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section.

DOE will conduct a performance-based closure of the CSSF pursuant to State-approved closure plans. DOE intends for CSSF closure to meet HWMA/RCRA closure performance standards for hazardous constituents. Under HWMA/RCRA closure performance standards, the CSSF may be closed in accordance with landfill standards established by the State of Idaho in a State-approved closure plan if clean closure cannot be achieved. Waste removal and decontamination activities needed to meet performance objectives for radioactive constituents will also result in removal of hazardous constituents in accordance with the HWMA/RCRA clean closure performance standards. DOE will verify the effectiveness of these activities to determine whether closure performance standards can be met. Appendix B provides further details specific to DOE's approach and strategy to ensure HWMA/RCRA closure performance standard will be met at the time of closure.

As discussed in Subsection 2.11.4, the CSSF closure is being performed under both DOE and Idaho DEQ requirements. This is an approach that includes preparing closure plans. Calcine stored in the CSSF is mixed HLW, and as such, the State of Idaho regulates the hazardous constituents and the DOE regulates the radioactive constituents. Because the CSSF stores mixed HLW, CSSF closure must comply with closure requirements for hazardous waste as well as radioactive waste. For hazardous waste, closure must comply with RCRA (42 USC 6901 et seq.) as implemented by the Idaho HWMA

(Idaho Code 39-4401 et seq.). CSSF closure specifically will meet the requirements of IDAPA 58.01.05.008. HWMA/RCRA closure will be integrated with a CERCLA non-time critical removal action in accordance with the FFA/CO (DOE-ID 1991). As such, closure of the CSSF will require development of State-approved HWMA/RCRA closure plans, per the *Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL* (PER-114) and CERCLA non-time critical removal action documentation under the FFA/CO (DOE-ID 1991) (see Appendix B), which will both be made available for public review and comment prior to finalization.

In accordance with these requirements, DOE will prepare a series of closure plans to support CSSF closure. These closure plans will present the strategy for closure of the CSSF and will be submitted to each regulatory authority for approval, as required, prior to closure. A HWMA/RCRA closure plan for the CSSF will be prepared and submitted to the State of Idaho for approval. Closure actions will not be initiated until the Idaho DEQ approves the applicable closure plan. See Appendix B for additional details related to DOE's closure approach and strategy for the CSSF.

9. CONCLUSION

Based on the preceding sections of this Draft CSSF 3116 Basis Document, the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure meet the criteria set forth in NDAA Section 3116(a) (Public Law 108-375).

In accordance with the first criterion in Section 3116(a), the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein do not raise any unique considerations that, notwithstanding the demonstration that all other NDAA Section 3116(a) criteria will have been met, require permanent isolation in a deep geologic repository.

This Draft CSSF 3116 Basis Document also demonstrates that the stabilized CSSF bins (including integral equipment), transport, lines and any residual calcine therein will have had HRRs removed to the maximum extent practical at the time of closure, thereby satisfying the second criterion in Section 3116(a). Removal of HRRs to the maximum extent practical at the CSSF will be accomplished using proven pneumatic waste retrieval technologies. Approximately 99% of the radioactivity attributable to HRRs will be removed from the CSSF. Moreover, further removal of HRRs is not practical and would, among other things, increase the risk to workers, and result in an insignificant reduction in the very low potential doses to a member of the public and the hypothetical human intruder.

Regarding the third criterion in Section 3116(a), the stabilized CSSF residual wastes at closure are anticipated to meet concentration limits for Class C LLW as set out in 10 CFR 61.55. DOE also will consult with the NRC on DOE's disposal plans for CSSF pursuant to the consultation process in NDAA Section 3116(a)(3)(B)(iii) to take full advantage of the consultation process established by NDAA Section 3116(a).

This Draft CSSF 3116 Basis Document demonstrates that the stabilized CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure will meet the 10 CFR 61, Subpart C, performance objectives so as to provide for the protection of the public health and the environment, thus meeting the criteria in Section 3116(a). These performance objectives address protection of the general population from radioactive releases, protection of individuals from inadvertent intrusion on the disposal site, protection of individuals during disposal facility operations, and the stability of the disposal site after closure.

Through use of the PA process, DOE has analyzed the possible methods by which a future member of the public or an inadvertent intruder could be exposed to the CSSF residuals. The following is a summary of the results from the CSSF PA/CA (DOE 2022a):

- The groundwater all-pathways doses for the CSSF receptors were considerably less than the 25-mrem annual performance objective during the 1,000-year post-closure period and beyond that period to peak dose. The groundwater all-pathways annual dose was essentially zero (6.79E-14 mrem) for the 1,000-year period after CSSF closure. Groundwater all-pathways doses for the 1,000- to 10,000-year post-closure period were highest at the CSSF 1 receptor (3.45E-02 mrem) followed by the CSSF 2 receptor (3.02E-02 mrem) (see Table 5-1). The peak annual groundwater all-pathways dose of 1.9E-01 mrem (rounded to two significant digits) was projected to occur 19,500 years after CSSF closure (see Table 5-1). This peak dose is well below the 25-mrem annual dose specified in the performance objective in 10 CFR 61.41.

- The annual ED from the air pathway for an MEI located off the INL Site during the 100-year period of assumed institutional control[72] was 1.35E-04 mrem. After the 100-year institutional control period and during the 1,000-year post-closure period, the maximum dose was 6.66E-06 mrem at the 100-m receptor, which is substantially less than the DOE M 435.1-1 Chg 3 10-mrem per year performance objective.
- The combined groundwater and air all-pathways doses for CSSF receptors were all less than the 25-mrem annual dose performance objective for the 10,000-year post-closure period, with a maximum annual dose of 3.00E-02 mrem at 10,000 years. The peak annual dose of 9.00E-01 mrem occurs at 19,500 years.
- The acute intruder drilling scenario yielded a peak total ED of 7.1E+00 mrem 500 years post-closure for the CSSF bins. The acute intruder drilling scenario yielded a peak total ED of 5.6E-02 mrem 500 years post closure for the CSSF transport lines.
- The chronic intruder post-drilling agriculture scenario yielded a peak annual dose of 3.6E+00 mrem 500 years post closure for the CSSF bins. The chronic intruder post-drilling agriculture scenario yielded a peak annual dose of 6.2E-02 mrem 500 years post closure for the CSSF transport lines.
- The maximum peak radon flux for a CSSF transport line within the 1,000-year post-closure period was estimated to be 4.35E-02 pCi/m^2/second at the surface above the line, assuming the line was partially filled with CSSF 1 inventory. This flux is below the DOE M 435.1-1 Chg 3 performance objective of 20 pCi/m^2/second.

Results from the CSSF PA/CA (DOE 2022a) show that there is reasonable assurance the peak annual all-pathways ED for a future hypothetical member of the public and a total ED for a hypothetical inadvertent intruder will remain below 25 mrem and 500 mrem, respectively, in compliance with the performance objectives in 10 CFR 61.41 and 61.42.

DOE has programs in place to ensure the protection of workers and the public during facility operations. As demonstrated in this Draft CSSF 3116 Basis Document, DOE requirements for occupational radiological protection and those for radiological protection of the public and the environment are comparable to the relevant requirements contained in the NRC performance objective in 10 CFR 61.43.

This Draft CSSF 3116 Basis Document demonstrates that residual wastes at the time of closure meet the performance objective in 10 CFR 61.44 concerning long-term site stability. DOE reviewed site characteristics, including demography, geography, meteorology, climatology, ecology, geology, seismology, and hydrogeology. As demonstrated in this Draft CSSF 3116 Basis Document, site conditions do not present hazards that impact CSSF stability. In addition, the CSSF closure methods will result in a facility closure that does not require ongoing maintenance.

With respect to the third criterion in Section 3116(a), the CSSF will be removed from service (operationally closed) and stabilized pursuant to State-approved closure plans, consistent with the *Partial Permit for HWMA Storage for the Calcined Solids Storage Facility at the INTEC on the INL* (PER-114).

DOE is consulting with the NRC and making this Draft CSSF 3116 Basis Document available to states, Tribal Nations, stakeholders, and the public for comment. After careful consideration of NRC consultation comments and comments received from states, Tribal Nations, stakeholders, and the public, DOE will perform any necessary revisions of analyses and technical documents, and prepare a Final CSSF 3116 Basis Document. Based on the Final CSSF 3116 Basis Document, the Secretary of Energy, in

72. In the CSSF PA/CA (DOE-ID 2022a), the analysis assumed a 100-year institutional control period. Future land use likely will be similar to current uses, with research facilities within INL Site boundaries and agricultural and open land surrounding the INL Site. DOE expects to retain ownership and control of the INL Site until at least 2095 and will continue to manage portions that cannot be released for unrestricted land use beyond 2095 (INL 2016).

consultation with the NRC, may potentially determine in the future whether the CSSF bins (including integral equipment), transport lines, and any residual calcine therein at the time of closure are not HLW and may be disposed of (closed) in place at the INL Site as LLW.

10. REFERENCES

10 CFR 20, 2022, "Standards for Protection Against Radiation," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20, Subpart A, 2022, "General Provisions," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20, Subpart B, 2022, "Radiation Protection Programs," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20, Subpart C, 2022, "Occupational Dose Limits," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20, Subpart D, 2022, "Radiation Dose Limits for Individual Members of the Public," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20, Subpart E, 2022, "Radiological Criteria for License Termination," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20.1003, 2019, "Definitions," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20.1101, 2022, "Radiation protection programs," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20.1201, 2019, "Occupational dose limits for adults," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20.1208, 2019, "Dose equivalent to an embryo/fetus," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 20.1301, 2022, "Dose limits for individual members of the public," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61, Subpart C, 2022, "Performance Objectives," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.40, 2022, "General Requirement," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.41, 2022, "Protection of the general population from releases of radioactivity," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.42, 2022, "Protection of individuals from inadvertent intrusion," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.43, 2022, "Protection of individuals during operations," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.44, 2022, "Stability of the disposal site after closure," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 61.55, 2022, "Waste classification," *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 71.4, 2022, “Definitions,” *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 820, 2022, “Procedural Rules for DOE Nuclear Activities,” *Code of Federal Regulations*, Office of the Federal Register.

10 CFR 835, 2022, “Occupational Radiation Protection,” *Code of Federal Regulations*, Office of the Federal Register.

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Appendix A

Comparability of U.S. Department of Energy and U.S. Nuclear Regulatory Commission Requirements for Low-Level Radioactive Waste Disposal

Appendix A

Comparability of U.S. Department of Energy and U.S. Nuclear Regulatory Commission Requirements for Low-Level Radioactive Waste Disposal

A-1 INTRODUCTION

The purpose of this appendix is to (1) identify U.S. Department of Energy (DOE) and U.S. Nuclear Regulatory Commission (NRC) performance objectives for disposal of low-level radioactive waste (LLW) from the Calcined Solids Storage Facility and (2) show that DOE and NRC LLW disposal requirements are comparable. Key points in this evaluation are:

- Requirements for LLW disposal are embodied in sets of performance objectives for the waste disposal facility
- The DOE performance objectives are described in DOE M 435.1-1 Chg 3, "Radioactive Waste Management Manual"
- The NRC performance objectives are described in 10 CFR 61, Subpart C, "Performance Objectives"
- Both DOE and NRC have provisions for imposing additional requirements for LLW disposal
- DOE and NRC performance objectives for LLW disposal are comparable.

Table A-1 presents a crosswalk comparing DOE and NRC requirements for LLW disposal.

Table A-1. Crosswalk between Nuclear Regulatory Commission and U.S. Department of Energy requirements for low-level radioactive waste disposal.

Section	Title	Description	Section	Title	Description	Discussion
10 CFR 61, Subpart C			DOE M 435.1-1 Chg 3			
10 CFR 61.40	General requirement	Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44.	Section IV.P(1)	Performance Objectives	LLW disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed of after September 26, 1988.	NRC requirements in 10 CFR 61.40 are nearly identical to those of the DOE general requirement. The DOE requirement adds the concept of maintenance, which is implicit in the NRC requirement. The DOE requirement does not mention control after closure, but this concept is embodied in DOE requirements for closure, specifically DOE M 435.1-1 Chg 3, Section IV.Q (2)(c), which requires DOE control until it can be shown that release of the disposal site for unrestricted use will not compromise DOE requirements for radiological protection of the public. The DOE general requirement for LLW disposal and the NRC general requirement of 10 CFR 61.40 are therefore comparable.
10 CFR 61.41	Protection of the general population from releases of radioactivity	Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment ALARA.	Section IV.P(1)	Performance Objectives	(a) Dose to representative members of the public shall not exceed 25 mrem (0.25 mSv) in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and its progeny in air. (b) Dose to representative members of the public via the air pathway shall not exceed 10 mrem (0.10 mSv) in a year total effective dose equivalent, excluding the dose from radon and its progeny. (c) Release of radon shall be less than an average flux of 20 $pCi/m^2/s$ (0.74 $Bq/m^2/s$) at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L (0.0185 Bq/L) of air may be applied at the boundary of the facility.	DOE uses more current radiation protection methodology, consistent with that used in NRC radiation protection standards in 10 CFR 20, "Standards for Protection Against Radiation." Because the NRC has not revised 10 CFR 61.41 to reflect the more current methodology in 10 CFR 20, DOE's requirements and those in 10 CFR 20 differ slightly from those in 10 CFR 61.41. However, the resulting allowable doses are comparable, as the NRC has acknowledged (NRC 2005). Both NRC and DOE use a performance assessment to assess whether the dose limit will be met. The DOE requirements go beyond this NRC performance objective by specifying an assessment of the impacts of LLW disposal on water resources [i.e., DOE M 435.1-1 Chg 3, Section IV.P (2)(g)]. The NRC requirement includes maintaining releases to the environment ALARA. Although this requirement is not included in the DOE performance objective, it is included in the performance assessment requirements [i.e., DOE M 435.1-1 Chg 3, Section IV.P (2)(f)].
10 CFR 61.42	Protection of individuals from inadvertent intrusion	Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.	Section IV.P(2)(h)	Performance Assessment	For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the LLW disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 mrem (1 mSv) in a year and 500 mrem (5 mSv) total effective dose equivalent excluding radon in air.	The DOE LLW disposal requirement that the performance assessment include an assessment of the impacts on a person inadvertently intruding into the disposal facility is more stringent than the NRC requirement. The NRC waste classification system is based on intruder calculations using a 500-mrem per year dose limit (NRC 1982). The DOE requirement uses a 100 mrem in a year limit for chronic exposures and a 500-mrem limit for acute exposures.
10 CFR 61.43	Protection of individuals during operations	Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in Part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by Section 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures ALARA.	Section I.1D(13)	Radiation Protection	Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR 835, "Occupational Radiation Protection," and DOE O 5400.5, "Radiation Protection of the Public and the Environment" [now DOE O 458.1 Chg 4].	The ALARA concept is an integral part of DOE radiation and environmental protection programs. DOE requirements for occupational radiological protection are addressed in 10 CFR 835, "Occupational Radiation Protection," and similar requirements for radiological protection of the public and the environment are addressed in DOE O 458.1 Chg 4. The NRC 10 CFR 61.43 requirement references 10 CFR 20, "Standards for Protection Against Radiation," which contains similar radiological protection standards for workers and the public.

Table A-1. (continued).

Section	Title	Description	Section	Title	Description	Discussion
10 CFR 61, Subpart C			DOE M 435.1-1 Chg 3			
10 CFR 61.44	Stability of the disposal site after closure	The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.	Sections IV.Q(1)(a) and (b) and IV.Q(2)(c)	Disposal Facility Closure Plans and Disposal Facility Closure	<u>Disposal Facility Closure Plans [Section IV.Q(1)(a) and (b)]</u> A preliminary closure plan shall be developed and submitted to DOE Headquarters for review with the performance assessment and composite analysis. The closure plan shall be updated following issuance of the disposal authorization statement to incorporate conditions specified in the disposal authorization statement. Closure plans shall: (a) Be updated as required during the operational life of the facility. (b) Include a description of how the disposal facility will be closed to achieve long-term stability and minimize the need for active maintenance following closure and to ensure compliance with the requirements of DOE O 5400.5, "Radiation Protection of the Public and the Environment" [now DOE O 458.1 Chg 4]. <u>Disposal Facility Closure (Section IV.Q(2)(c))</u> Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall continue until the facility can be released pursuant to DOE O 5400.5, "Radiation Protection of the Public and the Environment" [now DOE O 458.1 Chg 4].	The DOE LLW disposal requirements address long-term stability of the site by requiring a description of how closure will achieve stability in the closure plan and by a description of how closure will minimize the need for active maintenance following closure [DOE M 435.1-1 Chg 3, Section IV.Q (1)(b)]. Additionally, one of the performance assessment requirements [DOE M 435.1-1 Chg 3, Section IV.P (2)(c)] states: "Performance assessments shall address reasonably foreseeable natural processes that might disrupt barriers against release and transport of radioactive materials." Thus, the performance assessment will include a projection of the long-term stability of the site, considering reasonably foreseeable natural processes such as erosion, degradation of waste packages, etc.

ALARA as low as reasonably achievable
DOE U.S. Department of Energy
LLW low-level radioactive waste
NRC U.S. Nuclear Regulatory Commission

A-2 REFERENCES

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Appendix B

DOE Closure Approach for the CSSF

Appendix B

DOE Closure Approach for the CSSF

The purpose of this appendix is to outline the approach used by the U.S. Department of Energy (DOE) for the Calcined Solids Storage Facility (CSSF) bin sets (including integral equipment), transport lines, and residual calcine to meet the closure criteria specified in the closure plan provided in Volume 22 Hazardous Waste Management Act (HWMA)/Resource Conservation and Recovery Act (RCRA) Partial Permit for Storage at the Calcined Solids Storage Facility at the Idaho Nuclear Technology and Engineering Center on the Idaho National Laboratory (INL), EPA ID No. ID4890008952 (PER-114).

B-1 INTRODUCTION

The CSSF is a HWMA/RCRA-permitted storage facility. Therefore, closure of the CSSF will be conducted in accordance with requirements of the closure plan included in the State-approved HWMA/RCRA permit. The closure plan presented in the partial permit specifies the closure performance standards for clean closure; however, if clean closure is not possible, closure will be performed under a contingent landfill closure plan. Multiple facilities at the INL Site have been closed under HWMA/RCRA, and a similar approach will be followed for closure of the bin sets. The following outlines DOE's strategy for RCRA closure of the CSSF.

B-2 CLOSURE APPROACH

B-2.1 Closure Performance Standards

In accordance with HWMA/RCRA closure performance standards, to achieve "clean" closure, the CSSF bin sets will be closed in a manner that minimizes the need for further maintenance; controls, minimizes, or eliminates, to the extent necessary, exposure of hazardous waste to protect human health and the environment; and complies with the closure requirements for tank closures. The closure performance standard for tank systems requires removal or decontamination of all waste and waste residue from the tanks, piping, ancillary equipment, and surfaces of the system.

B-2.2 Closure Integration with the Comprehensive Environmental Response, Compensation, and Liability Act

Like previous closures of major facilities at the INL Site, the DOE approach for HWMA/RCRA closure of the bin sets will be integration with Comprehensive Environmental Response, Compensation, and Liability (CERCLA) as part of a planned non-time-critical removal action in accordance with the Federal Facilities Agreement and Consent Order (DOE-ID 1991).

B-2.3 Closure Plan

Because each bin set is separate and different in design, DOE will be conducting partial closure for each bin set. The existing closure plan in the Volume 22 CSSF Permit (PER-114) will be modified to include the closure boundaries (defining those components of the bin set that are RCRA-permitted and subject to closure). Other modifications to the closure plan may include changes to the facility structures and components, such as removing most or all of the minor facilities associated with each bin set as well as parts of the calcine vault to facilitate access to calcine storage bins. The permit modification request will include the final closure criteria to meet the clean closure performance standards for the bin sets. This permit modification request will be submitted to the State of Idaho for review and approval. A contingent landfill closure plan will be submitted to the State at the same time.

B-2.4 Closure and CERCLA Actions

Changes to the bin sets and removal or modification of permitted equipment will not proceed until the State approval of the permit modification. Upon approval of the permit modification, DOE will submit to the State a notification of intent to begin RCRA closure.

After completion of calcine retrieval and transfer operations, a remote visual inspection of the bins will be performed to determine if the closure criteria have been met. After it is determined the closure criteria have been met, a qualified professional engineer, the operator, and DOE (owner) will provide a certification of closure to the Idaho Department of Environmental Quality stating that the bin sets have been closed in accordance with the approved closure plan. DOE must obtain the Idaho Department of Environmental Quality's acceptance of the certification to finalize the closure. As mentioned previously, if clean closure cannot be achieved as described in the closure plan in the permit, closure as a landfill will be performed per the contingent landfill closure plan.

The closure is completed as non-time-critical removal action under authority of the General Action Memorandum (DOE-ID 2021a). Following closure in accordance with the RCRA requirements, decisions and actions regarding final capping, monitoring, and long-term maintenance of the closed facilities will be coordinated with the CERLCA program. DOE will prepare an engineering evaluation and cost analysis as well as a separate action memorandum to support the selection of a final end state for the CSSFs. DOE will place the CSSF in its final end state in accordance with the action memorandum, and a removal action report will be completed. If there are residual risks that require institutional controls, DOE will complete the new site identification process under Operable Unit 10-08.